PLAN_OF_.MLM

From: William Bearden, RII
To: GGA GANDERSON, RII
Date: 11/6/96 7:56am
Subject: Plan of Day

I am assigned as the primary maintenance inspector for St. Lucie. As such I have to trend verious info on St Lucie for the region. FP&L has me on distribution for their monthly reports but some of the info I need is not contained in their monthly reports. I believe the needed info is probably contained in the POD (plan of day) handout that they provide the residents every day (I am at Turkey Point this week and their POD has this info). The residents probably review this handout and discard it every day. I only need one about once a month. Would you ask Joel for a recent one (exact date unimportant) and mail it to me in the region so I could verify that it has the needed info.

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10/24/96

~~Pre-Decisional~~
Semiannual Plant Performance Assessment
St. Lucie 1 and 2

Current SALP Assessment Period: 1/7/96 through 3/97

	Last SALP Rating 1/2/94 - 1/6/96	Previous SALP Rating 5/3/92 - 1/1/94
Operations	2	1
Maintenance	2	1
Engineering	1	1
Plant Support	1	1

INPO assessment July 1995 - Category 1

I. Performance Overview

Since July 1995, there have been a series of events that led to questioning the plant's overall performance. An NRC root cause effort determined that, in addition to procedural adherence/adequacy weaknesses, the licensee suffered from weaknesses in both interfaces across organizational lines and corrective actions. The SALP board concluded that performance in the areas of Operations and Maintenance had declined to level 2. Since the SALP board, additional examples of declined performance were noted. These have included:

- Significant operator inattentiveness which resulted in the overdilution event on January 22, 1996, highlighted the recent large number of personnel errors and lack of command and control in the control room (SL3, CP).
- On February 22, 1996, a dropped CEA and an ensuing Unit 1 shutdown resulted in the declaration of an unusual event. During the shutdown, main feedwater regulating valve instabilities resulted in operators manually tripping the unit.
- On February 24, a containment radiation monitor was rendered inoperable for two days due to an improper valve lineup following a grab sample. As a result, the unit was started up without this TS-required component available. Several instances of failure to follow procedures and operator inattention led to the extended period of inoperability (SL4).
- On May an inspection indicated that a significant number of shifts had been worked with fire brigade members which were not medically qualified. A breakdown in the tracking of this data resulted from a key individual being laid off.
- On May 12, fuel movement was commenced on Unit 1 without only 1 of 2 wide range NI channels available. Operators performing a surveillance test on the inoperable channel did not coordinate

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with the refueling center properly. Additionally, the fuel offload was commenced without incorporating requirements from the spent fuel pool heat load calculation into the appropriate operational procedures.

- On June 6, Unit 2 was manually tripped due to high generator gas temperature. Root cause was a screw which vibrated loose and resulted a temperature control valve feedback arm falling free of its connection. This failure mode had been encountered before.

- On June 16, an inspection identified that 56 individual violations of overtime guidelines had occurred on the part of 4 individuals over a 30 day period. Evidence also existed that employees were regularly working longer hours than those reported on their timesheets.

- On July 20, Unit 1 experienced a loss of charging flow when due to a mispositioned board selector switch, both operating pumps stopped on a faulty indication of high pressurizer level, caused by I&C errors.

- A number of engineering-related problems have been identified, to include:

    - A number of annunciator response procedures which were inaccurate due to a failure to update them when design modifications took place.

    - Four similarly miswired nuclear instrumentation channels due to errors in control wiring diagrams implemented during a modification. The condition was identified at full power and resulted in an entry into TS 3.0.3.

    - Nonconservative errors were identified in auxiliary feedwater actuation system setpoints due to a failure to incorporate as-built data in instrument calibration calculations.

- Maintenance overtime usage was found excessive in that four individuals were responsible for 56 examples of non-approved exceedences of Technical Specification overtime guidelines.

- On August 14, glue was found in key lock switches on both units' hot shutdown panels, rendering the switches inoperable. The tampering instances appeared to be additional examples of padlocks and door locks which were identified in July.

In addition to the inspection findings above, the inspectors have noted a general low state of morale. A great number of both management and non-management employees have expressed concern with regard to the company's ongoing downsizing effort. The general feeling is that, unlike Turkey Point, which was afforded the budget and time to improve prior to downsizing, St. Lucie is expected to improve AND downsize simultaneously.

II. Functional Area Assessment - Operations

A. Assessment

Performance in Operations appears to have leveled. At the time of the last PPR, operator errors and operational events were on the increase. In the past six months, examples of improved operator attention to detail and conservative decision-making have been identified. Strong performance was identified in the area of reduced inventory operation. Weaknesses were identified in the areas of procedural quality and operability maintenance and decision-making. Improvements in control room environment, formality, and communications have been noted. The licensee has appeared to make inroads in the areas of operator self-assessment and documentation of adverse conditions.

B. Basis

1. Attention to Detail and Conservative Decision-Making

- Non-licensed operators were successful in identifying two cases of inadvertent containment radiation monitor inoperability and a breach in a fire-rated assembly.

- After a non-conservative decision which resulted in a late declaration of an NOUE for CVCS system leakage, operators have declared three NOUEs for similar circumstances (CVCS leakage outside containment which could not be quickly quantified). Management has been effective in encouraging conservative decision-making.

- Entry into a shutdown action statement when 4 Unit 2 control rods would not respond electrically.

- Five entries into reduced inventory during the period without error.

- Timely trip of Unit 1 due to apparent gas buildup in the 1B transformer.

- Terminating a Unit 1 startup due to predictions that xenon decay would invalidate the estimation of critical conditions.

2. Weaknesses in Procedures and Maintenance of Operability

- Numerous errors identified in annunciator response procedures.

- Full core offload began on Unit 1 without incorporating requirements from the fuel pool heat load calculation into operational procedures.

- Operator aids found in the field did not agree with procedural requirements for the tasks they described.

- Unit 1 fuel movement began without the required 2 operable channels of wide range nuclear instruments due to the performance of a surveillance test.

- Clearance hung during the Unit 1 outage resulted in inoperability of audible count rate in containment.

3. Other Observations

- Good performance was noted during a Unit 2 downpower due to low turbine auto-stop oil pressure, a Unit 2 trip due to a failed turbine cooling water valve, several startups, and fuel movements in Unit 1 containment.

- Poor performance was noted in the use of a single operator for fuel movement in the spent fuel pool, in the control of keys for PORV operation outside of the control room, in the control of backup charging pump selector switch position, and in performing a test of a turbine-driven AFP which resulted in a pump trip.

- Equipment failures continue to challenge operators, with the occurrence of two manual trips per unit this calendar year due to equipment failures.

C. Future Inspections

The high number of allegations and an increase in resident involvement with engineering activities has reduced the available time for core Operations inspections. The site has been brought to an N+1 staffing level; however, qualification of the new resident is not anticipated until February, 1997. Additionally, both assigned Resident Inspectors will be attendir[illegible] training at TTC for three weeks in October/November. An acti[illegible] esident has been arranged for the period; however, inspection at the N+1 level will not be possible until the end of the current SALP cycle (March 1997). Consequently, Senior Resident and Resident Inspectors objectivity visits, involving control room observations, are planned. Additionally, DRS inspections of the licensee's procedure development and approval process, which has recently changed in an effort to improve procedure quality, are planned.

III. Functional Area Assessment - Maintenance

A Assessment: An increase in personnel errors and equipment problems was noted. The majority of the equipment problems are BOP related. For the most part the licensee considered safety in establishment of goals and for monitoring of systems and

components in the maintenance rule. The maintenance program is adequate.

B. Basis:

The maintenance area was rated good overall the last SALP period. The last PPR indicated a problem with EDGs and procedure problems.

The plant matrix indicates 12 equipment failures, 12 personnel errors and 3 procedure problems during the last 6 months.
Examples of personnel errors were:
- 8/31/96 Improper use of M&TE for meggering NI cables
- 8/3/96 Freeze seal left unattended
- 7/30/96 3 of 4 linear NI channels found miswired
- 7/20/96 2 charging pumps tripped due to erroneous level signal

Power Reduction caused by Equipment Failures in the last 6 months:

- 4/20/96: Unit 2 - Turbine Stop Oil orifice blockage.
- 4/09/96: Unit 2 - Downpower due to Circ Water Piping leakage.
- 5/24/96: Unit 2 - Downpower due to CEDM problems.
- 5/31/96: Unit 2 - Downpower due to MSR TCV closure due to blown fuse.
- 6/06/96: Unit 2 - Reactor trip resulting from high generator H2 temp due to failed TCV.
- 6/22/96: Unit 2 - Downpower due to 2B FRV Controller problems.
- 7/23/96: Unit 1 - Manual trip due to turbine maintenance.

Maintenance Backlog:
- Non-outage corrective maintenance backlog: 1101 items, no significant changes since beginning of year.
- Overdue Preventive Maintenance Backlog: 30 Maintenance PMs were late

Maintenance Rule A(1) systems: 6 systems
- EDG governors, EDGs, 4.16 KV AC safety related breakers, PORVs, C AFW, and RCP seals.

C Future Inspections:

- Maintenance Rule follow-up: 62703 (RI) - 1 week
- ISI inspection: 73753 (core) - 1 week
- Integrated S/G Replacement Inspection: 73753 (RI) - 3 weeks

IV. Functional Area Assessment - Engineering

A. Assessment

St. Lucie received a SALP 1 rating during the SALP period that ended January 6, 1996. The licensee has declined in performance during this PPR period (March-September 1996) due to problems with configuration management/design control and a failure to identify an USQ.

B. Basis

PIM TRENDS/ISSUES: The trend indicated was for configuration management as described in design control issues below and an issue for failure to identify an USQ for a 50.59 evaluation (September 19, 1996).

ENFORCEMENT: Letter of violation issued September 19, 1996. One level III and two level IVs in the area of USQ and configuration management.

DESIGN CONTROL ISSUES: In enforcement identified two problems, one which failed to coordinate design changes to operating procedures with three examples: 1) Set point change to low level alarm in the Hydrazine tank, 2) removal or ICW lube water piping and did not change abnormal procedure which affects operator actions, and 3) disabled a steam dump valve annunciator without changing the annunicator response procedure. The second problem identified the failure to change ICW drawings after a modification (All three examples September 19, 1996).

OPERATING FOCUS: The licensee took steps to prevent tube failure of its steam generators on Unit 1 by plugging approximately 2300 tubes. These steam generators will be replaced in fall 1997 outage.

MAJOR INITIATIVES: Unit 2 outage 4/15/97 97, Unit 1 S/G replacement outage fall '97

FSAR INITIATIVES: A review has been conducted of approximately one-third of the FSAR (July 1996 inspection). This review was performed mostly on Unit 1 and was performed on text material and not for curves and tables. No USQ or operability problems were found. Approval pending for reviewing remaining part of FSAR.

DBD/R: A Design Basis Documentation was performed for 20 design basis documents. The program was completed near the end of 1995.

C. Future Inspections

Engineering-9 weeks, basis: Evaluate new engineering organization, FSAR project, configuration management and followup on design control issues.

V. Functional Area Assessments - Plant Support

A. Assessment

The last SALP cycle ended 1/6/96. Plant Support was Category 1. The licensee continues to maintain a satisfactory level of performance in the area of Plant Support. Some decline in Radiation Protection has been noted due to the loss of control of contaminated tools and exceeding dose goals. Emergency Preparedness ongoing inspection indicates a decline in

performance. Hurricane preparations for hurricane Bertha were conservative. Overall, site security has been adequate. Training and qualification noted as a strength and management observed to be aggressive in pursuing issues, but not aggressive in doing indepth review of events. Implementation of the fire protection program continued to be satisfactory.

B. Basis

Radiation Protection

NCV for failure to control contaminated tools used in RCA (96-04, p 45)

Violation (repeat of above NCV) for numerous examples of failure to control contaminated tools. (96-09, p 25)

Internal and external exposures below 10 CFR Part 20 limits. (96-04, p 45 and 96-04, p 23) (1996 dose levels?).

1995 dose was 412 person-rem. Unplanned maintenance and rework caused 1995 dose goal of 283 person-rem to be exceeded by 129 person-rem. (96-04, p 50)

Rad Techs decreased from 32 to 30 and 2 supervisors lost (96-04, p 48)

Decon staff reduced from 22 to 12 persons. Levels of contaminated equipment and materials increasing. (96-04, p 46)

Good radiological housekeeping and controls. (96-09, p 28)

The total area contaminated was at 250 ft^2. (96-04, p 47)

Licensee accreditation of the FP&L DADs a good example of Radiation Protection staff's technical capabilities. (96-04, p 44)

Emergency Preparedness

Conservative actions taken to prepare for Hurricane Bertha.(96-11, p 3)

Security

Failure to report a confirmed tampering event within one hour, which resulted in a violation.

Two events in prior to the above tampering event were documented as tampered or unauthorized work, but management failed to notify security of these events.

Numerous problems discovered by a QA audit determined the FFD program to be weak.

Fire Protection

A backup fire pump was installed to replace an out of service fire pump.

C. Future Inspections

Inspections	Rationale
Health Physics	(SALP 1 decline - maintain; watch)
Operational HP(83750)	2-Inspections with focus on procedure compliance; rework doses
Effl/RadWast(84/86750)	3-inspections with focus on accident/process monitor installation & maintenance
TI 133 Rad Waste	Combine with 86750
Emergency Preparedness Prog. (82701)	1-Inspection with focus on Self-Assessment results Regional Initiative inspection on allegation followup (3 weeks, 2 inspectors)
Security Prog (81700)	Core Insp. to review security audits, corrective actions, management support and effectiveness, and review protected area detection equipment
Sec. Prg/FFD (81700/81502)	One regional initiative to followup on tampering and FFD issues
Fire Protection	None

VI. Attachments

1. Power Profile
2. Plant Issues Matrix
3. Current NRC Performance Indicators
4. Licensee Organization Charts
5. Allegation Status
6. Enforcement History
7. Major Assessments
8. Recent Generic Issues Status List

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

September 11, 1996

MEMORANDUM TO: All NRR Project Managers
and Project Directors

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

SUBJECT: SPENT FUEL STORAGE POOL ACTION PLAN ISSUES: ISSUANCE OF FINAL STAFF REPORT AND NOTIFICATION OF PLANT-SPECIFIC, SAFETY ENHANCEMENT BACKFIT ANALYSES

1.0 Background

Since October 1994, the Plant Systems Branch has been reviewing spent fuel pool technical issues under the "Task Action Plan for Spent Fuel Storage Pool Safety." The detailed spent fuel pool design information gathered by Project Managers as part of the spent fuel licensing basis survey was used by the technical staff as part of the action plan review. The Plant Systems Branch has completed the majority of the work outlined in the action plan. The staff has prepared a report to the Commission, dated July 26, 1996, on its findings and recommendations (Attachment 1).

As a result of its review, the technical staff identified ten categories of design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. The staff determined that these design features existed at twenty-nine sites (encompassing forty seven reactors). The NRC staff plans to address seven of the ten categories of design features by pursuing regulatory analyses for safety enhancement backfits on a plant-specific basis pursuant to 10 CFR 50.109 at the operating reactor sites possessing one or more of these design features. Consistent with 10 CFR 50.109(a)(3), the regulatory analyses for safety enhancement backfits will consider whether modifications of the plant design to address the plant-specific design features identified by the NRC staff could provide a substantial increase in the overall protection of public health and safety and could be justified on a cost benefit basis. Plants identified as having design features in the remaining three categories will be evaluated more closely because the NRC staff's conclusions on the adequacy of design in these areas are dependent upon information not collected during the recent staff survey of spent fuel pool design and operation. The affected plants are described in the report and are listed in Attachment 2.

Staff plans to perform the plant-specific backfit analyses are under development. However, to facilitate the reviews, the staff is going to

CONTACT: Joe Shea, NRR
415-1428

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provide affected licensees with an opportunity to comment on (1) the accuracy of the staff's understanding of the plant design, (2) the safety significance of the design concern, (3) the cost of potential modifications to address the design concern, or (4) the existing protection from the design concern provided by administrative controls or other means.

2.0 Actions for Project Managers of Affected Plants

On July 31, 1996, Project Managers of affected plants distributed copies of the report to the respective licensees. To provide an opportunity for comments as described above, Project Managers of affected plants are requested to send a letter to their licensee that (1) formally transmits the July 26, 1996 report, (2) describes the plant-specific design features of concern and (3) provides an opportunity to submit comments.

To facilitate issuance of letters to affected licensees, plant-specific sample letters have been prepared and are provided in Attachment 3 of this memorandum for each affected facility. Project Managers are requested to issue letters for their plants by September 27, 1996 and include Joe Shea on distribution for all outgoing letters.

The schedule for performing the plant-specific backfit analyses for each plant is still under development. Project Managers of affected plants will be informed in the future of detailed plans and schedules, based in part on any comments received from licensees.

3.0 Actions for Project Managers of Non-Affected Plants

Project Managers for all other plants (i.e., all plants other than those listed in Attachment 2) are requested to forward the report to their licensees using the sample letter provided as Attachment 4.

4.0 Administrative Information

Time spent by Project Managers in issuing the plant-specific letters to both affected and non-affected licensees should be charged to TAC M88094.

Since the forwarding letters do not request a response, Office of Management and Budget clearance is not need for these letters.

If you have any questions regarding this effort, do not hesitate to contact Joe Shea at 415-1428.

Attachments: (1) Memo to the Commission, from J. Taylor, "Resolution of Spent Fuel Storage Pool Action Plan Issues," dated July 26, 1996
(2) List of Affected Plants
(3) Sample Letters for Affected Plants
(4) Sample Letter for All Plants Other than those listed in Attachment 2

*Previously Concurred

OFFICE	PDI-2/LA	PDI-2/PM	PDI-2/PD	DRPW/D*	DRPE/D*	SPLB/C*	DSSA*
NAME	MO'Brien	JShea	JStolz	JRoe	SVarga	LMarsh	GHolahan
DATE	9/12/96	9/12/96	9/12/96	09/09/96	09/11/96	09/06/96	09/10/96

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

July 26, 1996

MEMORANDUM TO: Chairman Jackson
Commissioner Rogers
Commissioner D'cus

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL ACTION PLAN ISSUES

In a meeting with Chairman Jackson on February 1, 1996, regarding spent fuel pool issues, the staff committed to prepare a course of action for resolving significant issues developed through the staff's Task Action Plan for Spent Fuel Storage Pool Safety. The significant issues examined within the framework of that plan were the reliability of spent fuel pool decay heat removal and the maintenance of an adequate spent fuel coolant inventory in the spent fuel pool. The staff was also directed to identify plant-specific and generic areas for regulatory analyses in support of further regulatory action.

The staff has completed its review and evaluation of design features related to the spent fuel pool associated with each operating reactor. Details of the staff's review and evaluation are presented in the attached report. The staff classified operating reactors on the basis of specific design features associated with the spent fuel pool in the following areas: coolant inventory control, coolant temperature control, and fuel reactivity control.

In comparing design features with NRC design requirements and guidance, the staff determined that design features related to coolant inventory control and reactivity control were more consistent with NRC guidance than were design features associated with coolant temperature control. The staff concluded that coolant inventory control design features were more consistent with present guidance because the staff had issued explicit guidance for prevention of coolant inventory loss in the form of design criteria before it issued most construction permits for currently operating reactors. These criteria are documented in plant specific AEC Design Criteria in each affected facility's safety analysis report; in the General Design Criteria of Appendix A to 10 CFR Part 50, which became effective in 1971; and in Safety Guide 13 (now Regulatory Guide 1.13), "Spent Fuel Storage Facility Design Basis," which was issued in March 1971. The staff concluded that reactivity control provisions are consistent because nearly all operating reactors have increased their spent fuel pool storage capacity since the NRC issued specific guidance for reactivity control, and such increases involve design and analysis of new fuel storage racks for criticality prevention. Conversely, the NRC staff did not issue specific guidance on the design of spent fuel pool cooling systems until the issuance of the Standard Review Plan (NUREG-75/087) in 1975, which was

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Attachment 1

after the issuance of most construction permits for currently operating reactors, and spent fuel storage capacity increases have seldom involved a sufficient increase in decay heat generation that an expanded cooling system was warranted.

The staff has found that existing structures, systems, and components related to storage of irradiated fuel provide adequate protection for public health and safety. Protection has been provided by several layers of defenses that perform accident prevention functions (e.g., quality controls on design, construction, and operation), accident mitigation functions (e.g., multiple cooling systems and multiple makeup water paths), radiation protection functions, and emergency preparedness functions. Design features addressing each of these areas for spent fuel storage have been reviewed and approved by the staff. In addition, the limited risk analyses available for spent fuel storage suggest that current design features and operational constraints cause issues related to spent fuel pool storage to be a small fraction of the overall risk associated with an operating light water reactor. Notwithstanding this finding, the staff has reviewed each operating reactor's spent fuel pool design to identify strengths and weaknesses, and to identify potential areas for safety enhancements.

The staff plans to address certain design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. We intend to pursue regulatory analyses for safety enhancement backfits on a plant-specific basis pursuant to 10 CFR 50.109 at the small number of operating reactors possessing each particular identified design feature. The specific plans for safety enhancement backfits and their bases are described in the attached report. Because of the relatively low safety significance of these issues, the staff recognizes that some, or all, of these potential enhancements may not pass the backfit tests.

The staff will provide the attached report to the licensees of all operating reactors. The staff intends to request that those licensees identified in the report for plant-specific regulatory analysis verify the applicability of the staff's findings and conclusions. The staff will also request that licensee's provide, on a voluntary basis, their perspective on the potential increase in the overall protection of public health and safety and information regarding the cost of potential modifications to address the design features identified in the staff report. Staff reviews of potential plant-specific or generic backfits will be appropriately coordinated with the Committee to Review generic Requirements (CRGR).

The staff also plans to address issues relating to the functional performance of spent fuel pool decay heat removal, as well as the operational aspects related to coolant inventory control and reactivity control, through expansion of the proposed, performance-based rule, "Shutdown Operations at Nuclear Power Plants" (10 CFR 50.67), to encompass fuel storage pool operations.

Concurrent with the regulatory analyses for the potential safety enhancements, the staff will develop guidance for implementing the proposed rule for fuel storage pool operations at nuclear power plants. The staff will also develop plans to improve existing guidance documents related to design reviews of spent fuel pool cooling systems. In addition, the staff will issue an information notice as a mechanism for distributing information in areas where regulatory analyses do not support rulemaking or plant-specific backfits.

Attachment: Plan for Resolving Spent Fuel Storage Pool Action Plan Issues

PLAN FOR RESOLVING SPENT FUEL STORAGE POOL ACTION PLAN ISSUES

1.0 INTRODUCTION

The NRC staff developed and implemented a generic action plan for ensuring the safety of spent fuel storage pools in response to two postulated event sequences involving the spent fuel pool (SFP) at two separate plants. The principal safety concerns addressed by the action plan involve the potential for a sustained loss of SFP cooling and the potential for a substantial loss of spent fuel coolant inventory that could expose irradiated fuel.

The first postulated event sequence was reported to the NRC staff in November 1992 by two engineers, who formerly worked under contract for the Pennsylvania Power and Light Company (PP&L). In the report, the engineers contended that the design of the Susquehanna station failed to meet regulatory requirements with respect to sustained loss of the cooling function to the SFP that could result from a loss-of-coolant accident (LOCA) or a loss of offsite power (LOOP). The heat and water vapor added to the reactor building atmosphere by subsequent SFP boiling could cause failure of accident mitigation or other safety equipment and an associated increase in the consequences of the initiating event. Using probabilistic and deterministic methods, the staff evaluated these issues as they related to Susquehanna and determined that public health and safety were adequately protected on the basis of existing design features and operating practices at Susquehanna (see attached safety evaluation for additional details). However, the staff also concluded that a broader evaluation of the potential for this type of event to occur at other facilities was justified.

The second postulated event sequence was based on an actual event that occurred at Dresden 1, which is permanently shut down. This plant experienced containment flooding because of freeze damage to the service water system inside the containment building on January 25, 1994. Commonwealth Edison reported that the configuration of the spent fuel transfer system between the SFP and the containment similarly threatened SFP coolant inventory control. At Dresden Unit 1, portions of the spent fuel transfer system piping inside the containment could have burst due to freezing at an elevation that would drain the spent fuel coolant to a level below the top of stored irradiated fuel in the SFP. A substantial loss of SFP coolant inventory could lead to such consequences as high local radiation levels due to loss of shielding, unmonitored release of radiologically contaminated coolant, and inadequate cooling of stored fuel. The staff concluded that the potential for this type of event to occur at other facilities should be evaluated.

Finally, the action plan itself called for a review of events related to wet storage of irradiated fuel. From this review and information from the two postulated event sequences that prompted development of the action plan, the staff identified areas to evaluate for further regulatory action. Design information to support this evaluation was developed through four onsite assessments, a safety analysis report review for several operating reactors, and the staff's survey of refueling practices completed in May 1996.

ATTACHMENT

Because the safety of fuel storage in the SFP is principally determined by coolant inventory, coolant temperature, and reactivity, the staff divided its evaluation into those areas. Coolant inventory affects the capability to cool the stored fuel, the degree of shielding provided for the operators, and the consequences of postulated fuel handling accidents. Coolant temperature affects operator performance during fuel handling, control of coolant chemistry and radionuclide concentration, generation of thermal stress within structures, and environmental conditions surrounding the SFP. Spent fuel storage pools are designed to maintain a substantial reactivity margin to criticality under all postulated storage conditions. In order for operators to promptly identify unsuitable fuel storage conditions, the spent fuel storage facility must have an appropriate means to notify operators of changes to the conditions in the SFP.

2.0 REGULATORY FRAMEWORK FOR SPENT FUEL POOL STORAGE

The NRC acceptance criteria for the design of structures, systems, and components related to the SFP has evolved from case-by-case reviews for early plants to the present guidance of the Standard Review Plan (SRP) - NUREG-0800 - and regulatory guides, and the requirements of the General Design Criteria (GDC) of Appendix A to 10 CFR Part 50, as implemented by 10 CFR 50.34. In addition, the increased use of high density storage racks to expand onsite irradiated fuel storage capability has required nearly all operating reactor licensees to request license amendments related to fuel storage. Consequently, the design of certain structures, systems, and components related to the SFP may vary among a group of plants, depending on the stage of evolution of acceptance criteria developed by the staff and the deviations from these criteria the staff found acceptable.

The Atomic Energy Commission (AEC) developed design criteria in the mid-60s that were used as guidance in evaluating plant design. These criteria were continually revised so that a consistent basis for acceptable design practices for the SFP was not established. As an example, Criterion 25 from a version of the AEC design criteria dated November 5, 1965, stated:

> The fuel handling and storage facilities must be designed to prevent criticality and to maintain adequate shielding and cooling under all anticipated normal and abnormal conditions, and credible accident conditions. Variables upon which the health and safety of the public depend must be monitored.

These AEC design criteria evolved into the GDC presented in Appendix A to 10 CFR Part 50, which the AEC issued in 1971. Criterion 61 of the GDC requires, in part, that the fuel storage system be designed with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal and be designed to prevent significant reduction in coolant inventory under accident conditions. Criterion 62 provides requirements for prevention of criticality, and Criterion 63 specifies requirements for systems to monitor fuel storage systems.

In 1970, the AEC developed and began issuing safety guides to make available specific methods acceptable to the staff for implementing regulations. Regulatory Guide 1.13 (formerly Safety Guide 13), "Spent Fuel Storage Facility Design Basis," was used as guidance in the licensing evaluation of many spent fuel storage facilities. Regulatory Guide 1.13 described an acceptable method of implementing General Design Criterion 61 in order to:

(1) Prevent loss of water from the fuel pool that would uncover fuel.

(2) Protect fuel from mechanical damage.

(3) Provide the capability for limiting the potential offsite exposures in the event of a significant release of radioactivity from the fuel.

Regulatory Guide 1.13 has no specific guidance for evaluating criticality prevention measures or SFP cooling system design features.

The SRP gives specific acceptance criteria derived from applicable GDC and other NRC regulations, and a method acceptable to the staff to demonstrate compliance with those acceptance criteria for various structures, systems, and components at commercial light water reactors. The SRP was first issued in 1975 as NUREG-75/087, and NUREG-0800 was issued in 1981. The SRP is not a substitute for NRC regulations, and compliance is not a requirement. However, 10 CFR 50.34 requires applications for light water reactor operating licenses and construction permits docketed after May 17, 1982, to include an evaluation of the facility against the SRP. Although currently operating reactors all had construction permits before 1982, the staff used the SRP in evaluating operating license applications for facilities that began commercial operation after 1982. Because compliance with the specific acceptance criteria in the SRP is not a requirement, use of the SRP in evaluating operating license applications does not mean that each reactor beginning commercial operation satisfies each acceptance criterion in the SRP. Rather, the staff used the SRP acceptance criteria as an aide in determining the acceptability of a structure, system, or component.

Detailed NRC guidance for evaluating the design of SFP storage facilities and the design of the SFP cooling and cleanup system is in SRP Sections 9.1.2 and 9.1.3, respectively. The acceptance criteria in SRP Section 9.1.2 relate to the SFP structural considerations for coolant inventory control, reactivity control criteria, and monitoring instrumentation. The acceptance criteria in SRP Section 9.1.3 relate to the SFP cooling system considerations for coolant inventory control and coolant temperature control. Both SRP sections reference Regulatory Guide 1.13 for specific criteria related to coolant inventory control.

Because of the unlikely prospects for successful reprocessing of civilian reactor fuel, the NRC developed Multi-Plant Action (MPA) A-28, "Increase in Spent Fuel Pool Storage Capacity," to address continued on-site storage of spent fuel. The staff developed a task action plan in the late 1970's to resolve MPA A-28. This action plan resulted in the development of guidance to address the increased number of SFP modifications involving replacement of low

density fuel storage racks with high density fuel storage racks. Operating reactor licensees pursued these modifications because, at the time many operating reactor spent fuel storage areas were designed, offsite storage and reprocessing of spent fuel was expected to limit the need for onsite storage.

On April 14, 1978, the NRC staff issued a letter to all power reactor licensees that forwarded the NRC guidance on SFP modifications. The guidance, entitled "Review and Acceptance of Spent Fuel Storage and Handling Applications," gave (1) guidance on the type and extent of information needed by the NRC staff to perform the review of proposed modifications to an operating reactor spent fuel storage pool and (2) the acceptance criteria to be used by the NRC staff in authorizing such modifications. The review areas addressed by this guidance included prevention of criticality, prevention of mechanical damage to fuel, and adequacy of cooling for the increased fuel storage capacity.

The actions recommended to resolve the action plan issues for MPA A-28 were to revise the NUREG-75/087 version of SRP Section 9.1.3 and the 1975 version of Regulatory Guide 1.13. Although revisions to Regulatory Guide 1.13 were developed that expanded the scope of the document to address SFP cooling and reactivity control, the revised version was not issued for comment. Minor revisions to SRP Section 9.1.3 were incorporated in the NUREG-0800 version in 1981.

In 1977, the NRC initiated the Systematic Evaluation Program (SEP) to review the designs of older operating nuclear reactors. Although the staff originally planned to conduct the SEP in several phases, the SEP was conducted in two phases. The first phase involved identification of issues for which regulatory guidance and requirements had changed enough since licensing of the older plants to warrant a re-evaluation of those older operating reactors. In the second phase, the staff re-evaluated 10 of the older operating reactors (7 of which are currently operating) against the guidance and requirements existing at the time of the re-evaluation. From the results of the second phase, the staff identified 27 issues, termed the SEP "lessons learned" issues, that involved some corrective action at one or more of the 10 reactors reviewed in the second phase of the SEP. The staff concluded that these 27 issues would be generally applicable to other older operating reactors that were not reviewed in the second phase of the SEP, and the staff proposed to include these issues in the Integrated Safety Assessment Program (ISAP). However, the ISAP was discontinued after reviews at two pilot plants. The SEP "lessons learned" issues were subsequently tracked as Generic Issue (GI) 156 until resolution of that GI in 1995.

Fuel storage was one of the issues identified in the first phase of the SEP. The purpose of the fuel storage review in the second phase of the SEP was to ensure that new and irradiated fuel are stored safely with respect to criticality prevention, cooling capability, shielding, and structural capability. For the seven currently operating reactors reviewed in the second phase of the SEP, the staff found that irradiated fuel was stored safely at those facilities on the basis of staff reviews conducted in the late 70s or early 80s that approved license amendments for increased spent fuel storage capacity. During the staff's review of the SEP program as part of our action

plan for spent fuel storage pool safety, the staff determined that three of the seven license amendments for spent fuel storage capacity increases were approved on the basis of substantial hardware modification to the SFP cooling system. Despite the hardware modifications necessary to satisfy the staff acceptance criteria at the time of the increase in spent fuel storage capacity, the staff did not identify the fuel storage issue as an SEP "lessons learned" issue.

3.0 PARAMETERS AFFECTING THE SAFE STORAGE OF IRRADIATED FUEL

3.1 Coolant Inventory

The coolant inventory in the SFP protects the fuel cladding by cooling the fuel, protects operators by serving as shielding, decreases fission product releases from postulated fuel handling events by retaining soluble and particulate fission products, and supports operation of forced cooling systems by providing adequate net positive suction head. Adequate cooling of the fuel and cladding is established by maintaining a coolant level above the top of the fuel (however, this condition does not ensure that the SFP structure and other non-fuel components will not be degraded by high temperature). A water depth of several feet above the top of irradiated fuel assemblies stored in racks serves as acceptable shielding, but additional water depth is necessary to provide adequate shielding during movement of fuel assemblies above the storage racks and to maintain operator dose as low as is reasonably achievable (ALARA). Consequence analyses for fuel handling accidents typically assume a water depth of 23 feet above the top of irradiated fuel storage racks, and this value is specified as a minimum depth for fuel handling operations in the NRC's Standard Technical Specifications. Because cooling system suction connections to the SFP are typically located well above the top of stored fuel to prevent inadvertent drainage, a substantial depth of water above the top of fuel storage racks is necessary to provide adequate net positive suction head for forced cooling system pumps.

Design features to reduce the potential for a loss of coolant inventory are common. On the basis of the staff's design review, all operating reactors have a reinforced-concrete SFP structure designed to retain their function following the design-basis seismic event (i.e., seismic Category I or Class 1) and a welded, corrosion-resistant SFP liner. Only one operating reactor lacks leak detection channels positioned behind liner plate welds to collect leakage and direct the leakage to a point where it can easily be monitored. Nearly all operating reactors have passive features preventing draining or siphoning of the SFP to a coolant level below the top of stored, irradiated fuel. Excluding paths used for irradiated fuel transfer, passive features at nearly all operating reactors prevent draining or siphoning of coolant to a level that provides inadequate shielding for fuel seated in the storage racks.

In the event that SFP coolant inventory decreases significantly, several indications are available to alert operators of that condition. The primary indication is a low-level alarm. A secondary indication of a loss of coolant level is provided by area radiation alarms. These alarms indicate a loss of shielding that occurs when SFP coolant inventory is lost. Except for the SFP located inside the containment building, the area radiation alarms are set to

alarm at a level low enough to detect a loss of coolant inventory early enough to allow for recovery before radiation levels could make such a recovery difficult.

The staff noted five categories of operating reactors that warrant further review based on specific design features that are contrary to guidance in Regulatory Guide 1.13. These categories are described in the next five sections.

3.1.1 Spent Fuel Pool Siphoning via Interfacing Systems

The SFPs serving four operating reactors lack passive anti-siphon devices for piping systems that could, through improper operation of the system, reduce coolant inventory to a level that provides insufficient shielding and eventually exposes stored fuel. These four operating reactors, all issued construction permits preceding the issuance of Safety Guide 13, have piping that penetrates the SFP liner several feet above the top of stored fuel, but the piping extends nearly to the bottom of the SFPs. Because, for each of these reactors, this piping is connected to the SFP cooling and cleanup system through a normally locked closed valve and lacks passive anti-siphon protection, mispositioning of the normally locked-closed valve coincident with a pipe break or refueling water transfer operation could reduce the SFP coolant inventory by siphon flow to a level below the top of the stored fuel.

This concern is related to a 1988 event at San Onofre Unit 2, which involved a partial loss of SFP coolant inventory due to an improper purification system alignment and inadequate anti-siphon protection. The NRC issued Information Notice 88-65, "Inadvertent Drainages of Spent Fuel Pools," to alert holders of operating licenses and construction permits of this event and similar system misalignments. Although the coolant inventory loss at San Onofre Unit 2 was not significant in this instance, the piping extended deep enough in the pool that failure of operator action to halt the inventory loss would have been of concern. Corrective action for this event included removing the portion of piping that extended below the technical specification limit on SFP level and strengthening administrative controls on system alignment.

Reduction in coolant inventory to an extremely low level is unlikely because of the low probability of the necessary coincident events, the long time period necessary for significant inventory loss through small siphon lines, and the many opportunities afforded operators to identify the inventory loss (e.g., SFP low-level alarm, SFP area high-radiation alarms, building sump high-level alarms, observed low level in SFP, and accumulation of water in unexpected locations). However, the staff believes that a design modification to introduce passive anti-siphon protection for the SFP could be easily implemented at the plants currently lacking this protection. Therefore, the staff will conduct a regulatory analysis to determine if such modifications are justified.

3.1.2 Spent Fuel Pool Drainage via the Fuel Transfer System

The SFPs serving five operating reactors contain fuel transfer tubes located at elevations below the top of fuel stored in the SFP racks. These five reactors also held construction permits preceding the issuance of Safety Guide 13. During refueling periods when the blank flange on the containment side of the transfer tube is removed, improper operation of the spent fuel transfer system or the SFP cooling and cleanup system could lead to a loss of coolant inventory from the SFP to the refueling cavity inside the containment through the transfer tube.

This concern is related to a 1984 event at Haddam Neck, which involved a massive loss of water from the reactor refueling cavity inside the containment caused by a failed refueling cavity seal. The spent fuel transfer tube at Haddam Neck, which separates the refueling cavity inside the containment from the SFP in the fuel handling building, enters the SFP at an elevation below the top of the stored fuel, and, had the transfer tube been open at the time of the refueling cavity seal failure, the water loss could have uncovered fuel stored in the SFP. The NRC issued Information Notice 84-93, "Potential for Loss of Water from the Refueling Cavity," to alert holders of operating licenses and construction permits of this event and of similar, but less severe, seal failures.

Since that event, the licensee for Haddam Neck has installed a cofferdam to prevent water loss through the transfer tube to such an extent that fuel could be uncovered and has also improved the design of the refueling cavity seal. With the exception of the five operating reactors with transfer tubes in their associated SFPs, operating reactors have some type of weir that separates the fuel transfer area from the storage area so that loss of coolant inventory through the fuel transfer system to a level below the top of the stored fuel is prevented by design.

A review of refueling cavity seal failure potential by all operating reactor licensees, which was performed in response to NRC Bulletin 84-03, "Refueling Cavity Water Seal," indicated that refueling cavity seal failures were more likely to occur at Haddam Neck than at other operating reactors because of the unique design of the Haddam Neck refueling cavity. The review also found that such failures would likely be less severe at other reactors than at Haddam Neck. Other potential drainage paths (e.g., refueling cavity drains and systems interfacing with the reactor coolant system) have a much lower maximum rate of water loss because of the smaller flow area. Therefore, similar to the loss of coolant inventory scenario by siphoning, water loss from the refueling cavity that exposes fuel in the SFP is unlikely because of the low probability of water loss from the refueling cavity when the transfer tube is open, the long time period necessary for the inventory loss, and the many opportunities for operators to identify the inventory loss. However, the staff concludes that the relative rarity of fuel transfer systems lacking passive design features to prevent uncovery of stored fuel warrants a more detailed review of the design features and administrative controls at the operating reactors that have this characteristic. The staff will perform regulatory analyses at these five reactors to determine if any safety enhancement backfits related to this design feature are justified under current guidance.

3.1.3 Spent Fuel Pool Drainage via Interfacing Systems

Of the five operating reactors associated with SFPs containing fuel transfer tubes at elevations below the top of the stored fuel, three have an interfacing system connected to the transfer tube. This interfacing system is designed to supply purified water from the SFP for reactor coolant pump seal injection during certain low-probability events postulated to occur during reactor operation. Administrative controls maintain the SFP inventory available to supply water to this interfacing system during reactor operation.

The configuration of this system increases the potential for inadvertent drainage that uncovers fuel. The configuration introduces the potential for improper alignment of the interfacing system or failure of the piping for the interfacing system so that coolant inventory is lost; the staff did not find this potential at any other operating reactor. By design, the system withdraws water from the SFP for reactor coolant pump seal injection at a rate that would leave insufficient water for shielding over the stored fuel after 72 hours of operation. The inadvertent drainage of the SFP to a level that would uncover the stored fuel is an unlikely event based on the long time period necessary for the inventory loss and the many opportunities for operators to discover the inventory loss. However, the staff has concluded that a safety enhancement modification to the SFP may be justified to ensure that the fuel remains covered for any potential occurrence involving the interfacing system piping. Therefore, the staff will conduct a regulatory analysis to determine if such a modification is justified.

3.1.4 Absence of a Direct Low Level Alarm

Absence of a direct SFP low level alarm could delay operator identification of a significant loss of SFP coolant inventory. The staff identified one operating reactor that does not have some type of SFP low-level alarm, but that reactor does have control room indication of SFP level and the SFP is inside the containment building. Additionally, six operating reactors have only indirect indication and alarm for a low SFP level. These six reactors have low-level alarms in the SFP cooling system surge tanks and low-discharge-pressure alarms for the SFP cooling system pumps. Surge tanks are used to accommodate movement of large objects, such as spent fuel storage casks, into and out of the SFP and thermal expansion or contraction of the coolant without a large change in coolant level. To accomplish this function, surge tanks are separated from the SFP by a weir slightly below the normal SFP water level, and the SFP cooling system pumps draw water from the surge tanks. With continuous operation of the SFP cooling system pumps, the surge tank low-level alarm is equivalent to the SFP level alarm because the surge tank would rapidly drain once the SFP level decreased below the surge tank entry weir. The SFP cooling system pump low-discharge-pressure alarms would alert the operators to a change in the status of the cooling system pumps. The staff will perform regulatory analyses at these seven reactors to determine if any safety enhancement backfits to improve SFP level monitoring capability are justified under current guidance.

3.1.5 Absence of Isolation Capability for Leakage Collection System

The absence of isolation capability for leakage identification systems could allow water to leak at a rate in excess of make-up capability for certain events that cause failure of the SFP liner. The staff identified four operating reactors with this characteristic, but this item was not included in our previous information collection efforts. However, the staff also has not collected the information necessary to evaluate makeup capability relative to credible leakage through the leakage detection channels. To address this omission, the staff will examine previous licensing reviews to determine if the staff had previously evaluated makeup capability relative to credible coolant inventory loss through the leakage detection channels. Because the four plants identified with this characteristic were not evaluated for inventory control using the SRP guidance, the staff believes that the depth of review for these plants would be indicative of the depth of review at other operating reactors. If this issue has not been previously addressed by the staff at the four operating reactors, the staff will initiate additional information collection activities for this design characteristic and conduct a regulatory analysis to determine if modification to the leakage detection system is justified.

3.2 Coolant Temperature

Coolant temperature has a less direct effect on safe storage of irradiated fuel than coolant inventory. Coolant temperature at the pool surface is limited by evaporative cooling from the free surface of the pool to a value of about 100°C [212°F], and the design of the pool storage racks provides adequate natural circulation to maintain the coolant in a subcooled state at the fuel cladding surface assuming the coolant inventory is at its normal level. Therefore, forced cooling is not required to protect the fuel cladding integrity when adequate water is supplied to makeup for coolant inventory loss. The temperature of the SFP does have an effect on structural loads, the operation of SFP purification systems, operator performance during fuel handling, and the environment around the SFP.

3.2.1 Structural Considerations

The SFP structure is evaluated to ensure that its structural integrity and leak tightness are retained under various operating, accidental, and environmental loadings. The reinforced concrete SFP walls and floors are required to withstand the loadings without exceeding the corresponding allowables set forth in the American Concrete Institute Code requirements for Nuclear Structures (ACI 349) as modified by Regulatory Guide 1.142. Appendix A, "Thermal Consideration," of ACI 349 limits the long-term temperature exposure of concrete surfaces to 150° F, and short term exposures temperature (under accident conditions) to 350° F. It permits long term temperature exposures higher than 150° F, provided tests are performed to evaluate reductions in the concrete strengths and elastic modulus, and these reductions are applied to design allowables. During the approval of Amendments related to reracking of SFPs, the staff reviews the structural, thermal and seismic loadings on the SFPs and the proposed storage racks to ensure their compliance with the regulatory provisions (relevant SRPs and Regulatory Guides).

Under normal operating conditions (including that associated with reactor refueling activities), the regulatory provisions ensure that the sustained concrete surface temperatures are below 150° F. However, during a rise in the SFP bulk temperature due to temporary loss of forced cooling, the low thermal diffusivity of concrete and the large thermal capacity of the SFP concrete cause the temperature distribution within the concrete structure to change slowly after a rise in the temperature. Evaporative cooling of the pool limits the maximum temperature attainable at the concrete surface following a temporary loss of forced cooling. Thus, the concrete material properties will not be affected due to a temporary rise in SFP bulk temperature above 150° F.

The inside surfaces of the concrete walls and floors of the SFP are provided with a leak tight and corrosion resistant (generally stainless steel) liner. The liner is anchored to the concrete walls and floor by means of structural shapes and/or headed studs. The liner between the anchors could move away from the walls and the floor under differential temperature effects on the walls, floor, and the liner. In most cases, the liner ductility and anchor strength would accommodate such differential temperature effects. However, some construction features of the liner and its anchorage could give rise to high stress concentrations and liner weld failure under high temperature exposures. Such failure, if they should occur would be localized, and would be detected during maintenance, and/or by the leakage detection system (see Section 3.1.5).

Therefore, it is reasonable to conclude that if thermal loads on pool structure are limited and their effects monitored as discussed above, no significant structural degradation of the SFP structure is likely to occur.

3.2.2 Coolant Purification

Temperature also has an indirect effect on fuel integrity and radiological conditions. All SFPs use an ion exchange and filtration processes to maintain the purity of the coolant. The chemical contaminants in the coolant affect the corrosion resistance of components in the fuel pool and the activity of the coolant. However, the ion exchange resins may degrade at temperatures above 60°C [140°F], and the degradation can cause the release of previously absorbed impurities in addition to reducing the effectiveness of the resin. Some SFP purification subsystems operate using water from the outlet of the SFP heat exchanger, which protects the ion exchange resin in these subsystems from high pool temperature. The purification subsystems for other SFPs must be isolated to protect the resin when pool temperature is high.

Prolonged isolation of the purification subsystem creates the potential for increased operator exposure from radionuclide accumulation in the pool coolant and increased corrosion from impurities that accumulate in the coolant. However, chemical and radiological monitoring of SFP water is routinely specified in each facility's safety analysis report and operating procedures. Such monitoring ensures that the coolant is maintained sufficiently pure to avoid excessive accumulation of radionuclides or chemical impurities in the SFP coolant.

3.2.3 Fuel Handling

Lastly, SFP temperature affects operator performance during fuel handling. A pool temperature above 37°C [100°F] can lead to frequent operator rotation during fuel movement to prevent heat stress, and higher pool temperatures can result in fogging on the operating floor that interferes with an operator's ability to observe fuel assembly position. To avoid these problems, most operating reactor licensees have implemented administrative controls to maintain pool temperature in a range that does not hinder operator performance.

3.2.4 Environmental Effects of High Temperature in the SFP

At very high temperatures in the SFP, the evaporative cooling that occurs on the pool surface can add a significant amount of latent heat and water vapor to the atmosphere of the building surrounding the SFP. Depending on the ventilation system design and capability, the added heat and water vapor could increase building temperature and condensation on equipment. The higher temperature and condensation could impair the operation of essential safety systems.

The staff has extensively evaluated this issue at one operating reactor site, Susquehanna. The deterministic analysis of Susquehanna indicated that systems used to cool the spent fuel storage pool were adequate to prevent unacceptable challenges to the safety related systems needed to protect public health and safety during and following design basis events. The probabilistic review at Susquehanna indicated that event sequences leading to a sustained loss of SFP cooling have a low frequency of occurrence. In particular, the staff found that loss of operator access to SFP cooling system components, which was a principal contention of the report filed pursuant to 10 CFR Part 21 regarding loss of SFP cooling at Susquehanna, is not a significant contributor to the frequency of sustained loss of SFP cooling events because the probability of severe core damage that has the potential to deny operator access to the building housing the SFP is very low. The staff recognized that the mechanisms by which the operators would be unable to provide cooling to the SFP were not limited to the design basis events and operator access considerations. Therefore, the staff modeled other event sequences leading to SFP boiling. The staff concluded that, even with consideration of the additional event sequences, loss of SFP cooling events presented a challenge of low safety significance to the plant.

On the basis of deterministic and probabilistic evaluations at Susquehanna, the staff concluded that this concern can be adequately addressed through provision of a reliable SFP cooling system or through administrative controls that extend the time available to institute recovery actions following a loss of cooling. The reliability of the SFP cooling function at each operating reactor is dependent on the design of the SFP cooling system and each licensee's administrative controls on availability of systems capable of cooling the SFP. The time available for recovery action following a loss of SFP cooling is dependent on the initial temperature of the SFP coolant, the decay heat rate of the stored fuel, and the available passive heat sinks. Because the decay heat rate within the SFP is at least an order of magnitude higher during refueling operations involving a full-core discharge than during

reactor operation and because refueling is a controlled evolution, administrative controls on refueling operations affect the time available for recovery following a loss of SFP cooling.

Through the extensive evaluation of Susquehanna, the NRC staff identified certain design characteristics that increase the probability that an elevated SFP temperature will interfere with the safe operation of a reactor either at power or shutdown. The first characteristic is an open path from the area around the SFP to areas housing safety systems. This path may be through personnel or equipment access ports, ventilation system ducting, or condensate drain paths. Without an open path, the large surface area of the enclosure around a SFP would allow water vapor to condense and return to the SFP and allow heat to be rejected through the enclosure to the environment without affecting reactor safety systems. The second characteristic is a short time for the SFP to reach elevated temperatures. The time for the SFP to reach an elevated temperature is affected by initial temperature, coolant inventory, and the decay heat rate of irradiated fuel. On the basis of operating practices and administrative limits on SFP temperature, the NRC staff has determined that short times to reach elevated temperatures are credible only when nearly the entire core fuel assembly inventory has been transferred to the SFP and the reactor has been shut down for a short period after extended operation at power.

These conditions establish the third design characteristic, which is a reactor site with multiple operating units sharing structures and systems related to the SFP. At a single-unit site, large coolant inventories in the SFP and in the reactor cavity act as a large passive heat sink for irradiated fuel during fuel transfer. When the entire core fuel assembly inventory has been transferred to the SFP at a single-unit site, safety systems associated with the reactor are not essential because no fuel remains in the reactor vessel. Multi-unit sites with no shared structures can be treated as a single-unit site. At a multi-unit site with shared structures, a short time to reach an elevated temperature can exist in the SFP associated with a reactor in refueling while safety systems in communication with the area around that SFP are supporting operation of another reactor at power.

When these three design characteristics coexist at a single site, one SFP could reach an elevated temperature in a short time (i.e., between 4 and 10 hours) after a sustained loss of cooling, the heat and water vapor could propagate to systems necessary for shutdown of an operating reactor, and these systems could subsequently fail while needed to support shutdown.

The staff has determined through its survey of SFP design features that these three design characteristics coexist at no more than seven operating reactor sites in addition to Susquehanna. The staff determined through its review of design information and operational controls that immediate regulatory action is not warranted on the basis of the capability of available cooling systems, the passive heat capacity of the SFP, and the operational limits imposed by administrative controls at these seven sites. In making this determination, the staff considered the findings from its review of this issue at Susquehanna. Nevertheless, the staff will conduct detailed reviews to

identify enhancements to refueling procedures or cooling system reliability that are justified based on the reduced potential for SFP conditions to impact safety systems supporting an operating reactor at these seven sites.

3.2.5 Cooling System Reliability and Capability

The SFP cooling system reliability and capability affect the ability of the licensee to maintain SFP temperature within an appropriate band. Through its survey of operating reactors, the staff identified some commonality with respect to control of the cooling system, but substantial variation in the design of fuel pool cooling systems with respect to reliability and capability.

The large, passive heat sink provided by the SFP coolant reduces the significance of a short-term loss of cooling by providing ample time for operator diagnosis of problems and implementation of corrective action. Consequently, SFP cooling systems are typically aligned, operated, and controlled by manual actions. Most plants have SFP cooling system pump controls only at local control stations near the pumps.

The staff identified a wide range of SFP cooling system configurations. The least reliable configuration consisted of a single-train system with no backup system capable of providing SFP cooling. This system was designed with two 50-percent flow-capacity pumps supplying a single heat exchanger. The electrical distribution system serving this reactor was not configured to supply onsite power to the SFP cooling pumps. At the other end of the range, the SFP cooling system consisted of two redundant, high-capacity, safety-grade trains of cooling. The primary SFP cooling system was supported by the safety-grade shutdown cooling system, which was capable of being aligned to cool the SFP.

The staff analyzed design information collected during the survey to determine the susceptibility of SFP cooling systems to a sustained loss of SFP cooling. Specifically, the staff examined the minimum design capacity of the system with no failures, the capacity of the system assuming long-term failure of a single pump, the capacity assuming a LOOP, the passive thermal capacity of the SFP, and the availability of a large-capacity backup system. In order to have a consistent basis for comparison, the staff developed a numerical rating for each reactor based on a ratio of heat removal capacity under limiting conditions relative to the rated thermal power of each reactor.

On the basis of design information collected through the staff's survey effort and onsite assessment visits, the staff identified events that are most likely to lead to extended reductions in SFP cooling capability. Because the SFP cooling systems typically do not maintain train separation in control cabinets and power cable raceways, events such as fires or internal floods may cause a complete loss of SFP cooling. Also, the primary SFP cooling systems often are designed such that their cooling capacity would be eliminated during a LOOP. However, operators are more likely to recover from minor electrical and control system failures by rerouting power cables and bypassing control cabinets than they are to recover from mechanical failures requiring a unique part for repair in the time available before the SFP reaches elevated temperatures. On this basis, the staff concludes that the operating reactors

identified with relatively low cooling capacity that lack redundancy of mechanical components are more likely to experience elevated SFP temperatures than those reactors with greater SFP cooling capacity or mechanical component redundancy. Similarly, those reactors without an onsite source of power to a system capable of cooling the SFP are more likely to experience elevated SFP temperatures than reactors having a cooling system designed to be powered from an onsite power source. However, once again, the long period of time available for operator diagnosis of a problem and identification of appropriate corrective action reduces the level of risk from elevated SFP temperatures.

The staff noted that the SFPs for all but seven operating reactors are capable of being cooled by a system powered from an onsite source without special reconfiguration of the electrical distribution system. However, nine of the operating reactors with onsite power available to a system capable of cooling the SFP rely on backup SFP cooling using a mode of the reactor shutdown cooling system. This mode of system operation often requires significant realignment for fuel pool cooling.

The staff concluded that all SFPs associated with U.S. operating reactors can withstand, without bulk boiling in the SFP, a long-term loss of one SFP cooling system pump or cooling water system (i.e., service water or closed cooling water system) pump and maintain 50 to 100 percent of full decay heat removal capability using redundant or installed spare pumps. However, with reduced cooling capability, the rate of water vapor production from the SFP may be significant for operating reactors with lower heat removal capability under certain conditions.

To address concerns with the reliability and capability of SFP cooling systems, the staff will conduct evaluations and regulatory analyses at selected operating reactors. The first category of operating reactors are those seven operating reactors lacking a design capability to supply onsite power to a system capable of cooling the SFP. The staff will examine the capability to supply onsite power to the SFP cooling system relative to the time available for recovery actions based on procedural controls to determine the need for regulatory analyses. The second category of operating reactors are operating reactors identified with low primary SFP cooling system cooling capacity relative to potential spent fuel decay heat generation that have no backup cooling capability. The staff will examine the administrative controls with respect to SFP temperature and available recovery time at four operating reactors with low SFP cooling capacity to determine the need for regulatory analyses. The final category of operating reactors are those reactors reliant on infrequently operated backup SFP cooling systems to address long-term LOOP events and mechanical failures. The staff will examine administrative controls on the availability of the backup cooling systems during refueling and technical analyses demonstrating the capability of these backup systems to cool the SFP at the ten operating reactors in this category to determine the need for further regulatory analyses.

3.2.6 Absence of Direct Instrumentation for Loss of the SFP Cooling Function

Inadequate SFP cooling can be indicated by a high SFP temperature alarm, a SFP cooling system low flow alarm, a cooling system high temperature alarm, or a

SFP cooling system pump low discharge pressure alarm. The staff's survey results indicate that ten operating reactors lack a direct-reading high SFP temperature alarm to identify a sustained loss of SFP cooling and, of those ten reactors, one lacks any associated alarms for a loss of cooling. Because the associated alarms provide annunciation of SFP cooling problem at nine of the operating reactors, because the SFP for the tenth operating reactor is located inside primary containment where equipment is qualified for harsh environments, and because routine operator monitoring also has the potential to detect a loss of the SFP cooling function, the staff determined that immediate regulatory action was not warranted. However, the staff will examine these reactor sites further to determine if additional instrumentation or operational controls are warranted on a safety enhancement basis.

3.3 Fuel Reactivity

All irradiated fuel storage racks are designed to maintain a substantial shutdown reactivity margin for normal and abnormal storage conditions. The NRC staff acceptance criterion for all storage conditions, including abnormal or accident storage conditions (e.g., fuel handling accident, mispositioned fuel assembly, or storage temperature outside of normal range), is a very high confidence that the effective neutron multiplication factor is 0.95 or less. Every licensee is required to maintain this shutdown reactivity margin as a design feature technical specification or as a commitment contained in each licensee's safety analysis report. The NRC staff has accepted credit taken for the negative reactivity introduced by soluble boron in abnormal or accident storage conditions where dilution of the boron concentration would not be a possible outcome of the abnormal or accident condition alone.

3.3.1 Solid Neutron Absorbers

To maintain a substantial shutdown reactivity margin in a regular array of fuel assemblies, the storage geometry, the neutron absorption characteristics of the storage array, and the reactivity and position of fuel assemblies in the array are controlled. Reliance on geometry alone results in a low-density storage configuration. No operating reactor currently uses only low-density storage in its associated SFP. Intermediate storage density can be achieved by either special construction of the storage racks to form "flux traps" or by controlling the position and reactivity of fuel stored in the rack. The reactivity of each fuel assembly is typically determined by its initial enrichment in the uranium-235 isotope, its integrated irradiation (burnup), and its integral burnable neutron poison inventory. The highest density fuel storage has been achieved through the use of solid neutron absorbers as integral parts of the storage racks.

All solid neutron absorbers used at U.S. operating reactors utilize the high neutron absorption cross-section of the boron-10 isotope. Boron held in a silicon-rubber matrix (Boraflex) is the most common solid neutron absorber, followed by an aluminum/boron carbide alloy (Boral). Boron carbide clad in a metal sheathing is the next most common neutron absorber. Borated stainless steel pins are in use at one SFP associated with an operating reactor. The SFP storage racks associated with 14 of 109 U.S. operating reactors contain no solid neutron absorbers. The remaining SFPs use one or more of the solid neutron absorbers identified above to achieve higher storage density.

Because boron-10 is consumed by the interaction with neutrons, storage racks containing neutron absorbers are designed assuming a finite neutron irradiation and, therefore, a finite operating life. Other mechanisms that deplete the boron-10 inventory in the storage racks can reduce the operating life of the storage racks under design storage conditions. Although the SFP environment is relatively benign for most of the neutron absorbers in use, Boraflex has been observed to degrade by two mechanisms (1) gamma irradiation-induced shrinkage and (2) boron washout following long-term gamma irradiation combined with exposure to the wet pool environment. In addition to issuing three information notices regarding Boraflex degradation, the NRC staff issued Generic Letter (GL) 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks," on June 26, 1996. This GL requires licensees using Boraflex in their spent fuel storage racks to submit information to the NRC staff regarding their plans to address potential degradation of Boraflex material. This action on Boraflex is outside the staff's action plan activities.

A review of neutron absorber performance as part of the action plan for spent fuel storage pool safety indicates that degradation in neutron absorption performance has not been observed in materials other than Boraflex. Some neutron absorbing panels have been observed to swell due to gas accumulation within the cladding material, but this effect has not degraded neutron absorption performance.

3.3.2 Soluble Boron

Soluble boron is used in pressurized water reactors (PWRs) to control reactor coolant system reactivity. Because the SFP interfaces with the reactor coolant system during refueling, an adequate boron concentration must be maintained in the SFP to preclude inadvertent dilution of the reactor coolant system. In addition, the boron concentration maintained in PWR SFPs is also credited with mitigating reactivity transients caused by abnormal or accident fuel storage conditions. The NRC staff found that soluble boron concentration was adequately controlled by administrative controls or technical specifications at PWRs.

4.0 PLANNED ACTIONS

The staff has identified three courses of action to address the areas described in Section 3.0. These courses of action are (1) plant-specific evaluations or regulatory analyses for safety enhancement backfits, (2) rulemaking, and (3) revision of staff guidance for SFP evaluation. In addition, the staff will issue an information notice as a mechanism for distributing information in areas where regulatory analyses do not support rulemaking or plant-specific backfits.

4.1 Plant Specific Evaluations and Regulatory Analyses

The staff has identified several areas for additional plant-specific evaluation. The bases for these additional reviews was described in Section 3.0. The staff has identified specific operating reactors in each of the following categories for further evaluation:

1. Absence of Passive Antisiphon Devices on Piping Extending Below Top of Stored Fuel

2. Transfer Tube(s) Within SFP Rather Than Separate Transfer Canal

3. Piping Entering Pool Below Top of Stored Fuel

4. Limited Instrumentation for Loss of Coolant Events

5. Absence of Leak Detection Capability or Absence of Isolation Valves in Leakage Detection System Piping

6. Shared Systems and Structures at Multi-Unit Sites

7. Absence of On-site Power Supply for Systems Capable of SFP Cooling

8. Limited SFP Decay Heat Removal Capability

9. Infrequently Used Backup SFP Cooling Systems

10. Limited Instrumentation for Loss of Cooling Events

The specific operating reactors in each category are named in the following summaries. Each summary also describes existing design features at the named reactors and other capabilities that limit the risk from each identified concern.

Inventory Control Issues

1. Absence of Passive Antisiphon Devices on Piping Extending Below the Top of Stored Fuel

Plants:	Davis-Besse, Robinson, and Turkey Point 3 & 4
Concern:	Misconfiguration of system has the potential to syphon coolant to such an extent that fuel could be exposed to air.
Current Protection:	Locked closed valve on line at level of pool liner penetration, liner penetration well above top of stored fuel, low level alarm, and operator action (stop syphon flow and add make-up water)
Action:	Regulatory analysis to assess potential enhancements

2. Transfer Tube(s) Within SFP Rather Than Separate Transfer Canal

Plants:	Crystal River, Maine Yankee, and Oconee 1, 2, & 3
Concern:	Transfer tubes are normally open during refueling operations. When these openings are below the top

of stored fuel, any drain path from the refueling cavity has the potential to reduce coolant inventory to an extent that stored fuel could be exposed to air.

Current Protection: Low-level alarm, blank flange closure during reactor operation, and operator action (stop drainage and add makeup water)

Action: Regulatory analysis to assess potential enhancements

3. Piping Entering Pool Below Top of Stored Fuel

Plants: Oconee Units 1, 2, & 3

Concern: Pipe break or misconfiguration of piping supporting the standby shutdown facility (SSF) at Oconee has potential to drain coolant to such an extent that fuel could be exposed to air. [The SSF at Oconee uses SFP coolant as a supply of reactor coolant pump seal water for certain low-probability events. The supply pipe for the SSF is a 3 inch diameter, seismically-qualified pipe that ties into a transfer tube for each unit. The Oconee safety analysis report states that the transfer tube gate valve is normally open during reactor operation to support SSF initiation.]

Current Protection: Seismic qualification of piping, normally closed valves on line, low level alarm, and operator action (stop drainage flow and add make-up water)

Action: Regulatory analysis to assess potential enhancements

4. Limited Instrumentation for Loss of SFP Coolant Events

Plants: Big Rock Point, Dresden 2 & 3, Peach Bottom 2 & 3, and Hatch 1 & 2

Concern: Insufficient instrumentation to reliably alert operators to a loss of SFP coolant inventory or a sustained loss of SFP cooling.

Current Protection: Related alarms, operating procedures, and operator identification

Action: Regulatory analysis to assess potential enhancements

5. Absence of Leak Detection Capability or Absence of Isolation Valves in Leakage Detection System Piping

Plants: D. C. Cook 1 & 2, Indian Point 2, and Salem 1 & 2

[possibly others - ~~leak detection system drain~~ isolation information was not part of design survey - staff will conduct further review of other sites]

Concern: Coolant inventory loss is not easily isolated following events that breach the SFP liner.

Current Protection: Limited flow area through leak detection system tell-tale drains, low leak rate through concrete structure, controls on movement of loads over fuel pool, and operator action (plug leak detection system drains and add make-up)

Action: Further Evaluation of Condition

Decay Heat Removal Reliability Issues

6. Shared Systems and Structures at Multi-Unit Sites

Plants: Calvert Cliffs 1 & 2, D. C. Cook 1 & 2, Dresden 2 & 3, Hatch 1 (Hatch 2 lower levels are a separate secondary containment zone), LaSalle 1 & 2, Point Beach 1 & 2, and Quad Cities 1 & 2

Concern: With one unit in refueling, the decay heat rate in the SFP may be sufficiently high that the pool could reach boiling in a short period of time following a loss of cooling. Communication between the fuel pool area and areas housing safety equipment supporting the operating unit through shared ventilation systems or shared structures may cause failure or degradation of those systems.

Current Protection: Restrictive administrative controls on refueling operations, reliable SFP cooling systems, and operator actions to restore forced cooling and protect essential systems from the adverse environmental conditions that may develop during SFP boiling

Action: Regulatory analysis to assess potential enhancements

7. Absence of On-site Power Supply for Systems Capable of SFP Cooling

Plants: ANO 2, Prairie Island 1 & 2, Surry 1 & 2, and Zion 1 & 2

Concern: A sustained loss of offsite power at plants without an on-site power supply for SFP cooling may lead to departure from subcooled decay heat removal in the fuel pool, increased thermal stress in pool structures, loss of coolant inventory, increased levels of airborne radioactivity, and adverse

environmental effects in areas communicating with the SFP area.

Current Protection: Operator action (align a temporary power supply from an on-site source or establish alternate cooling such as feed and bleed using diesel powered pump), high temperature alarm, filtered ventilation, and separation/isolation of areas containing equipment important to safety from the SFP area

Action: Regulatory analysis to assess potential enhancements

8. Limited SFP Decay Heat Removal Capability

Plants: Indian Point 2, Indian Point 3, and Salem 1 & 2

Concern: Assuming a full core discharges at an equivalent time after reactor shutdown during a period of peak ultimate heat sink temperature, these plants will have higher SFP equilibrium temperatures and shorter recovery times than other similar plants.

Current Protection: Administrative controls on refueling operations

Action: Evaluation of administrative controls

9. Infrequently Used Backup SFP Cooling Systems

Plants: Browns Ferry 2 & 3, Davis-Besse, Dresden 2 & 3, Fermi, Fitzpatrick, Hatch 1 & 2, and WNP-2

Concern: These plants are more reliant on infrequently operated backup cooling systems than other similar plants because of the absence of an onsite power supply for the primary SFP cooling system or low relative capacity of the primary cooling system.

Current Protection: Administrative controls on refueling operations and availability of backup SFP cooling capability

Action: Evaluation of capability to effectively use backup system

10. Limited Instrumentation for Loss of Cooling Events

Plants: ANO-1, Big Rock Point, Brunswick 1 & 2, Cooper, Hatch 1 & 2, LaSalle 1 & 2, and Millstone 1

Concern: Instrumentation to alert operators to a sustained loss of SFP cooling is limited in capability.

Current Protection: Related alarms at most of above reactors, operating rocedures, and operator identification

Action: Regulatory analysis to assess potential enhancements

4.2 Implementation of the Shutdown Rule for Spent Fuel Pool Operations

The primary benefit of including SFP operations in the shutdown rule is the establishment of clear and consistent performance standards for forced cooling of the SFP. Existing design features and operational controls provide assurance that a substantial shutdown reactivity margin will be maintained within the SFP. Similarly, common SFP design features have resulted in a low probability of a significant loss of SFP coolant inventory. Those facilities that lack specific design features are best examined on a plant-specific basis to determine if any enhancements to operating procedures or modifications to structures or systems are warranted.

A performance-based shutdown rule addressing SFP cooling would establish a consistent level of safety with specific performance goals. Those reactors with more capable cooling systems and those licensees that more carefully plan refueling cycles would benefit from increased maintenance flexibility during refueling outages. This approach is more appropriate from a safety standpoint than is the current situation of applying stringent design basis limits to reactors with more capable cooling systems.

4.3 Revision of Staff Guidance

The staff will develop guidance supporting implementation of the Shutdown Rule for SFP shutdown operations. The staff will also develop revisions to Regulatory Guide 1.13 and SRP Section 9.1.3. Regulatory Guide 1.13 will be expanded to include guidance related to design performance of SFP cooling systems, and SRP Section 9.1.3 will be revised to be consistent with that regulatory guide.

5.0 CONCLUSIONS

The staff has found that existing structures, systems, and components related to the storage of irradiated fuel provide adequate protection for public health and safety. Protection has been provided by several layers of defenses that perform accident prevention functions, accident mitigation functions, radiation protection functions, and emergency preparedness functions. Design features addressing each of these areas for spent fuel storage have been reviewed and approved by the staff. In addition, the limited risk analyses available for spent fuel storage suggest that current design features and operational constraints cause issues related to SFP storage to be a small fraction of the overall risk associated with an operating light water reactor. Notwithstanding this finding, the staff has reviewed each operating reactor's spent fuel pool design to identify strengths and weaknesses, and to identify potential areas for safety enhancements.

The staff plans to address issues relating to the functional performance of SFP decay heat removal, as well as the operational aspects related to coolant inventory control and reactivity control, through expansion of the proposed, performance-based rule for Shutdown Operations at Nuclear Power Plants (10 CFR 50.67) tc encompass fuel storage pool operations.

The staff also plans to address certain design features that reduce the reliability of SFP decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. We intend to pursue regulatory analyses for safety enhancement backfits on a plant-specific basis pursuant to 10 CFR 50.109 at the operating reactor sites possessing one or more of these design features.

Concurrent with the regulatory analyses for the potential safety enhancements, the staff will develop guidance for implementing the proposed rule for fuel storage pool operations at nuclear power plants. The staff will also develop plans to improve existing guidance documents related to SFP storage.

PLANTS TO BE NOTIFIED ABOUT PLANNED PLANT-SPECIFIC SPENT FUEL POOL ANALYSES

ANO 1
ANO 2
Big Rock Point
Browns Ferry 2,3
Brunswick 1,2
Calvert Cliffs 1,2
Cooper
Crystal River
DC Cook 1,2
Davis-Besse
Dresden 2,3
Fermi
Fitzpatrick
Hatch 1,2
Indian Point 2
Indian Point 3
LaSalle 1,2
Millstone 1
Maine Yankee
Oconee 1,2,3
Quad Cities 1,2
Peach Bottom 2,3
Point Beach 1,2
Prairie Island 1,2
Robinson
Salem 1,2
Surry 1,2
Turkey Point 3,4
WNP 2
Zion 1,2

Attachment 2

SAMPLE LETTERS FOR AFFECTED PLANTS

Attachment 3

SAMPLE LETTER FOR: H.B. Robinson, Turkey Point 3 and 4
[Licensee Address]

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL SAFETY ISSUES: ISSUANCE OF FINAL STAFF REPORT AND NOTIFICATION OF STAFF PLANS TO PERFORM PLANT-SPECIFIC, SAFETY ENHANCEMENT BACKFIT ANALYSES, **[PLANT NAME, (TAC NO.)]**

Dear .....:

The Nuclear Regulatory Commission staff recently completed a detailed review of spent fuel storage pool safety issues. The results of the staff's review are documented in a report to the Commission which is enclosed for your information. In the report, the staff concludes that existing structures, systems, and components related to the storage of irradiated fuel provide adequate protection of public health and safety.

Notwithstanding this finding, the staff has also identified certain design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. The staff intends to conduct plant-specific regulatory analyses to evaluate potential safety enhancement backfits pursuant to 10 CFR 50.109(a)(3) at a number of operating plants that possess one or more of these design features.

The NRC staff determined through a review of spent fuel pool design information that **[Plant Name]** has piping that extends to near the bottom of the spent fuel pool. The design information indicated that this piping lacks a passive anti-siphon device, but the piping is normally isolated by a locked closed valve. Because this piping is connected to the SFP cooling and cleanup system through a normally locked closed valve and lacks passive anti-siphon protection, mispositioning of the normally locked-closed valve coincident with a pipe break or refueling water transfer operation could reduce the SFP coolant inventory by siphon flow to a level below the top of the stored fuel. Although consistent with design standards at the time of construction, the absence of a passive anti-siphon device on piping that extends below the top of stored fuel is rare among U.S. operating reactors and contrary to current design standards for spent fuel pools. On this basis, the staff intends to initiate activities to evaluate the potential for a safety enhancement backfit.

If you wish to comment on the accuracy of the staff's understanding of the plant design, the safety significance of the above design features, the cost of potential modifications to address the above design features, or the existing protection from the above design concerns, which may be provided by administrative controls or other means, comments received before November 15, 1996, will be considered in developing plans for inspections and other activities associated with the planned regulatory analysis.

If you have any questions regarding this matter, please do not hesitate to contact [PM] at [PM phone].

Sincerely,
PROJECT MANAGER

Docket No(s).

Enclosure: Memo to the Commission, from J. Taylor, "Resolution of Spent Fuel Storage Pool Action Plan Issues," dated July 26, 1996

cc w/encl: See next page
CONCURRENCE: LA, PM, PD
NOTE: PLEASE INCLUDE J SHEA ON DISTRIBUTION W/O ENCLOSURE.

SAMPLE LETTER FOR: Davis-Besse
[Licensee Address]

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL SAFETY ISSUES: ISSUANCE OF FINAL STAFF REPORT AND NOTIFICATION OF STAFF PLANS TO PERFORM PLANT-SPECIFIC, SAFETY ENHANCEMENT BACKFIT ANALYSES, **[PLANT NAME, (TAC NO.)]**

Dear .....:

The Nuclear Regulatory Commission staff recently completed a detailed review of spent fuel storage pool safety issues. The results of the staff's review are documented in a report to the Commission which is enclosed for your information. In the report, the staff concludes that existing structures, systems, and components related to the storage of irradiated fuel provide adequate protection of public health and safety.

Notwithstanding this finding, the staff has also identified certain design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. The staff intends to conduct plant-specific regulatory analyses to evaluate potential safety enhancement backfits pursuant to 10 CFR 50.109(a)(3) at a number of operating plants that possess one or more of these design features. The staff also identified a number of facilities where additional technical evaluation is necessary to determine the need for further regulatory analysis.

The NRC staff determined through a review of spent fuel pool design information that **[Plant Name]** has piping that extends to near the bottom of the spent fuel pool. The design information indicated that this piping lacks a passive anti-siphon device, but the piping is normally isolated by a locked closed valve. Because this piping is connected to the SFP cooling and cleanup system through a normally locked closed valve and lacks passive anti-siphon protection, mispositioning of the normally locked-closed valve coincident with a pipe break or refueling water transfer operation could reduce the SFP coolant inventory by siphon flow to a level below the top of the stored fuel. Although consistent with design standards at the time of construction, the absence of a passive anti-siphon device on piping that extends below the top of stored fuel is rare among U.S. operating reactors and contrary to current design standards for spent fuel pools. On this basis, the staff intends to initiate activities to evaluate the potential for a safety enhancement backfit.

In addition, the staff noted that **[Plant Name]** appears to be reliant on infrequently operated backup SFP cooling systems to address long-term loss-of-offsite-power events and mechanical failures. The staff will examine administrative controls on the availability of the backup cooling systems during refueling and technical analyses demonstrating the capability of these backup systems to cool the SFP at **[Plant Name]** to determine the need for further regulatory analyses.

If you wish to comment on the accuracy of the staff's understanding of the plant design, the safety significance of the above design features, the cost

of potential modifications to address the above design features, or the existing protection from the above design concerns which may be provided by administrative controls or other means, comments received before November 15, 1996, will be considered in developing plans for inspections and other activities associated with the planned regulatory analysis.

If you have any questions regarding this matter, please do not hesitate to contact [PM] at [PM phone].

Sincerely,
PROJECT MANAGER

Docket No(s).

Enclosure: Memo to the Commission, from
J. Taylor, "Resolution of Spent Fuel
Storage Pool Action Plan Issues,"
dated July 26, 1996

cc w/encl: See next page
CONCURRENCE: LA, PM, PD
NOTE: PLEASE INCLUDE J SHEA ON DISTRIBUTION W/O ENCLOSURE.

SAMPLE LETTER FOR: Oconee 1, 2 and 3
[Licensee Address]

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL SAFETY ISSUES: ISSUANCE OF FINAL STAFF REPORT AND NOTIFICATION OF STAFF PLANS TO PERFORM PLANT-SPECIFIC, SAFETY ENHANCEMENT BACKFIT ANALYSES, **[PLANT NAME, (TAC NO.)]**

Dear .....:

The Nuclear Regulatory Commission staff recently completed a detailed review of spent fuel storage pool safety issues. The results of the staff's review are documented in a report to the Commission which is enclosed for your information. In the report, the staff concludes that existing structures, systems, and components related to the storage of irradiated fuel provide adequate protection of public health and safety.

Notwithstanding this finding, the staff has also identified certain design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. The staff intends to conduct plant-specific regulatory analyses to evaluate potential safety enhancement backfits pursuant to 10 CFR 50.109(a)(3) at a number of operating plants that possess one or more of these design features.

The NRC staff determined through a review of spent fuel pool design information that **[Plant Name]** have fuel transfer tubes that penetrate the spent fuel pool wall at an elevation below the top of fuel stored in the SFP racks. During refueling periods when the blank flange on the containment side of the transfer tube is removed, improper operation of the spent fuel transfer system or the SFP cooling and cleanup system could lead to a loss of coolant inventory from the SFP to the refueling cavity inside the containment through the transfer tube. The staff concludes that the relative rarity of fuel transfer systems lacking passive design features to prevent uncovery of stored fuel warrants a more detailed review of the design features and administrative controls at the operating reactors that have this characteristic. The staff will perform regulatory analyses of your plant to determine if any safety enhancement backfits related to this design feature are justified under current guidance.

In addition to having fuel transfer tubes with an elevation below the top of the stored fuel, **[Plant Name]** have an interfacing system connected to the transfer tube. This interfacing system is designed to supply purified water from the SFP for reactor coolant pump seal injection during certain low-probability events postulated to occur during reactor operation. Administrative controls maintain the SFP inventory available to supply water to this interfacing system during reactor operation. The configuration of this system increases the potential for inadvertent drainage that uncovers fuel. The configuration introduces the potential for improper alignment of the interfacing system or failure of the piping for the interfacing system so that coolant inventory is lost; the staff did not find this potential at any other operating reactor. By design, the system withdraws water from the SFP for reactor coolant pump seal injection at a rate that would leave

insufficient water for shielding over the stored fuel after 72 hours of operation. The inadvertent drainage of the SFP to a level that would uncover the stored fuel is an unlikely event based on the long time period necessary for the inventory loss and the many opportunities for operators to discover the inventory loss. However, the staff has concluded that a safety enhancement modification to the SFP may be justified to ensure that the fuel remains covered for any potential occurrence involving the interfacing system piping. Therefore, the staff will conduct a regulatory analysis to determine if such a modification is justified.

If you wish to comment on the accuracy of the staff's understanding of the plant design, the safety significance of the above design features, the cost of potential modifications to address the above design features, or the existing protection from the above design concerns which may be provided by administrative controls or other means, comments received before November 15, 1996, will be considered in developing plans for inspections and other activities associated with the planned regulatory analysis.

If you have any questions regarding this matter, please do not hesitate to contact [PM] at [PM phone].

Sincerely,
PROJECT MANAGER

Docket No(s).

Enclosure: Memo to the Commission, from
J. Taylor, "Resolution of Spent Fuel
Storage Pool Action Plan Issues,"
dated July 26, 1996

cc w/encl: See next page
CONCURRENCE: LA, PM, PD
NOTE: PLEASE INCLUDE J SHEA ON DISTRIBUTION W/O ENCLOSURE.

SAMPLE LETTER FOR: Crystal River, Maine Yankee
[Licensee Address]

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL SAFETY ISSUES: ISSUANCE OF FINAL STAFF REPORT AND NOTIFICATION OF STAFF PLANS TO PERFORM PLANT-SPECIFIC, SAFETY ENHANCEMENT BACKFIT ANALYSES, **[PLANT NAME, (TAC NO.)]**

Dear .....:

The Nuclear Regulatory Commission staff recently completed a detailed review of spent fuel storage pool safety issues. The results of the staff's review are documented in a report to the Commission which is enclosed for your information. In the report, the staff concludes that existing structures, systems, and components related to the storage of irradiated fuel provide adequate protection of public health and safety.

Notwithstanding this finding, the staff has also identified certain design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. The staff intends to conduct plant-specific regulatory analyses to evaluate potential safety enhancement backfits pursuant to 10 CFR 50.109(a)(3) at a number of operating plants that possess one or more of these design features.

The NRC staff determined through a review of spent fuel pool design information that **[Plant Name]** has a fuel transfer tube that penetrates the spent fuel pool (SFP) wall at an elevation below the top of fuel stored in the SFP racks. During refueling periods when the blank flange on the containment side of the transfer tube is removed, improper operation of the spent fuel transfer system or the SFP cooling and cleanup system could lead to a loss of coolant inventory from the SFP to the refueling cavity inside the containment through the transfer tube. The staff concludes that the relative rarity of fuel transfer systems lacking passive design features to prevent uncovery of stored fuel warrants a more detailed review of the design features and administrative controls at the operating reactors that have this characteristic. The staff will perform regulatory analyses of your plant to determine if any safety enhancement backfits related to this design feature are justified under current guidance.

If you wish to comment on the accuracy of the staff's understanding of the plant design, the safety significance of the above design features, the cost of potential modifications to address the above design features, or the existing protection from the above design concerns which may be provided by administrative controls or other means, comments received before November 15, 1996, will be considered in developing plans for inspections and other activities associated with the planned regulatory analysis.

If you have any questions regarding this matter, please do not hesitate to contact [PM] at [PM phone].

Sincerely,
PROJECT MANAGER

Docket No(s).

Enclosure: Memo to the Commission, from J. Taylor, "Resolution of Spent Fuel Storage Pool Action Plan Issues," dated July 26, 1996

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CONCURRENCE: LA, PM, PD
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SAMPLE LETTER FOR: **Peach Bottom 2 and 3**
[Licensee Address]

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL SAFETY ISSUES: ISSUANCE OF FINAL STAFF REPORT AND NOTIFICATION OF STAFF PLANS TO PERFORM PLANT-SPECIFIC, SAFETY ENHANCEMENT BACKFIT ANALYSES,
[PLANT NAME, (TAC NO.)]

Dear .....:

The Nuclear Regulatory Commission staff recently completed a detailed review of spent fuel storage pool safety issues. The results of the staff's review are documented in a report to the Commission which is enclosed for your information. In the report, the staff concludes that existing structures, systems, and components related to the storage of irradiated fuel provide adequate protection of public health and safety.

Notwithstanding this finding, the staff has also identified certain design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. The staff intends to conduct plant-specific regulatory analyses to evaluate potential safety enhancement backfits pursuant to 10 CFR 50.109(a)(3) at a number of operating plants that possess one or more of these design features.

The staff identified that **[Plant Name]** has only indirect indication and alarm for a low SFP level. **[Plant Name]** has low-level alarms in the SFP cooling system surge tanks and low-discharge-pressure alarms for the SFP cooling system pumps. Surge tanks are used to accommodate movement of large objects, such as spent fuel storage casks, into and out of the SFP and thermal expansion or contraction of the coolant without a large change in coolant level. To accomplish this function, surge tanks are separated from the SFP by a weir slightly below the normal SFP water level, and the SFP cooling system pumps draw water from the surge tanks. With continuous operation of the SFP cooling system pumps, the surge tank low-level alarm is equivalent to the SFP level alarm because the surge tank would rapidly drain once the SFP level decreased below the surge tank entry weir. The SFP cooling system pump low-discharge-pressure alarms would alert the operators to a change in the status of the cooling system pumps. Although consistent with design standards at the time of construction, absence of a direct SFP low level alarm could delay operator identification of a significant loss of SFP coolant inventory. The staff will perform regulatory analyses at **[Plant Name]** to determine if any safety enhancement backfits to improve SFP level monitoring capability are justified under current guidance.

If you wish to comment on the accuracy of the staff's understanding of the plant design, the safety significance of the above design features, the cost of potential modifications to address the above design features, or the existing protection from the above design concerns which may be provided by administrative controls or other means, comments received before November 15, 1996, will be considered in developing plans for inspections and other activities associated with the planned regulatory analysis.

If you have any questions regarding this matter, please do not hesitate to contact [PM] at [PM phone].

Sincerely,
PROJECT MANAGER

Docket No(s).

Enclosure: Memo to the Commission, from J. Taylor, "Resolution of Spent Fuel Storage Pool Action Plan Issues," dated July 26, 1996

cc w/encl: See next page
CONCURRENCE: LA, PM, PD
NOTE: PLEASE INCLUDE J SHEA ON DISTRIBUTION W/O ENCLOSURE.

SAMPLE LETTER FOR: Dresden 2 and 3,
[Licensee Address]

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL SAFETY ISSUES: ISSUANCE OF FINAL STAFF REPORT AND NOTIFICATION OF STAFF PLANS TO PERFORM PLANT-SPECIFIC, SAFETY ENHANCEMENT BACKFIT ANALYSES, **[PLANT NAME, (TAC NO.)]**

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The Nuclear Regulatory Commission staff recently completed a detailed review of spent fuel storage pool safety issues. The results of the staff's review are documented in a report to the Commission which is enclosed for your information. In the report, the staff concludes that existing structures, systems, and components related to the storage of irradiated fuel provide adequate protection of public health and safety.

Notwithstanding this finding, the staff has also identified certain design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. The staff intends to conduct plant-specific regulatory analyses to evaluate potential safety enhancement backfits pursuant to 10 CFR 50.109(a)(3) at a number of operating plants that possess one or more of these design features.

The staff identified that **[Plant Name]** has only indirect indication and alarm for a low SFP level. **[Plant Name]** has low-level alarms in the SFP cooling system surge tanks and low-discharge-pressure alarms for the SFP cooling system pumps. Surge tanks are used to accommodate movement of large objects, such as spent fuel storage casks, into and out of the SFP and thermal expansion or contraction of the coolant without a large change in coolant level. To accomplish this function, surge tanks are separated from the SFP by a weir slightly below the normal SFP water level, and the SFP cooling system pumps draw water from the surge tanks. With continuous operation of the SFP cooling system pumps, the surge tank low-level alarm is equivalent to the SFP level alarm because the surge tank would rapidly drain once the SFP level decreased below the surge tank entry weir. The SFP cooling system pump low-discharge-pressure alarms would alert the operators to a change in the status of the cooling system pumps. Although consistent with design standards at the time of construction, absence of a direct SFP low level alarm could delay operator identification of a significant loss of SFP coolant inventory. The staff will perform regulatory analyses at **[Plant Name]** to determine if any safety enhancement backfits to improve SFP level monitoring capability are justified under current guidance.

Through the extensive evaluation of loss of spent fuel pool cooling concerns at the Susquehanna Steam Electric Station, the NRC staff identified certain design characteristics that increase the probability that an elevated SFP temperature will interfere with the safe operation of a reactor either at power or shutdown. The first characteristic is an open path from the area around the SFP to areas housing safety systems. This path may be through personnel or equipment access ports, ventilation system ducting, or condensate drain paths. Without an open path, the large surface area of the enclosure

around a SFP would allow water vapor to condense and return to the SFP and allow heat to be rejected through the enclosure to the environment without affecting reactor safety systems. The second characteristic is a short time for the SFP to reach elevated temperatures. The time for the SFP to reach an elevated temperature is affected by initial temperature, coolant inventory, and the decay heat rate of irradiated fuel. On the basis of operating practices and administrative limits on SFP temperature, the NRC staff has determined that short times to reach elevated temperatures are credible only when nearly the entire core fuel assembly inventory has been transferred to the SFP and the reactor has been shut down for a short period after extended operation at power.

These conditions establish the third design characteristic, which is a reactor site with multiple operating units sharing structures and systems related to the SFP. At a single-unit site, large coolant inventories in the SFP and in the reactor cavity act as a large passive heat sink for irradiated fuel during fuel transfer. When the entire core fuel assembly inventory has been transferred to the SFP at a single-unit site, safety systems associated with the reactor are not essential because no fuel remains in the reactor vessel. Multi-unit sites with no shared structures can be treated as a single-unit site. At a multi-unit site with shared structures, a short time to reach an elevated temperature can exist in the SFP associated with a reactor in refueling while safety systems in communication with the area around that SFP are supporting operation of another reactor at power.

When these three design characteristics coexist at a single site, one SFP could reach an elevated temperature in a short time (i.e., between 4 and 10 hours) after a sustained loss of cooling, the heat and water vapor could propagate to systems necessary for shutdown of an operating reactor, and these systems could subsequently fail while needed to support shutdown.

The staff has determined through its survey of SFP design features that these three design characteristics coexist at no more than seven operating reactor sites in addition to Susquehanna. **[Plant Name]** was one of the plants identified. The staff determined through its review of design information and operational controls that immediate regulatory action is not warranted on the basis of the capability of available cooling systems, the passive heat capacity of the SFP, and the operational limits imposed by administrative controls at these seven sites. In making this determination, the staff considered the findings from its review of this issue at Susquehanna. Nevertheless, the staff will conduct detailed reviews to identify enhancements to refueling procedures or cooling system reliability that are justified based on the reduced potential for SFP conditions to impact safety systems supporting an operating reactor at these seven sites, including **[Plant Name]**.

The staff noted that **[Plant Name]** appears to be reliant on infrequently operated backup SFP cooling systems to address long-term LOOP events and mechanical failures. The staff will examine administrative controls on the availability of the backup cooling systems during refueling and technical

analyses demonstrating the capability of these backup systems to cool the SFP at **[Plant Name]** in this category to determine the need for further regulatory analyses.

If you wish to comment on the accuracy of the staff's understanding of the plant design, the safety significance of the above design features, the cost of potential modifications to address the above design features, or the existing protection from the above design concerns which may be provided by administrative controls or other means, comments received before November 15, 1996, will be considered in developing plans for inspections and other activities associated with the planned regulatory analysis.

If you have any questions regarding this matter, please do not hesitate to contact [PM] at [PM phone].

Sincerely,
PROJECT MANAGER

Docket No(s).

Enclosure: Memo to the Commission, from J. Taylor, "Resolution of Spent Fuel Storage Pool Action Plan Issues," dated July 26, 1996

cc w/encl: See next page
CONCURRENCE: LA, PM, PD
NOTE: PLEASE INCLUDE J SHEA ON DISTRIBUTION W/O ENCLOSURE.

SAMPLE LETTER FOR: Hatch 1 and 2
[Licensee Address]

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL SAFETY ISSUES: ISSUANCE OF FINAL STAFF REPORT AND NOTIFICATION OF STAFF PLANS TO PERFORM PLANT-SPECIFIC, SAFETY ENHANCEMENT BACKFIT ANALYSES, **[PLANT NAME, (TAC NO.)]**

Dear .....:

The Nuclear Regulatory Commission staff recently completed a detailed review of spent fuel storage pool safety issues. The results of the staff's review are documented in a report to the Commission which is enclosed for your information. In the report, the staff concludes that existing structures, systems, and components related to the storage of irradiated fuel provide adequate protection of public health and safety.

Notwithstanding this finding, the staff has also identified certain design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. The staff intends to conduct plant-specific regulatory analyses to evaluate potential safety enhancement backfits pursuant to 10 CFR 50.109(a)(3) at a number of operating plants that possess one or more of these design features.

The staff identified that **[Plant Name]** has only indirect indication and alarm for a low SFP level. **[Plant Name]** has low-level alarms in the SFP cooling system surge tanks and low-discharge-pressure alarms for the SFP cooling system pumps. Surge tanks are used to accommodate movement of large objects, such as spent fuel storage casks, into and out of the SFP and thermal expansion or contraction of the coolant without a large change in coolant level. To accomplish this function, surge tanks are separated from the SFP by a weir slightly below the normal SFP water level, and the SFP cooling system pumps draw water from the surge tanks. With continuous operation of the SFP cooling system pumps, the surge tank low-level alarm is equivalent to the SFP level alarm because the surge tank would rapidly drain once the SFP level decreased below the surge tank entry weir. The SFP cooling system pump low-discharge-pressure alarms would alert the operators to a change in the status of the cooling system pumps. Although consistent with design standards at the time of construction, absence of a direct SFP low level alarm could delay operator identification of a significant loss of SFP coolant inventory. The staff will perform regulatory analyses at **[Plant Name]** to determine if any safety enhancement backfits to improve SFP level monitoring capability are justified under current guidance.

Through the extensive evaluation of loss of spent fuel pool cooling concerns at the Susquehanna Steam Electric Station, the NRC staff identified certain design characteristics that increase the probability that an elevated SFP temperature will interfere with the safe operation of a reactor either at power or shutdown. The first characteristic is an open path from the area around the SFP to areas housing safety systems. This path may be through personnel or equipment access ports, ventilation system ducting, or condensate drain paths. Without an open path, the large surface area of the enclosure around a SFP would allow water vapor to condense and return to the SFP and

allow heat to be rejected through the enclosure to the environment without affecting reactor safety systems. The second characteristic is a short time for the SFP to reach elevated temperatures. The time for the SFP to reach an elevated temperature is affected by initial temperature, coolant inventory, and the decay heat rate of irradiated fuel. On the basis of operating practices and administrative limits on SFP temperature, the NRC staff has determined that short times to reach elevated temperatures are credible only when nearly the entire core fuel assembly inventory has been transferred to the SFP and the reactor has been shut down for a short period after extended operation at power.

These conditions establish the third design characteristic, which is a reactor site with multiple operating units sharing structures and systems related to the SFP. At a single-unit site, large coolant inventories in the SFP and in the reactor cavity act as a large passive heat sink for irradiated fuel during fuel transfer. When the entire core fuel assembly inventory has been transferred to the SFP at a single-unit site, safety systems associated with the reactor are not essential because no fuel remains in the reactor vessel. Multi-unit sites with no shared structures can be treated as a single-unit site. At a multi-unit site with shared structures, a short time to reach an elevated temperature can exist in the SFP associated with a reactor in refueling while safety systems in communication with the area around that SFP are supporting operation of another reactor at power.

When these three design characteristics coexist at a single site, one SFP could reach an elevated temperature in a short time (i.e., between 4 and 10 hours) after a sustained loss of cooling, the heat and water vapor could propagate to systems necessary for shutdown of an operating reactor, and these systems could subsequently fail while needed to support shutdown.

The staff has determined through its survey of SFP design features that these three design characteristics coexist at no more than seven operating reactor sites in addition to Susquehanna. **[Plant Name- Hatch1 only for this issue]** was one of the plants identified. The staff determined through its review of design information and operational controls that immediate regulatory action is not warranted on the basis of the capability of available cooling systems, the passive heat capacity of the SFP, and the operational limits imposed by administrative controls at these seven sites. In making this determination, the staff considered the findings from its review of this issue at Susquehanna. Nevertheless, the staff will conduct detailed reviews to identify enhancements to refueling procedures or cooling system reliability that are justified based on the reduced potential for SFP conditions to impact safety systems supporting an operating reactor at these seven sites, including **[Plant Name- Hatch 1 only for this issue]**.

The staff observed that inadequate SFP cooling can be indicated by a high SFP temperature alarm, a SFP cooling system low flow alarm, a cooling system high temperature alarm, or a SFP cooling system pump low discharge pressure alarm. The staff observed that ten operating reactors, including **[Plant Name]** lack a direct-reading high SFP temperature alarm to identify a sustained loss of SFP

cooling and, of those ten reactors, one, lacks any associated alarms for a loss of cooling. Because the associated alarms provide annunciation of SFP cooling problem at nine of the operating reactors, because the SFP for the tenth operating reactor is located inside primary containment where equipment is qualified for harsh environments, and because routine operator monitoring also has the potential to detect a loss of the SFP cooling function, the staff determined that immediate regulatory action was not warranted. However, the staff will examine these reactor sites further to determine if additional instrumentation or operational controls are warranted on a safety enhancement basis.

The staff noted that **[Plant Name]** appears to be reliant on infrequently operated backup SFP cooling systems to address long-term LOOP events and mechanical failures. The staff will examine administrative controls on the availability of the backup cooling systems during refueling and technical analyses demonstrating the capability of these backup systems to cool the SFP at **[Plant Name]** in this category to determine the need for further regulatory analyses.

If you wish to comment on the accuracy of the staff's understanding of the plant design, the safety significance of the above design features, the cost of potential modifications to address the above design features, or the existing protection from the above design concerns which may be provided by administrative controls or other means, comments received before November 15, 1996, will be considered in developing plans for inspections and other activities associated with the planned regulatory analysis.

If you have any questions regarding this matter, please do not hesitate to contact [PM] at [PM phone].

Sincerely,
PROJECT MANAGER

Docket No(s).

Enclosure: Memo to the Commission, from J. Taylor, "Resolution of Spent Fuel Storage Pool Action Plan Issues," dated July 26, 1996

cc w/encl: See next page
CONCURRENCE: LA, PM, PD
NOTE: PLEASE INCLUDE J SHEA ON DISTRIBUTION W/O ENCLOSURE.

SAMPLE LETTER FOR: Big Rock Point
[Licensee Address]

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL SAFETY ISSUES: ISSUANCE OF FINAL STAFF REPORT AND NOTIFICATION OF STAFF PLANS TO PERFORM PLANT-SPECIFIC, SAFETY ENHANCEMENT BACKFIT ANALYSES, **[PLANT NAME, (TAC NO.)]**

Dear .....:

The Nuclear Regulatory Commission staff recently completed a detailed review of spent fuel storage pool safety issues. The results of the staff's review are documented in a report to the Commission which is enclosed for your information. In the report, the staff concludes that existing structures, systems, and components related to the storage of irradiated fuel provide adequate protection of public health and safety.

Notwithstanding this finding, the staff has also identified certain design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. The staff intends to conduct plant-specific regulatory analyses to evaluate potential safety enhancement backfits pursuant to 10 CFR 50.109(a)(3) at a number of operating plants that possess one or more of these design features.

The staff identified that **[Plant Name]** does not have a spent fuel pool (SFP) low-level alarm, but that reactor does have control room indication of SFP level and the SFP is inside the containment building. Absence of a direct SFP low level alarm could delay operator identification of a significant loss of SFP coolant inventory. The staff will perform regulatory analyses at **[Plant Name]** to determine if any safety enhancement backfits to improve SFP level monitoring capability are justified under current guidance.

The staff observed that inadequate SFP cooling can be indicated by a high SFP temperature alarm, a SFP cooling system low flow alarm, a cooling system high temperature alarm, or a SFP cooling system pump low discharge pressure alarm. The staff observed that ten operating reactors, including **[Plant Name]** lack a direct-reading high SFP temperature alarm to identify a sustained loss of SFP cooling and, of those ten reactors, one, **[Big Rock Point]**, lacks any associated alarms for a loss of cooling. Because the associated alarms provide annunciation of SFP cooling problem at nine of the operating reactors, because the SFP for the tenth operating reactor is located inside primary containment where equipment is qualified for harsh environments, and because routine operator monitoring also has the potential to detect a loss of the SFP cooling function, the staff determined that immediate regulatory action was not warranted. However, the staff will examine these reactor sites further to determine if additional instrumentation or operational controls are warranted on a safety enhancement basis.

If you wish to comment on the accuracy of the staff's understanding of the plant design, the safety significance of the above design features, the cost of potential modifications to address the above design features, or the existing protection from the above design concerns which may be provided by administrative controls or other means, comments received before November 15, 1996, will be considered in developing plans for inspections and other activities associated with the planned regulatory analysis.

If you have any questions regarding this matter, please do not hesitate to contact [PM] at [PM phone].

Sincerely,
PROJECT MANAGER

Docket No(s).

Enclosure: Memo to the Commission, from J. Taylor, "Resolution of Spent Fuel Storage Pool Action Plan Issues," dated July 26, 1996

cc w/encl: See next page
CONCURRENCE: LA, PM, PD
NOTE: PLEASE INCLUDE J SHEA ON DISTRIBUTION W/O ENCLOSURE.

SAMPLE LETTER FOR: D. C. Cook 1 and 2
[Licensee Address]

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL SAFETY ISSUES: ISSUANCE OF FINAL STAFF REPORT AND NOTIFICATION OF STAFF PLANS TO PERFORM PLANT-SPECIFIC, SAFETY ENHANCEMENT BACKFIT ANALYSES, **[PLANT NAME, (TAC NO.)]**

Dear .....:

The Nuclear Regulatory Commission staff recently completed a detailed review of spent fuel storage pool safety issues. The results of the staff's review are documented in a report to the Commission which is enclosed for your information. In the report, the staff concludes that existing structures, systems, and components related to the storage of irradiated fuel provide adequate protection of public health and safety.

Notwithstanding this finding, the staff has also identified certain design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. The staff intends to conduct plant-specific regulatory analyses to evaluate potential safety enhancement backfits pursuant to 10 CFR 50.109(a)(3) at a number of operating plants that possess one or more of these design features.

The staff observed that the absence of isolation capability for leakage identification systems could allow water to leak at a rate in excess of make-up capability for certain events that cause failure of the SFP liner. The staff identified four operating reactors, including **[Plant Name]**, with this characteristic, but this item was not included in our previous information collection efforts. However, the staff also has not collected the information necessary to evaluate makeup capability relative to credible leakage through the leakage detection channels. To address this omission, the staff will examine previous licensing reviews to determine if the staff had previously evaluated makeup capability relative to credible coolant inventory loss through the leakage detection channels. Because the four plants identified with this characteristic, including **[Plant Name]**, were not evaluated for inventory control using the SRP guidance, the staff believes that the depth of review for these plants would be indicative of the depth of review at other operating reactors. If this issue has not been previously addressed by the staff at the four operating reactors, the staff will initiate additional information collection activities for this design characteristic and conduct a regulatory analysis to determine if modification to the leakage detection system is justified.

Through the extensive evaluation of loss of spent fuel pool cooling concerns at the Susquehanna Steam Electric Station, the NRC staff identified certain design characteristics that increase the probability that an elevated SFP temperature will interfere with the safe operation of a reactor either at power or shutdown. The first characteristic is an open path from the area around the SFP to areas housing safety systems. This path may be through personnel or equipment access ports, ventilation system ducting, or condensate drain paths. Without an open path, the large surface area of the enclosure around a SFP would allow water vapor to condense and return to the SFP and

allow heat to be rejected through the enclosure to the environment without affecting reactor safety systems. The second characteristic is a short time for the SFP to reach elevated temperatures. The time for the SFP to reach an elevated temperature is affected by initial temperature, coolant inventory, and the decay heat rate of irradiated fuel. On the basis of operating practices and administrative limits on SFP temperature, the NRC staff has determined that short times to reach elevated temperatures are credible only when nearly the entire core fuel assembly inventory has been transferred to the SFP and the reactor has been shut down for a short period after extended operation at power.

These conditions establish the third design characteristic, which is a reactor site with multiple operating units sharing structures and systems related to the SFP. At a single-unit site, large coolant inventories in the SFP and in the reactor cavity act as a large passive heat sink for irradiated fuel during fuel transfer. When the entire core fuel assembly inventory has been transferred to the SFP at a single-unit site, safety systems associated with the reactor are not essential because no fuel remains in the reactor vessel. Multi-unit sites with no shared structures can be treated as a single-unit site. At a multi-unit site with shared structures, a short time to reach an elevated temperature can exist in the SFP associated with a reactor in refueling while safety systems in communication with the area around that SFP are supporting operation of another reactor at power.

When these three design characteristics coexist at a single site, one SFP could reach an elevated temperature in a short time (i.e., between 4 and 10 hours) after a sustained loss of cooling, the heat and water vapor could propagate to systems necessary for shutdown of an operating reactor, and these systems could subsequently fail while needed to support shutdown.

The staff has determined through its survey of SFP design features that these three design characteristics coexist at no more than seven operating reactor sites in addition to Susquehanna. **[Plant Name]** was one of the plants identified. The staff determined through its review of design information and operational controls that immediate regulatory action is not warranted on the basis of the capability of available cooling systems, the passive heat capacity of the SFP, and the operational limits imposed by administrative controls at these seven sites. In making this determination, the staff considered the findings from its review of this issue at Susquehanna. Nevertheless, the staff will conduct detailed reviews to identify enhancements to refueling procedures or cooling system reliability that are justified based on the reduced potential for SFP conditions to impact safety systems supporting an operating reactor at these seven sites, including **[Plant Name]**.

If you wish to comment on the accuracy of the staff's understanding of the plant design, the safety significance of the above design features, the cost of potential modifications to address the above design features, or the existing protection from the above design concerns which may be provided by administrative controls or other means, comments received before November 15,

1996, will be considered in developing plans for inspections and other activities associated with the planned regulatory analysis. If you have any questions regarding this matter, please do not hesitate to contact [PM] at [PM phone].

Sincerely,
PROJECT MANAGER

Docket No(s).

Enclosure: Memo to the Commission, from
J. Taylor, "Resolution of Spent Fuel
Storage Pool Action Plan Issues,"
dated July 26, 1996

cc w/encl: See next page
CONCURRENCE: LA, PM, PD
NOTE: PLEASE INCLUDE J SHEA ON DISTRIBUTION W/O ENCLOSURE.

SAMPLE LETTER FOR: Salem 1 and 2
[Licensee Address]

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL SAFETY ISSUES: ISSUANCE OF FINAL STAFF REPORT AND NOTIFICATION OF STAFF PLANS TO PERFORM PLANT-SPECIFIC, SAFETY ENHANCEMENT BACKFIT ANALYSES, **[PLANT NAME, (TAC NO.)]**

Dear .....:

The Nuclear Regulatory Commission staff recently completed a detailed review of spent fuel storage pool safety issues. The results of the staff's review are documented in a report to the Commission which is enclosed for your information. In the report, the staff concludes that existing structures, systems, and components related to the storage of irradiated fuel provide adequate protection of public health and safety.

Notwithstanding this finding, the staff has also identified certain design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. The staff intends to conduct plant-specific regulatory analyses to evaluate potential safety enhancement backfits pursuant to 10 CFR 50.109(a)(3) at a number of operating plants that possess one or more of these design features.

The staff observed that the absence of isolation capability for leakage identification systems could allow water to leak at a rate in excess of make-up capability for certain events that cause failure of the SFP liner. The staff identified four operating reactors, including **[Plant Name]**, with this characteristic, but this item was not included in our previous information collection efforts. However, the staff also has not collected the information necessary to evaluate makeup capability relative to credible leakage through the leakage detection channels. To address this omission, the staff will examine previous licensing reviews to determine if the staff had previously evaluated makeup capability relative to credible coolant inventory loss through the leakage detection channels. Because the four plants identified with this characteristic, including **[Plant Name]**, were not evaluated for inventory control using the SRP guidance, the staff believes that the depth of review for these plants would be indicative of the depth of review at other operating reactors. If this issue has not been previously addressed by the staff at the four operating reactors, the staff will initiate additional information collection activities for this design characteristic and conduct a regulatory analysis to determine if modification to the leakage detection system is justified.

The SFP cooling system reliability and capability affect the ability of the licensee to maintain SFP temperature within an appropriate band. The staff analyzed design information to determine the susceptibility of SFP cooling systems to a sustained loss of SFP cooling. Specifically, the staff examined the minimum design capacity of the system with no failures, the capacity of the system assuming long-term failure of a single pump, the capacity assuming a LOOP, the passive thermal capacity of the SFP, and the availability of a large-capacity backup system.

The staff noted that the SFPs **[Plant Name]** has a low primary SFP cooling system capacity relative to the potential decay heat generation and has no backup coolng capability. The staff intends to examine the administrative controls with respect to SFP temperature and available recovery time at **[Plant Name]** to determine the need for regulatory analyses.

If you wish to comment on the accuracy of the staff's understanding of the plant design, the safety significance of the above design features, the cost of potential modifications to address the above design features, or the existing protection from the above design concerns which may be provided by administrative controls or other means, comments received before November 15, 1996, will be considered in developing plans for inspections and other activities associated with the planned regulatory analysis.

If you have any questions regarding this matter, please do not hesitate to contact [PM] at [PM phone].

Sincerely,
PROJECT MANAGER

Docket No(s).

Enclosure: Memo to the Commission, from J. Taylor, "Resolution of Spent Fuel Storage Pool Action Plan Issues," dated July 26, 1996

cc w/encl: See next page
CONCURRENCE: LA, PM, PD
NOTE: PLEASE INCLUDE J SHEA ON DISTRIBUTION W/O ENCLOSURE.

SAMPLE LETTER FOR: Indian Point 2
[Licensee Address]

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL SAFETY ISSUES: ISSUANCE OF FINAL STAFF REPORT AND NOTIFICATION OF STAFF PLANS TO PERFORM PLANT-SPECIFIC, SAFETY ENHANCEMENT BACKFIT ANALYSES, **[PLANT NAME, (TAC NO.)]**

Dear .....:

The Nuclear Regulatory Commission staff recently completed a detailed review of spent fuel storage pool safety issues. The results of the staff's review are documented in a report to the Commission which is enclosed for your information. In the report, the staff concludes that existing structures, systems, and components related to the storage of irradiated fuel provide adequate protection of public health and safety.

Notwithstanding this finding, the staff has also identified certain design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. The staff intends to conduct plant-specific regulatory analyses to evaluate potential safety enhancement backfits pursuant to 10 CFR 50.109(a)(3) at a number of operating plants that possess one or more of these design features.

The staff observed that the absence of spent fuel pool liner leakage identification piping at **[Plant Name]**. The staff will examine how liner leakage is monitored at **[Plant Name]**. The staff will conduct a technical evaluation to determine the need for further regulatory analysis.

The SFP cooling system reliability and capability affect the ability of the licensee to maintain SFP temperature within an appropriate band. The staff analyzed design information to determine the susceptibility of SFP cooling systems to a sustained loss of SFP cooling. Specifically, the staff examined the minimum design capacity of the system with no failures, the capacity of the system assuming long-term failure of a single pump, the capacity assuming a LOOP, the passive thermal capacity of the SFP, and the availability of a large-capacity backup system.

The staff noted that the SFPs **[Plant Name]** has a low primary SFP cooling system capacity relative to the potential decay heat generation and has no backup coolng capability. The staff intends to examine the administrative controls with respect to SFP temperature and available recovery time at **[Plant Name]** to determine the need for regulatory analyses.

If you wish to comment on the accuracy of the staff's understanding of the plant design, the safety significance of the above design features, the cost of potential modifications to address the above design features, or the existing protection from the above design concerns which may be provided by administrative controls or other means, comments received before November 15, 1996, will be considered in developing plans for inspections and other activities associated with the planned regulatory analysis.

If you have any questions regarding this matter, please do not hesitate to contact [PM] at [PM phone].

Sincerely,
PROJECT MANAGER

Docket No(s).

Enclosure: Memo to the Commission, from J. Taylor, "Resolution of Spent Fuel Storage Pool Action Plan Issues," dated July 26, 1996

cc w/encl: See next page
CONCURRENCE: LA, PM, PD
NOTE: PLEASE INCLUDE J SHEA ON DISTRIBUTION W/O ENCLOSURE.

SAMPLE LETTER FOR: Calvert Cliffs 1 and 2, Point Beach 1 and 2, Quad Cities 1 and 2 /

[Licensee Address]

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL SAFETY ISSUES: ISSUANCE OF FINAL STAFF REPORT AND NOTIFICATION OF STAFF PLANS TO PERFORM PLANT-SPECIFIC, SAFETY ENHANCEMENT BACKFIT ANALYSES,
[PLANT NAME, (TAC NO.)]

Dear .....:

The Nuclear Regulatory Commission staff recently completed a detailed review of spent fuel storage pool safety issues. The results of the staff's review are documented in a report to the Commission which is enclosed for your information. In the report, the staff concludes that existing structures, systems, and components related to the storage of irradiated fuel provide adequate protection of public health and safety.

Notwithstanding this finding, the staff has also identified certain design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. The staff intends to conduct plant-specific regulatory analyses to evaluate potential safety enhancement backfits pursuant to 10 CFR 50.109(a)(3) at a number of operating plants that possess one or more of these design features.

Through the extensive evaluation of loss of spent fuel pool cooling concerns at the Susquehanna Steam Electric Station, the NRC staff identified certain design characteristics that increase the probability that an elevated SFP temperature will interfere with the safe operation of a reactor either at power or shutdown. The first characteristic is an open path from the area around the SFP to areas housing safety systems. This path may be through personnel or equipment access ports, ventilation system ducting, or condensate drain paths. Without an open path, the large surface area of the enclosure around a SFP would allow water vapor to condense and return to the SFP and allow heat to be rejected through the enclosure to the environment without affecting reactor safety systems. The second characteristic is a short time for the SFP to reach elevated temperatures. The time for the SFP to reach an elevated temperature is affected by initial temperature, coolant inventory, and the decay heat rate of irradiated fuel. On the basis of operating practices and administrative limits on SFP temperature, the NRC staff has determined that short times to reach elevated temperatures are credible only when nearly the entire core fuel assembly inventory has been transferred to the SFP and the reactor has been shut down for a short period after extended operation at power.

These conditions establish the third design characteristic, which is a reactor site with multiple operating units sharing structures and systems related to the SFP. At a single-unit site, large coolant inventories in the SFP and in the reactor cavity act as a large passive heat sink for irradiated fuel during fuel transfer. When the entire core fuel assembly inventory has been transferred to the SFP at a single-unit site, safety systems associated with the reactor are not essential because no fuel remains in the reactor vessel. Multi-unit sites with no shared structures can be treated as a single-unit site. At a multi-unit site with shared structures, a short time to reach an

elevated temperature can exist in the SFP associated with a reactor in refueling while safety systems in communication with the area around that SFP are supporting operation of another reactor at power.

When these three design characteristics coexist at a single site, one SFP could reach an elevated temperature in a short time (i.e., between 4 and 10 hours) after a sustained loss of cooling, the heat and water vapor could propagate to systems necessary for shutdown of an operating reactor, and these systems could subsequently fail while needed to support shutdown.

The staff has determined through its survey of SFP design features that these three design characteristics coexist at no more than seven operating reactor sites in addition to Susquehanna. **[Plant Name]** was one of the plants identified. The staff determined through its review of design information and operational controls that immediate regulatory action is not warranted on the basis of the capability of available cooling systems, the passive heat capacity of the SFP, and the operational limits imposed by administrative controls at these seven sites. In making this determination, the staff considered the findings from its review of this issue at Susquehanna. Nevertheless, the staff will conduct detailed reviews to identify enhancements to refueling procedures or cooling system reliability that are justified based on the reduced potential for SFP conditions to impact safety systems supporting an operating reactor at these seven sites, including **[Plant Name]**.

If you wish to comment on the accuracy of the staff's understanding of the plant design, the safety significance of the above design features, the cost of potential modifications to address the above design features, or the existing protection from the above design concerns which may be provided by administrative controls or other means, comments received before November 15, 1996, will be considered in developing plans for inspections and other activities associated with the planned regulatory analysis.

If you have any questions regarding this matter, please do not hesitate to contact [PM] at [PM phone].

Sincerely,
PROJECT MANAGER

Docket No(s).

Enclosure: Memo to the Commission, from
J. Taylor, "Resolution of Spent Fuel
Storage Pool Action Plan Issues,"
dated July 26, 1996

cc w/encl: See next page
CONCURRENCE: LA, PM, PD
NOTE: PLEASE INCLUDE J SHEA ON DISTRIBUTION W/O ENCLOSURE.

SAMPLE LETTER FOR: LaSalle 1 and 2
[Licensee Address]

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL SAFETY ISSUES: ISSUANCE OF FINAL STAFF REPORT AND NOTIFICATION OF STAFF PLANS TO PERFORM PLANT-SPECIFIC, SAFETY ENHANCEMENT BACKFIT ANALYSES, **[PLANT NAME, (TAC NO.)]**

Dear .....:

The Nuclear Regulatory Commission staff recently completed a detailed review of spent fuel storage pool safety issues. The results of the staff's review are documented in a report to the Commission which is enclosed for your information. In the report, the staff concludes that existing structures, systems, and components related to the storage of irradiated fuel provide adequate protection of public health and safety.

Notwithstanding this finding, the staff has also identified certain design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. The staff intends to conduct plant-specific regulatory analyses to evaluate potential safety enhancement backfits pursuant to 10 CFR 50.109(a)(3) at a number of operating plants that possess one or more of these design features.

Through the extensive evaluation of loss of spent fuel pool cooling concerns at the Susquehanna Steam Electric Station, the NRC staff identified certain design characteristics that increase the probability that an elevated SFP temperature will interfere with the safe operation of a reactor either at power or shutdown. The first characteristic is an open path from the area around the SFP to areas housing safety systems. This path may be through personnel or equipment access ports, ventilation system ducting, or condensate drain paths. Without an open path, the large surface area of the enclosure around a SFP would allow water vapor to condense and return to the SFP and allow heat to be rejected through the enclosure to the environment without affecting reactor safety systems. The second characteristic is a short time for the SFP to reach elevated temperatures. The time for the SFP to reach an elevated temperature is affected by initial temperature, coolant inventory, and the decay heat rate of irradiated fuel. On the basis of operating practices and administrative limits on SFP temperature, the NRC staff has determined that short times to reach elevated temperatures are credible only when nearly the entire core fuel assembly inventory has been transferred to the SFP and the reactor has been shut down for a short period after extended operation at power.

These conditions establish the third design characteristic, which is a reactor site with multiple operating units sharing structures and systems related to the SFP. At a single-unit site, large coolant inventories in the SFP and in the reactor cavity act as a large passive heat sink for irradiated fuel during fuel transfer. When the entire core fuel assembly inventory has been transferred to the SFP at a single-unit site, safety systems associated with the reactor are not essential because no fuel remains in the reactor vessel. Multi-unit sites with no shared structures can be treated as a single-unit site. At a multi-unit site with shared structures, a short time to reach an elevated temperature can exist in the SFP associated with a reactor in

refueling while safety systems in communication with the area around that SFP are supporting operation of another reactor at power.

When these three design characteristics coexist at a single site, one SFP could reach an elevated temperature in a short time (i.e., between 4 and 10 hours) after a sustained loss of cooling, the heat and water vapor could propagate to systems necessary for shutdown of an operating reactor, and these systems could subsequently fail while needed to support shutdown.

The staff has determined through its survey of SFP design features that these three design characteristics coexist at no more than seven operating reactor sites in addition to Susquehanna. **[Plant Name]** was one of the plants identified. The staff determined through its review of design information and operational controls that immediate regulatory action is not warranted on the basis of the capability of available cooling systems, the passive heat capacity of the SFP, and the operational limits imposed by administrative controls at these seven sites. In making this determination, the staff considered the findings from its review of this issue at Susquehanna. Nevertheless, the staff will conduct detailed reviews to identify enhancements to refueling procedures or cooling system reliability that are justified based on the reduced potential for SFP conditions to impact safety systems supporting an operating reactor at these seven sites, including **[Plant Name]**.

The staff observed that inadequate SFP cooling can be indicated by a high SFP temperature alarm, a SFP cooling system low flow alarm, a cooling system high temperature alarm, or a SFP cooling system pump low discharge pressure alarm. The staff observed that ten operating reactors, including **[Plant Name]** lack a direct-reading high SFP temperature alarm to identify a sustained loss of SFP cooling and, of those ten reactors, one, lacks any associated alarms for a loss of cooling. Because the associated alarms provide annunciation of SFP cooling problem at nine of the operating reactors, because the SFP for the tenth operating reactor is located inside primary containment where equipment is qualified for harsh environments, and because routine operator monitoring also has the potential to detect a loss of the SFP cooling function, the staff determined that immediate regulatory action was not warranted. However, the staff will examine these reactor sites further to determine if additional instrumentation or operational controls are warranted on a safety enhancement basis.

If you wish to comment on the accuracy of the staff's understanding of the plant design, the safety significance of the above design features, the cost of potential modifications to address the above design features, or the existing protection from the above design concerns which may be provided by administrative controls or other means, comments received before November 15, 1996, will be considered in developing plans for inspections and other activities associated with the planned regulatory analysis.

If you have any questions regarding this matter, please do not hesitate to contact [PM] at [PM phone].

Sincere
PROJECT MANAGER

Docket No(s).

Enclosure: Memo to the Commission, from J. Taylor, "Resolution of Spent Fuel Storage Pool Action Plan Issues," dated July 26, 1996

cc w/encl: See next page
CONCURRENCE: LA, PM, PD
NOTE: PLEASE INCLUDE J SHEA ON DISTRIBUTION W/O ENCLOSURE.

SAMPLE LETTER FOR: ANO-2, Prairie Island 1 and 2, Surry 1 and 2, Zion 1 and 2
[Licensee Address]

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL SAFETY ISSUES: ISSUANCE OF FINAL STAFF REPORT AND NOTIFICATION OF STAFF PLANS TO PERFORM PLANT-SPECIFIC, SAFETY ENHANCEMENT BACKFIT ANALYSES, **[PLANT NAME, (TAC NO.)]**

Dear .....:

The Nuclear Regulatory Commission staff recently completed a detailed review of spent fuel storage pool safety issues. The results of the staff's review are documented in a report to the Commission which is enclosed for your information. In the report, the staff concludes that existing structures, systems, and components related to the storage of irradiated fuel provide adequate protection of public health and safety.

Notwithstanding this finding, the staff has also identified certain design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. The staff intends to conduct plant-specific regulatory analyses to evaluate potential safety enhancement backfits pursuant to 10 CFR 50.109(a)(3) at a number of operating plants that possess one or more of these design features.

The SFP cooling system reliability and capability affect the ability of the licensee to maintain SFP temperature within an appropriate band. The staff analyzed design information to determine the susceptibility of SFP cooling systems to a sustained loss of SFP cooling. Specifically, the staff examined the minimum design capacity of the system with no failures, the capacity of the system assuming long-term failure of a single pump, the capacity assuming a LOOP, the passive thermal capacity of the SFP, and the availability of a large-capacity backup system.

On the basis of design information collected through the staff's survey effort and onsite assessment visits, the staff identified events that are most likely to lead to extended reductions in SFP cooling capability. Because the SFP cooling systems typically do not maintain train separation in control cabinets and power cable raceways, events such as fires or internal floods may cause a complete loss of SFP cooling. Also, the primary SFP cooling systems often are designed such that their cooling capacity would be eliminated during a LOOP. However, operators are more likely to recover from minor electrical and control system failures by rerouting power cables and bypassing control cabinets than they are to recover from mechanical failures requiring a unique part for repair in the time available before the SFP reaches elevated temperatures. On this basis, the staff concludes that the operating reactors identified with relatively low cooling capacity that lack redundancy of mechanical components are more likely to experience elevated SFP temperatures than those reactors with greater SFP cooling capacity or mechanical component redundancy. Similarly, those reactors without an onsite source of power to a system capable of cooling the SFP are more likely to experience elevated SFP temperatures than reactors having a cooling system designed to be powered from an onsite power source.

The staff noted that the SFPs at [**Plant Name**] are not capable of being cooled by a system powered from an onsite source without special re-configuration of the electrical distribution system. To address concerns with the reliability and capability of SFP cooling systems, the staff will conduct evaluations and regulatory analyses at selected operating reactors. The staff will examine the capability to supply onsite power to the SFP cooling system relative to the time available for recovery actions at [**Plant Name**] based on procedural controls to determine the need for regulatory analyses.

If you wish to comment on the accuracy of the staff's understanding of the plant design, the safety significance of the above design features, the cost of potential modifications to address the above design features, or the existing protection from the above design concerns which may be provided by administrative controls or other means, comments received before November 15, 1996, will be considered in developing plans for inspections and other activities associated with the planned regulatory analysis.

If you have any questions regarding this matter, please do not hesitate to contact [PM] at [PM phone].

Sincerely,
PROJECT MANAGER

Docket No(s).

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SAMPLE LETTER FOR: Indian Point 3
[Licensee Address]

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL SAFETY ISSUES: ISSUANCE OF FINAL STAFF REPORT AND NOTIFICATION OF STAFF PLANS TO PERFORM PLANT-SPECIFIC, SAFETY ENHANCEMENT BACKFIT ANALYSES, **[PLANT NAME, (TAC NO.)]**

Dear .....:

The Nuclear Regulatory Commission staff recently completed a detailed review of spent fuel storage pool safety issues. The results of the staff's review are documented in a report to the Commission which is enclosed for your information. In the report, the staff concludes that existing structures, systems, and components related to the storage of irradiated fuel provide adequate protection of public health and safety.

Notwithstanding this finding, the staff has also identified certain design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. The staff intends to conduct plant-specific regulatory analyses to evaluate potential safety enhancement backfits pursuant to 10 CFR 50.109(a)(3) at a number of operating plants that possess one or more of these design features.

The SFP cooling system reliability and capability affect the ability of the licensee to maintain SFP temperature within an appropriate band. The staff analyzed design information to determine the susceptibility of SFP cooling systems to a sustained loss of SFP cooling. Specifically, the staff examined the minimum design capacity of the system with no failures, the capacity of the system assuming long-term failure of a single pump, the capacity assuming a LOOP, the passive thermal capacity of the SFP, and the availability of a large-capacity backup system.

The staff noted that the SFPs **[Plant Name]** has a low primary SFP cooling system capacity relative to the potential decay heat generation and has no backup coolng capability. To address concerns with the reliability and capability of SFP cooling systems, the staff will conduct evaluations and regulatory analyses at selected operating reactors. The staff will examine the administrative controls with respect to SFP temperature and available recovery time at **[Plant Name]** to determine the need for regulatory analyses.

If you wish to comment on the accuracy of the staff's understanding of the plant design, the safety significance of the above design features, the cost of potential modifications to address the above design features, or the existing protection from the above design concerns which may be provided by administrative controls or other means, comments received before November 15, 1996, will be considered in developing plans for inspections and other activities associated with the planned regulatory analysis.

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CONCURRENCE: LA, PM, PD
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SAMPLE LETTER FOR: Browns Ferry 2 and 3, Fermi, FitzPatrick, WNP-2
[Licensee Address]

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL SAFETY ISSUES: ISSUANCE OF FINAL STAFF REPORT AND NOTIFICATION OF STAFF PLANS TO PERFORM PLANT-SPECIFIC, SAFETY ENHANCEMENT BACKFIT ANALYSES, **[PLANT NAME, (TAC NO.)]**

Dear .....:

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The staff noted that **[Plant Name]** appears to be reliant on infrequently operated backup SFP cooling systems to address long-term LOOP events and mechanical failures. The staff will examine administrative controls on the availability of the backup cooling systems during refueling and technical analyses demonstrating the capability of these backup systems to cool the SFP at **[Plant Name]** to determine the need for further regulatory analyses.

If you wish to comment on the accuracy of the staff's understanding of the plant design, the safety significance of the above design features, the cost of potential modifications to address the above design features, or the existing protection from the above design concerns which may be provided by administrative controls or other means, comments received before November 15, 1996, will be considered in developing plans for inspections and other activities associated with the planned regulatory analysis.

If you have any questions regarding this matter, please do not hesitate to contact [PM] at [PM phone].

Sincerely,
PROJECT MANAGER

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SAMPLE LETTER FOR: ANO-1, Brunswick 1 and 2, Cooper, Millstone 1
[Licensee Address]

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL SAFETY ISSUES: ISSUANCE OF FINAL STAFF REPORT AND NOTIFICATION OF STAFF PLANS TO PERFORM PLANT-SPECIFIC, SAFETY ENHANCEMENT BACKFIT ANALYSES, **[PLANT NAME, (TAC NO.)]**

Dear .....:

The Nuclear Regulatory Commission staff recently completed a detailed review of spent fuel storage pool safety issues. The results of the staff's review are documented in a report to the Commission which is enclosed for your information. In the report, the staff concludes that existing structures, systems, and components related to the storage of irradiated fuel provide adequate protection of public health and safety.

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The staff observed that inadequate SFP cooling can be indicated by a high SFP temperature alarm, a SFP cooling system low flow alarm, a cooling system high temperature alarm, or a SFP cooling system pump low discharge pressure alarm. The staff observed that ten operating reactors, including **[Plant Name]** lack a direct-reading high SFP temperature alarm to identify a sustained loss of SFP cooling and, of those ten reactors, one, lacks any associated alarms for a loss of cooling. Because the associated alarms provide annunciation of SFP cooling problem at nine of the operating reactors, because the SFP for the tenth operating reactor is located inside primary containment where equipment is qualified for harsh environments, and because routine operator monitoring also has the potential to detect a loss of the SFP cooling function, the staff determined that immediate regulatory action was not warranted. However, the staff will examine these reactor sites further to determine if additional instrumentation or operational controls are warranted on a safety enhancement basis.

If you wish to comment on the accuracy of the staff's understanding of the plant design, the safety significance of the above design features, the cost of potential modifications to address the above design features, or the existing protection from the above design concerns which may be provided by administrative controls or other means, comments received before November 15, 1996, will be considered in developing plans for inspections and other activities associated with the planned regulatory analysis.

If you have any questions regarding this matter, please do not hesitate to contact [PM] at [PM phone].

Sincerely,
PROJECT MANAGER

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SAMPLE LETTER: For All Plants Other Than Those Listed in Attachment 2

[Licensee Address]

SUBJECT: RESOLUTION OF SPENT FUEL STORAGE POOL SAFETY ISSUES: ISSUANCE OF FINAL REPORT, [PLANT NAME] (TAC NO. M88094)

Dear .....:

The Nuclear Regulatory Commission staff recently completed a detailed review of spent fuel storage pool safety issues. The results of the staff's review are documented in a report to the Commission which is enclosed for your information. In the report, the staff concludes that existing structures, systems, and components related to the storage of irradiated fuel provide adequate protection of public health and safety.

Notwithstanding this finding, the staff has also identified certain design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. The staff intends to conduct plant-specific regulatory analyses to evaluate potential safety enhancement backfits pursuant to 10 CFR 50.109(a)(3) at a number of operating plants that possess one or more of these design features.

Although [PLANT NAME] is not identified for a plant-specific safety enhancement backfit analysis by the staff, it is requested that you review the enclosed report for applicability to your facility and consider actions, as appropriate, related to the design of spent fuel pool decay heat removal systems at your facility. This letter does not require any response.

If you have any questions regarding this matter, please do not hesitate to contact [PM] at [PM phone].

Sincerely,
Project Manager

Docket No(s).

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Attachment 4

REGION: 2

SITES:

ABBR	NAME
STL	ST LUCIE

DOCKETS:

NUMBER	NAME

EA-CASE:
NUMBERS

ALL

ORGANIZATION:

CODE	NAME

EMPLOYEES:

ID	NAME
ALL	** ALL EMPLOYEES **

REPORT ON: ALL
INSPR-NO: ALL
CAUSE CD: A
STATUS: OPEN
REACTOR TYPE: POWER
COMMENT TEXT IS REQUESTED

FROM DATE:
TO DATE:
ITEM TYPE: ALL
SEVERITY CD: A
FUNCTIONAL AREA: ALL
ENFORCEMENT IND: ALL

FF/11

ITEM DETAIL REPORTS   IFS - INSPECTION FOLLOW-UP SYSTEM   DATE: 08/23/96
ST LUCIE OPEN ITEMS LIST   REPORT BY SITE   TIME: 10:59:31
                                                                                            PAGE: 2

SITE: STL ST LUCIE

UNIT ABBR | RPT/IFS/ EA/ NBR | SEQ-NO NOV-ID | ITEM TYPE | REF NBR / EA-NBR | SEVERITY SUPLMNT | SALP AREA | REPORT / EVENT DT | STS | CREATE DATE | CLOSEOUT PRJ/ACT* | CLSOUT ORG NO | CLSOUT EMP | UPDATING INSPECTION REPORTS

STL2 I 93-025 01 IFI 12/01/93 0 12/07/93 2350
TITLE: REVIEW OPERABILITY OF UNIT 2 MOV MV-08-13 DURING THE PROC NUMBER: 37700
COMMENTS: REVIEW OPERABILITY OF UNIT 2 MOV MV-08-13 DURING THE PERIOD
JULY 20 TO OCTOBER 19, 1993.

STL1 I 94-008 03 URI ENG 04/08/94 0 04/13/94 2320 JVL
TITLE: QUALITY LEVEL OF PORV AND SRV DISCHARGE PIPING PROC NUMBER: 37700

STL1 I 94-011 01 VIO 4/1 ENG 06/13/94 0 06/08/94 2350
TITLE: INADEQUATE CORRECTIVE ACTION FOR MOVS WHICH STALLED PROC NUMBER: 2515/[illegible]

STL1 I 94-011 02 IFI ENG 06/13/94 0 06/08/94 2350
TITLE: INADEQUATE RECOGNITION OF MOV TEST PRESSURE AND FLOW PROC NUMBER: 2515/109

STL1 I 94-011 03 IFI ENG 06/13/94 0 06/08/94 2350
TITLE: LACK OF INSTRUCTIONS OR GUIDANCE FOR TRENDING PROC NUMBER: 2515/109

STL1 I 94-300 01 IFI OPS 11/17/94 0 11/28/94 2330 96-004
TITLE: PROCEDURAL GUIDANCE FOR REMOVAL OF RCPS PRIOR TO 500 PROC NUMBER: 71707
COMMENTS: PROCEDURAL GUIDANCE IS LACKING FOR REMOVAL OF AN RCP FROM
SERVICE PRIOR TO RCS TEMPERATURE DECREASING BELOW 500 DEGREES
FAHRENHEIT. PLANT PARAMETERS CAN CHANGE FAST ENOUGH SUCH
THAT PROCEDURAL GUIDANCE IS NOT OBTAINED UNTIL AFTER THE RCS
TEMPERATURE HAS DECREASED BELOW THE 500 DEGREES FAHRENHEIT.

STL2 N 94-332 LER 94-006-01 07/14/94 0 10/11/94 2230 HAS BEEN CLOSED IR 94-24 (Rev 0) 12/14/94
TITLE: TRIP CIRCUIT BREAKER FAILURE DUE TO A BROKEN PIECE OF PROC NUMBER:

STL1 N 95-005 LER 94-009-00 11/22/94 0 01/04/95 2230 REV 1 CLOSED IR 95-09
TITLE: INADVERTENT SAFETY INJECTION ACTUATION SIGNAL/CONTAINME PROC NUMBER:
COMMENTS: INADVERTENT SAFETY INJECTION ACTUATION SIGNAL/CONTAINMENT
ISOLATION SIGNAL DUE TO AN INVALID HIGH PRESSURIZER PRESSURE
SIGNAL.

STL1 I 95-015 05 VIO 4/1 OPS 10/16/95 0 10/24/95 2230 95-015
TITLE: FAILURE TO FOLLOW PROCEDURE AND DOCUMENT A DEFICIENCY ON PROC NUMBER: 61726
COMMENTS: ON CONTAINMENT SPRAY VALVE SURVEILLANCE TEST. THE FAILURE TO
DOCUMENT FINDINGS VIA THE STAR PROCESS.

STL1 I 95-015 07 VIO 4/1 OPS 10/16/95 0 10/24/95 2230 95-015
TITLE: FAILURE TO FOLLOW PROCEDURES DURING VENTING OF ECCS PROC NUMBER: 71707

SITE: STL ST LUCIE

UNIT ABBR	RPT/IFS/ EA/ NBR	SEQ-NO NOV-ID	ITEM TYPE	REF NBR / EA-NBR	SEVERITY SUPLMNT	SALP AREA	REPORT / EVENT DT	STS	CREATE DATE	CLOSEOUT PRJ/ACT*	CLSOUT ORG NO	CLSOUT EMP	UPDATING INSPECTION REPORTS

COMMENTS: THE FAILURE TO INCLUDE APPROPRIATE INITIAL CONDITIONS IN OP 1-0430060.

STL1 I 95-016 01 EEI 95-180 /1 09/08/95 0 09/13/95 2230
TITLE: LTOP INOPERABILITY DUE TO PORV FAILURE PROC NUMBER: 92700
STL2 I 95-016 01 EEI 95-180 /1 09/08/95 0 09/13/95 2230

SHOULD BE CLOSED TO VIO

STL1 I 95-017 01 VIO 4/1 ENG 01/05/96 0 01/11/96 2320
TITLE: FAILURE TO FOLLOW PROCEDURES FOR MATERIAL CONTROLS-3 EXAMPLES PROC NUMBER: 38703
COMMENTS: EXAMPLE 1-FAILURE TO PERFORM COMMERCIAL GRADE DEDICATION TEST. EXAMPLE 2-FAILURE TO INCORPORATE SHELF LIFE MAINTENANCE REQUIREMENT INTO IN-STORAGE MAINTENANCE PROGRAM. EXAMPLE 3-FAILURE TO DISPOSITION A SUPPLIER DEVIATION AS DIRECTE BY THE ASSOCIATED ENGINEERING EVALUATION.
STL2 I 95-017 01 VIO 4/1 ENG 01/05/96 0 01/11/96 2320

STL1 I 95-018 01 VIO 4/1 OPS 11/28/95 0 12/07/95 2230 95-018 VERIFY ACTUAL VIO
TITLE: FAILURE TO FOLLOW PROCEDURES AND MAINTAIN CURRENT PROC NUMBER: 71707
COMMENTS: CONTRARY TO THE ABOVE ON SEPTEMBER 15, 1995, DURING THE CLEANING OF UNIT 2 CONDENSER WATER BOXES, THE 2B2 WATERBOX MANWAY WAS REMOVED TO REPLACE A LEAKING GASKET WITHOUT IMPLEMENTING A CLEARANCE. WHEN THE MAINTENANCE FOREMAN AND MECHANIC ATTEMPTED TO REMOVE THE MANWAY COVER, THE NEGATIVE PRESSURE THAT EXISTED ACROSS THE MANWAY SUCKED THE COVER BACK ON THE WATERBOX AND SEVERED A PORTION OF THE MECHANICS FINGER.

OPERATORS FAILED TO PROPERLY LOG THE ISSUANCE OF AN AFAS BYPASS KEY. OPERATORS FUTHER FAILED TO PROPERLY REVIEW KEY LOG.

STL1 I 95-018 02 VIO 4/1 OPS 11/28/95 0 12/07/95 2230 95-018
TITLE: FAILURE TO FOLLOW CLEARANCE PROCEDURES. PROC NUMBER: 71707
COMMENTS: CONTRARY TO THE ABOVE, THE LICENSEE IMPLEMENTED A UNIT 2 EMERGENCY DIESEL GENERATOR (EDG) CONTROL LOGIC DESIGN THAT DID NOT TRIP THE EDG OUTPUT BREAKER ON RECEIPT OF A CSAS OR CIAS SIGNAL WHEN PARALLELLED WITH OFFSITE POWER. THIS INADEQUATE DESIGN RESULTED IN SHIFTING THE GOVERNOR TO THE ISOCHRONOUS MODE, BYPASSING ALL PROTECTIVE RELAYS EXCEPT OVERSPEED AND DIFFERENTIAL CURRENT DURING INTEGRATED SAFEGUARDS TESTING ON OCTOBER 12, 1995. THIS RESULTED IN OPERATING THE EDG AS A SYNCHRONOUS MOTOR FOR APPROX. 45 SECONDS UNTIL THE CIAS SIGNAL WAS RESET. OPERATION IN THE ISOCHRONOUS MODE WHILE PARALLELED WITH OFFSITE POWER COULD EXPOSE THE ENGIENE AND GENERATOR TO EXCESSIVE MECHANICAL STRESS OR ELECTRICAL OVERCURRENT CONDITIONS.

VERIFY WHERE THIS BELONGS

STL2 I 95-018 03 VIO 4/1 MAINT 11/28/95 0 12/07/95 2230 95-018
TITLE: FAILURE TO ADEQUATELY DESIGN AND TEST THE EMERGENCY PROC NUMBER: 61726

SITE: STL ST LUCIE

UNIT ABBR	RPT/IFS/ EA/ NBR	SEQ-NO NOV-ID	ITEM TYPE	REF NBR / EA-NBR	SEVERITY SUPLMNT	SALP AREA	REPORT / EVENT DT	STS	CREATE DATE	CLOSEOUT PRJ/ACT*	CLSOUT ORG NO	CLSOUT EMP	UPDATING INSPECTION REPORTS
STL1	I 95-020	01	EEI	95-222	/1	ENG	10/26/95	0	11/16/95		2230		95-020

TITLE: FAILURE TO TAKE PROMPT CORRECTIVE ACTIONS FOR RELIEF PROC NUMBER: 92700
COMMENTS: VALVE DEFICIENCIES.

STL2	I 95-020	01	EEI	95-222	/1	ENG	10/26/95	0	11/16/95		2230		95-020

SHOULD BE CLOSED OUT TO VIO

STL2	I 95-021	01	VIO		4/1	OPS	12/08/95	0	12/28/95		2230		95-021

TITLE: FAILURE TO FOLLOW CLEARANCE PROCEDURE'S PROC NUMBER: 71707
COMMENTS: CONTRARY TO THE ABOVE, ON NOVEMBER 11, 1995, EQUIPMENT CLEARANCE ORDER 2-95-11-128 WAS ISSUED REQUIRING NO INDEPENDENT VERIFICATI N THAT THE 125 VDC POWER SUOPPLY BREAKER TO BLOCK VALVE MOV-08-1 7 FOR ATMOSPHERIC DUMP MV-08-19B WAS IN THE REQUIRED POSITION OF F. AN OUT-OF-SERVICE LOG ENTRY ON NOVEMBER 5, IDENTIFIED MV-08-17 AS TECH SPEC. EQUIPMENT REQUIRED PRIOR TO MODE 1 AS "OUT ON CLEARANCE."

STL2	I 95-021	02	VIO		4/1	OPS	12/08/95	0	12/28/95		2230		95-021

TITLE: FAILURE TO FOLLOW THE EQUIPMENT CLEARANCE ORDER PROCEDURE PROC NUMBER: 71707
COMMENTS: CONTRARY TO THE ABOVE, ON NOVEMBER 20, 1995 V09120 ISOLATION VAL VE FOR AFQ PUMP 2A TO STEM GENERATOR 2A FEEDWATER INLET SPECIFI-CIED ON EQUIPMENT CLEARANCE ORDER 2-95-09-062 TAG #13 ISSUED NOVEMBER 17 WAS FOUND IN THE CLOSED INSTEAD OF REQUIRED LOCKED C LOSED POSTITION. THIS VALVE ALSO REQUIRED INDEPENDENT VERIFI-CATION.

STL2	I 95-021	03	VIO		4/1	MAINT	12/08/95	0	12/28/95		2230		95-021

TITLE: FAILURE TO PERFORM RCP SYSTEM BORON SURVEILLANCE PROC NUMBER: 61726
COMMENTS: CONTRARY TO THE ABOVE, UNIT 2 ENTERED MODE 5 AT 3:10 AM ON NOV. 27, 1995 WHICH REQUIRED THAT THE SAMPLING PERIODICITY BE IN-CREASED TO EVERY EIGHT HRS. SINCE THE PAST PREVIOUS SAMPLE WAS AT 1:00 AM ON OCTOBER 27, THE NEXT SAMPLE WAS DUE AT 9:00 AM AND EVERY EIGHT HRS THEREAFTER. AT 11:45 PM OPERATORS DISCOVERED THAT THE SAMPLING HAD NOT BEEN ACCOMPLISHED AT THE REQUIRED TIME ON THE DAY AND PEAK SHIFT. CHEMISTRY WAS NOTIFIED AND THE SAMPLE WAS TAKEN AND ANALYZED AT 12:00 PM ON NOVEMBER 27.

STL1	I 95-022	01	IFI				01/09/96	0	01/17/96		2320		

TITLE: COMPLETION OF CERTIFICATION PROGRAM FOR QUALIFIED SAFETY PROC NUMBER: 37001

STL2	I 95-022	01	IFI				01/09/96	0	01/17/96		2320		
STL1	I 95-022	03	IFI			ENG	01/09/96	0	02/28/96		2320		

TITLE: SG LEVEL CHANNEL IN ACCURACEIS DUE TO SENSING LINE BLOCKAGE. PROC NUMBER: 93702

STL2	I 95-022	03	IFI			ENG	01/09/96	0	02/28/96		2320		

SHOULDN'T THIS BE OURS?

SITE: STL ST LUCIE

UNIT ABBR	RPT/IFS/ EA/ NBR	SEQ-NO NOV-ID	ITEM TYPE	REF NBR / EA-NBR	SEVERITY SUPLMNT	SALP AREA	REPORT / EVENT DT	STS	CREATE DATE	CLOSEOUT PRJ/ACT*	CLSOUT ORG NO	CLSOUT EMP	UPDATING INSPECTION REPORTS
STL1	N 95-181		LER	95-003-00			07/08/95	O	08/10/95		2230		95-003
STL1	N 95-195		LER	95-005-00			08/09/95	O	08/28/95		2230		
STL1	N 95-196		LER	95-006-00			08/10/95	O	08/28/95		2320		
STL1	N 95-205		LER	95-007-00			08/17/95	O	08/31/95		2230		
STL1	N 95-206		LER	95-004-00			08/01/95	O	08/31/95		2230		
STL1	N 95-231		LER	95-003-01			07/08/95	O	10/17/95		2230		
STL1	N 95-233		LER	95-008-00			08/28/95	O	10/19/95		2230		
STL2	N 95-254		LER	95-004-00			10/10/95	O	11/07/95		2230		
STL1	N 95-268		LER	95-009-00			10/19/95	O	11/24/95		2230		
STL1	N 95-297		LER	95-010-00			11/16/95	O	12/19/95		2230		
STL2	N 96-001		LER	95-005-00			11/20/95	O	01/03/96		2230		

STL1 N 95-181 LER 95-003-00
TITLE: AUTOMATIC REACTOR TRIP DURING TURBINE OVERSPEED SURVEILLANCE PROC NUMBER:
COMMENTS: SURVEILLANCE TESTING DUE TO PERSONNEL ERROR.

STL1 N 95-195 LER 95-005-00
TITLE: PRESSURIZER POWER OPERATED RELIEF VALVES (PORV) PROC NUMBER:
COMMENTS: INOPERABLE DUE TO PERSONNEL ERROR.

STL1 N 95-196 LER 95-006-00
TITLE: LOSS OF REACTOR COOLANT INVENTORY THROUGH A SHUTDOWN PROC NUMBER:
COMMENTS: COOLING RELIEF VALVE DUE TO LACK OF DESIGN MARGIN.

STL1 N 95-205 LER 95-007-00
TITLE: INADVERTENT CONTAINMENT SPRAY VIA 1A LOW PRESSURE PROC NUMBER:
COMMENTS: SAFETY INJECTION PUMP WHILE VENTING THE EMERGENCY CORE COOLING SYSTEM DURING STARTUP DUE TO INADEQUATE PROCEDURE.

STL1 N 95-206 LER 95-004-00
TITLE: HURRICANE ERIN AT ST. LUCIE PROC NUMBER:

STL1 N 95-231 LER 95-003-01
TITLE: AUTOMATIC REACTOR TRIP DURING TURBINE OVERSPEED PROC NUMBER:
COMMENTS: SURVEILLANCE TESTING DUE TO PERSONNEL ERROR.

STL1 N 95-233 LER 95-008-00
TITLE: HIGH PRESSURE SAFETY INJECTION PUMP OPERATION DURING PROC NUMBER:
COMMENTS: PLANT CONDITIONS NOT ALLOWED BY TECHNICAL SPECIFICATIONS DUE TO PERSONNEL ERROR.

STL2 N 95-254 LER 95-004-00
TITLE: REACTOR COOLANT SYSTEM INSTRUMENT NOZZLE LEAKAGE CAUSED PROC NUMBER:
COMMENTS: BY PRIMARY WATER STRESS CORROSION CRACKING.

STL1 N 95-268 LER 95-009-00
TITLE: MISSED TECHNICAL SPECIFICATION SCHEDULED SURVEILLANCE PROC NUMBER:
COMMENTS: DUE TO PERSONNEL ERROR.

STL1 N 95-297 LER 95-010-00
TITLE: MANUALLLL REACTOR TRIP DUE TO LOW STEAM GENERATOR WATER PROC NUMBER:
COMMENTS: LEVEL CAUSED BY MAIN FEEDWAER REGULATING VALVE CLOSURE.

STL2 N 96-001 LER 95-005-00
TITLE: 2A EMERENCY DIESEL GENERATOR RELAY SOCKET FAILURES DUE PROC NUMBER:

SITE: STL ST LUCIE

UNIT ABBR	RPT/IFS/ EA/ NBR	SEQ-NO NOV-ID	ITEM TYPE	REF NBR / EA-NBR	SEVERITY SUPLMNT	SALP AREA	REPORT / EVENT DT	STS	CREATE DATE	CLOSEOUT PRJ/ACT*	CLSOUT ORG NO	CLSOUT EMP	UPDATING INSPECTION REPORTS

COMMENTS: TO HIGH CYCLE FATIGUE.

STL2 I 96-001 01 VIO 4/1 OPS 03/18/96 0 03/22/96 2320
TITLE: TEMPORARY CHANGES TO PROCEDURES IMPROPERLY CHANGED PROC NUMBER: 40500
COMMENTS: CONTRARY TO THE ABOVE, THE LICENSEE MADE TEMPORARY CHANGES TO PROCEDURES REQUIRED BY TECHNICAL SPECIFICATION 6.8.1, WITHOUT PRIOR REVIEW BY THE FRG OR APPROVAL BY THE PLANT GENERAL MANAGER THAT ALTERED THE INTENT OT THE ORIGINAL PROCEDURES.

STL1 I 96-001 02 URI MAINT 03/18/96 0 03/22/96 2320 96-001
ISN'T THIS OURS, CLOSED TO NCV?
TITLE: IMPROPER HEALTH PHYSICS PRACTICES. PROC NUMBER: 62703
COMMENTS: WORKERS IDENTIFIED TO BE WORKING IN CONTAMINATED AREA WITHOUT PROPERLY EMPLOYING RWP REQUIREMENTS.

STL1 I 96-003 01 EEI 96-040 /1 OPS 02/22/96 0 02/28/96 2320
TITLE: OPERATORS FAILED TO FOLLOW PROCEDURES FOR BORON DILUTION PROC NUMBER: 92700
COMMENTS: WATCH TURNOVER, PROCEDURE ADHERENCE, AND EVENT REPORTING.
STL2 I 96-003 01 EEI 96-040 /1 OPS 02/22/96 0 02/28/96 2320

STL1 I 96-003 02 EEI 96-040 /1 OPS 02/22/96 0 02/28/96 2320
TITLE: INADEQUATE DESIGN CONTROL OF REACTOR COOLANT SYSTEM BORON PROC NUMBER: 92700
COMMENTS: DILUTUION PROCEDURE.
STL2 I 96-003 02 EEI 96-040 /1 OPS 02/22/96 0 02/28/96 2320

SHOULDN'T THESE BE CLOSED TO VIOS?

STL1 I 96-003 03 EEI 96-040 /1 OPS 02/22/96 0 02/28/96 2320
TITLE: INADEQUAE 10 CFR 50.59 SAFETY EVALUATION OF CHANGE TO PROC NUMBER: 92700
COMMENTS: BORON DILUTUION PROCEDURE.
STL2 I 96-003 03 EEI 96-040 /1 OPS 02/22/96 0 02/28/96 2320

STL1 I 96-004 01 VIO 4/1 OPS 04/29/96 0 05/08/96 2230
TITLE: FAILURE TO FOLLOW PROCEDURES LEAD TO UNIT 1 CONTAINMENT PIG PROC NUMBER: 71707
COMMENTS: HP TECH FAILED TO PROPERLY EMPLOY PROCEDURE FOR GRAB SAMPLE OF U-1 CONTAINMENT. NON-LICENSED OPERATORS FAILED TO QUESTION & RESOLVE RESULTING OUT OF SPECIFICATION DATA, AS REQ'D BY PROCEDURE.

STL1 I 96-004 02 VIO 4/1 OPS 04/29/96 0 05/08/96 2230
TITLE: FAILURE TO MAKE REQUIRED LOG ENTRIES PROC NUMBER: 71707
COMMENTS: OPERATORS FAILED TO MAKE CHRONOLOGICAL LOG ENTRIES AS REQUIRED BY PROCEDURE.

STL1 I 96-004 03 VIO 4/1 OPS 04/29/96 0 05/08/96 2230
TITLE: FAILURE TO FOLLOW PROCEDURES RESULTS IN EMERGENCY DIESEL PROC NUMBER: 71707

SITE: STL ST LUCIE

UNIT ABBR	RPT/IFS/ EA/ NBR	SEQ-NO NOV-ID	ITEM TYPE	REF NBR / EA-NBR	SEVERITY SUPLMNT	SALP AREA	REPORT / EVENT DT	STS	CREATE DATE	CLOSEOUT PRJ/ACT*	CLSOUT ORG NO	CLSOUT EMP	UPDATING INSPECTION REPORTS
COMMENTS:	OPERATOR PLACED EDG FOST ON RECIRC WITHOUT ADHERING TO PROCEDURE ERROR RESULTED IN ISOLATING TANK. SAME OPERATOR FAILED TO INFORM COUNTROL ROOM OF HIS ACTION, AS REQUIRED BY PROCEDURE.												
STL1	I 96-004	04	VIO		4/1	MAINT	04/29/96	0	05/08/96		2230		
TITLE:	FAILURE TO ADQUATELY REVIEW TEST DATA AND IDENTIFY									PROC NUMBER: 62703			
COMMENTS:	I& C SUPERVISOR FAILED TO ADEQUATELY REVIEW TEST DATA AND IDENTIFY OUT-OF-SPEC CONDITIONS.												
STL1	I 96-004	05	URI			OPS	04/29/96	0	05/08/96		2230		CLOSE TO EE1, CLOSE TO VIO
TITLE:	CONFIGURATION CONTROL MANAGEMENT									PROC NUMBER: 71707			
STL2	I 96-004	05	URI			OPS	04/29/96	0	05/08/96		2230		
STL1	I 96-004	09	URI			ENG	04/29/96	0	05/08/96		2230		
TITLE:	FAILURE TO UPDATE FSAR									PROC NUMBER: 71707			
STL2	I 96-004	09	URI			ENG	04/29/96	0	05/08/96		2230		
STL1	I 96-006	05	VIO		4/1	PLTSUP	06/07/96	0	06/26/96		2230		
TITLE:	FAILURE TO MAINTAIN QUALIFICATIONS OF FIRE BRIGADE MEMBERS									PROC NUMBER: 92904			
COMMENTS:	DURING APRIL 1996 9 OF 11 FIRE TEAM MEMBERS WITH EXPIRED MEDICAL WERE ASSIGNED 60 SHIFTS. 2 FIRE TEAM LEADERS NOT ON ROSTER WERE ASSIGNED 31 SHIFTS. ONE FIRE TEAM MEMBER WITH EXPIRED MEDICAL AND NOT ON ROSTER WAS ASSIGNED ONE SHIFT.												
STL2	I 96-006	05	VIO		4/1	PLTSUP	06/07/96	0	06/26/96		2230		
STL1	I 96-008	03	URI			MAINT	07/08/96	0	07/18/96		2230		
TITLE:	ADEQUACY OF DOCUMENTATION FOR REPEATED PROCEDURE STEPS.									PROC NUMBER: 62703			
STL2	I 96-008	03	URI			MAINT	07/08/96	0	07/18/96		2230		
STL1	I 96-008	05	URI			ENG	07/08/96	0	07/18/96		2230		
TITLE:	LICENSEE IDENTIFIED 11 UFSAR DEFICIENCIES									PROC NUMBER: 37551			
STL2	I 96-008	05	URI			ENG	07/08/96	0	07/18/96		2230		
STL1	I 96-008	06	VIO		4/1	OPS	07/08/96	0	07/18/96		2230		SHOULD GO AWAY
TITLE:	FAILURE TO FOLLOW PROCEDURE DURING EDG TESTING									PROC NUMBER: 71707			
COMMENTS:	FAILURE TO RECORD COOLING WATER OUTLET TEMPERATURE EVERYY THIRTH Y MINUTES AS REQUIRED BY OP-2200508.												
STL1	I 96-010	01	IFI			PLTSUP	07/26/96	0	08/02/96		2350		
TITLE:	FFD STAFF REORG AND CORRECTIVE ACTION									PROC NUMBER: 81700			
STL2	I 96-010	01	IFI			PLTSUP	07/26/96	0	08/02/96		2350		
STL1	N 96-067		LER	96-001-00			02/19/96	0	03/20/96		2230		
TITLE:	CONTROL ROOM EMERGENCY VENTILATION SYSTEM INOPERABLE									PROC NUMBER:			

SITE: STL ST LUCIE

UNIT ABBR	RPT/IFS/ EA/ NBR	SEQ-NO NOV-ID	ITEM TYPE	REF NBR / EA-NBR	SEVERITY SUPLMNT	SALP REPORT / AREA EVENT DT	STS	CREATE DATE	CLOSEOUT PRJ/ACT*	CLSOUT ORG NO	CLSOUT EMP	UPDATING INSPECTION REPORTS

COMMENTS: DUE TO IMPROPER SYSTEM CONFIGURATION.

STL2 N 96-071 LER 96-001-00 01/05/96 0 03/20/96 2230
TITLE: MANUAL REACTOR TRIP DUE TO HIGH MAIN GENERATOR COLD PROC NUMBER:
COMMENTS: GAS TEMPERATURE.

STL1 N 96-086 LER 96-003-00 02/24/96 0 03/29/96 2230 ALREADY IN VIO - TIE THE TWO TOGETHER
TITLE: CONTAINMENT PARTICULATE AND GASEOUS MONITOR OUT OF PROC NUMBER:
COMMENTS: SERVICE RESULTING IN A CONDITION PROHIBITED BY TECHNICAL
SPECIFICATIONS DUE TO PERSONNEL ERROR.

STL1 N 96-089 LER 96-004-00 02/27/96 0 04/04/96 2230
TITLE: INADVERTENT MANUAL START OF THE 1A EMERGENCY DIESEL PROC NUMBER:
COMMENTS: GENERATOR DUE TO PERSONNEL ERROR.

STL1 N 96-202 LER 96-005-00 05/14/96 0 06/28/96 2230
TITLE: WIDE RANGE NUCLEAR INSTRUMENTATION CHANNEL INOPERABLE PROC NUMBER:
COMMENTS: WHEN REQUIRED TO BE IN SERVICE FOR FUEL MOVEMENT.

STL1 N 96-211 LER 96-006-00 06/01/96 0 07/09/96 2230
TITLE: INADVERTENT LOSS OF CONTAINMENT AUDIBLE COUNT RATE PROC NUMBER:
COMMENTS: INDICATION DUE TO PROCEDURAL DEFICIENCY.

STL2 N 96-217 LER 96-002-00 06/06/96 0 07/15/96 2230
TITLE: MANUAL REACTOR TRIP DUE TO HIGH MAIN GENERATOR COLD GAS PROC NUMBER:
COMMENTS: TEMPERATURE CAUSED BY VALVE FAILURE.

STL1 N 96-219 LER 96-008-00 06/08/96 0 07/15/96 2230
TITLE: INADVERTANT ACTUATION OF THE SAFETY INJECTION ACTUATION PROC NUMBER:
COMMENTS: SIGNAL DUE TO LOSS OF THE 15 VDC REGULATED POWER SUPPLY
DURING MAINTENANCE.

STL1 N 96-221 LER 96-007-00 06/07/96 0 07/17/96 2230
TITLE: INADVERTENET START OF THE 1B EMERGENCY DIESEL GENERATOR DURING PROC NUMBER:
COMMENTS: DURING "B" CHANNEL CONTAINMENT ISOLATION ACTUATION SIGNAL TEST-
ING DUE TO PROCEDURAL INADEQUACY.

STL2 N 96-238 LER 96-003-00 06/25/96 0 08/12/96 2230
TITLE: SAFETY INJECTION TANKS DISCHARGE VALVES PROCEDURALLY ISOLATED PROC NUMBER:
COMMENTS: ISOLATED IN MODE 4 DUE TO PERSONNEL ERROR.

NOTE: DEFINITION FOR CHARACTER PRECEEDING REPORT NO: I = INSPECTION REPORT NUMBER, E = EA NUMBER (ENFORCEMENT / NOV ITEM)
N = IFS NUMBER, NUMBER USED TO IDENTIFY NON-INSPECTION ITEMS

TOTAL OPEN ITEMS ===============>> 72
TOTAL OPEN REPORT SEQUENCES ====>> 58

*IF ITEM IS OPEN, PROJECTED CLOSEOUT DATE IS REPORTED
IF ITEM IS CLOSED, ACTUAL CLOSEOUT DATE IS REPORTED

August 5, 1996

EA 96-263

Florida Power & Light Company
ATTN: T. F. Plunkett
President - Nuclear Division
P. O. Box 14000
Juno Beach, Florida 33408-0420

SUBJECT: NRC INTEGRATED INSPECTION REPORT 50-335/96-09, 50-389/96-09 and NOTICE OF VIOLATION.

Dear Mr. Plunkett:

On July 6, 1996, the NRC completed an inspection at your St. Lucie 1 and 2 reactor facilities. The enclosed report presents the results of that inspection.

Based on the results of this inspection, the NRC has determined that violations of NRC requirements occurred. The violations are of concern because they collectively indicate that weaknesses in regulatory and procedural compliance extend across organizational lines, to include management, maintenance planning and performance, and health physics.

An area of particular concern involves the lack of control over the use of overtime by personnel performing safety-related activities. The number of examples identified in a relatively small population of personnel audited indicates that management was ineffective in controlling the excessive use of unapproved overtime. This indicated a weakness in management commitment to maintaining overtime usage at acceptable levels in order to minimize human error.

In addition, an element of your contamination control program is of concern, in that it had not been effective in preventing the improper use of contaminated tools.

The violations are cited in the enclosed Notice of Violation, and the circumstances surrounding the violations are described in detail in the enclosed report. Please note that you are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

FF/12

~~9608230195~~ 3PP

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room (PDR).

Sincerely,

Kerry D. Landis, Chief
Reactor Projects Branch 3
Division of Reactor Projects

Docket Nos. 50-335, 50-389
License Nos. DPR-67, NPF-16

Enclosures: Notice of Violation
Inspection Report 50-335/96-09, 50-389/96-09

cc w/encl:
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cc w/encl continued: See page 3

cc w/encl: Continued
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Distribution w/encl: See page 4

NOTICE OF VIOLATION

Florida Power & Light Company
St. Lucie 1 and 2

Docket Nos. 50-335 and 50-389
License Nos. DPR-67 and NPF-16
EA 96-263

During an NRC inspection conducted on June 9 through July 6, 1996, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," (60 FR 34381; June 30, 1995/NUREG 1600), the violations are listed below:

A. Technical Specification 6.2.f, requires that the hours expended by personnel performing safety-related functions be limited and that during extended periods of shutdown for refueling, the following guidelines be observed:

- An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

- An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time.

The Specification further requires that any deviations from the above guidelines be authorized by the Plant General Manager or his deputy, or higher levels of management, in accordance with established plant procedures and with documentation of the basis for the deviation. AP 0010119, Revision 14, "Overtime Limitations for Plant Personnel," implemented this requirement and provided an administrative vehicle for the approval of deviations from the specified guidelines.

Contrary to the above, during the period from May 13 through June 14, 1996, five individuals who performed safety related functions were found to have contributed to 38 deviations from the 72-hour-in-any-seven-day-period requirement, 15 deviations from the 24-hour-in-any-48-hour requirement, and 3 deviations from the 16-hour-in-any-24-hour-requirement without obtaining authorization from the Plant General Manager, his deputy, or higher levels of management.

This is a Severity Level IV violation (Supplement I).

B. Technical Specification 6.2.f requires, in part, that deviations from overtime guidelines be approved in accordance with established procedures and that controls be included in established procedures such that individual overtime be reviewed monthly by the Plant General Manager or his designee to assure that excessive hours have not been assigned.

ENCLOSURE 1

~~9608230199~~ 6PP

Contrary to the above, these requirements were not met in that:

1. Established plant procedures were inadequate to assure that deviations from overtime guidelines would be approved, in that controls included in Administrative Procedure (AP) 0010119, Revision 14, "Overtime Limitations for Plant Personnel," lacked specificity in defining when an overtime deviation request was required, resulting in inconsistencies in application.

2. Established plant procedures were inadequate to assure that proper reviews were performed monthly to assure that excessive hours were not assigned, in that controls included in AP 0010119, Revision 14, "Overtime Limitations for Plant Personnel," did not include an appropriate level of specificity in defining how such a review was to be conducted. Sources of information were not specified, sample size was not defined, and what, if any, records of the reviews' results were to be generated were not defined. As a result, management failed to identify that unauthorized deviations from the overtime guidelines contained in Technical Specification 6.2.2.f were occurring.

This is a Severity Level IV violation (Supplement I).

C. Technical Specification Section 6.8.1.c states that written procedures shall be established implemented and maintained covering surveillance and test activities of safety related equipment.

Procedure QI 5-PR/PSL-1, Revision 71, "Preparation, Revision, Review/Approval of Procedures," Section 5.11.1, stated, in part, that for maintenance that can affect the performance of safety related equipment, nuclear plant work orders invoking detailed vendor technical manual step-by-step instructions required Facility Review Group review and Plant General Manager approval.

Procedure QI 5-PR/PSL-1, Revision 71, "Preparation, Revision, Review/Approval of Procedures," Section 5.11.3, stated that changes to technical manuals received from the vendor or changes initiated by FPL shall be forwarded to Production Engineering Group/Juno Beach for review and approval.

Procedure QI 5-PR/PSL-1, Revision 71, "Preparation, Revision, Review/Approval of Procedures," Section 5.11.5, stated that maintenance and preventative maintenance requirements specified in technical manuals shall be considered when writing maintenance procedures and that vendor recommendations for preventative maintenance activities or frequencies contained in these Vendor Technical Manuals may be deviated from, provided a technical review is performed by the respective maintenance engineering group.

Contrary to the above, these requirements were not met in that:

1. Maintenance and tests performed during the Unit 1 1996 refueling outage on the Unit 1 reactor cavity pressure relief dampers invoked the use of the vendor's technical manual but did not require that this manual and work order be reviewed by the Facility Review Group and approved by the Plant General Manager.

2. The frequency for the maintenance and tests specified by the vendor manual for the Unit 1 reactor cavity pressure relief dampers was changed from annually to once every 54 months without a technical review by the respective maintenance engineering group.

3. The torque values specified for the Unit 2 reactor cavity pressure relief dampers were changes to vendor specified criteria and were implemented without a technical review by the respective maintenance engineering group. In addition, these changes were not sent to the Production Engineering Group/Juno Beach for review and approval.

This is a Severity Level IV violation (Supplement I).

D. 10 CFR 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Criterion XI, "Test Control," states, in part, that " . . . a test program shall be established to assure that all testing required to demonstrate that structures, systems and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents . . . Test procedures shall include provisions for assuring that all prerequisites for the given test have been met . . . Test results shall be documented and evaluated to assure that test requirements have been satisfied."

Contrary to the above, adequate procedures and controls were not established for the tests performed on the Unit 1 and Unit 2 reactor cavity pressure relief dampers to assure that these dampers would perform satisfactorily in service and meet the requirements specified by the design documents and the vendor's technical manual, in that:

1. Tests of the Unit 1 reactor cavity pressure relief dampers completed on December 7, 1990, indicated that 4 of the 7 damper blades to damper numbers 1, and 3 of the 7 damper blades to damper number 2, failed to meet the acceptance criteria of the vendor's technical manual. The work order records did not indicate that the dampers were adjusted to meet the acceptance criteria.

2. Tests of the Unit 2 reactor cavity pressure relief dampers completed on October 24, 1990, indicated that 4 of the 7 damper blades for damper number 1, and 5 of the 7 damper blades for damper number 2, did not meet the acceptance criteria of the vendor's technical manual.

3. Tests of the Unit 2 reactor cavity pressure relief dampers completed on May 21, 1992, indicated that all 7 damper blades for damper number 2 did not meet the acceptance criteria of the vendor's technical manual. The work order records did not indicate that the damper was adjusted to meet the acceptance criteria. The test results indicated that damper number 1 was satisfactory.

4. Tests of the Unit 2 reactor cavity pressure relief dampers completed on February 17, 1994, indicated that all 7 damper blades for damper number 1 did not meet the acceptance criteria of the vendor's technical manual. The work order records did not indicate that the damper was adjusted to meet the acceptance criteria. The test results indicated that damper number 2 was not tested.

This is a Severity Level IV violation (Supplement I).

E. Technical Specification 6.8.1.a requires that written procedures be established, implemented, and maintained covering the activities recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February, 1978. Appendix A, paragraph 1.d includes administrative procedures for procedural adherence. QI 5-PR/PSL-1, Revision 71, "Preparation, Revision Review/Approval of Procedures," Section 5.13.1, states that all procedures shall be strictly adhered to.

HP-2, Florida Power and Light (FPL) Health Physics Manual, Revision 10, describes the radiation protection program at FPL's nuclear power plants. The licensee's contamination guidelines are summarized in Table 4.2, "Contamination Guidelines," of the manual. The following contamination limits are described in Table 4.2.

The licensee's contamination limits for materials, tools, equipment and solid waste unconditionally released from the Radiation Control Area (RCA) are:

1,000 dpm/100 cm^2 for loose beta and gamma contamination and

5,000 dpm/100 cm^2 for fixed beta and gamma contamination (direct measurement)

The licensee's contamination limits for tools and equipment used in the RCA are:

1,000 dpm/100 cm^2 for loose beta and gamma contamination and

10 mrem/hr for fixed beta and gamma contamination

Contrary to the above, these requirements were not met in that:

1. On June 18 and 19, 1996, licensee health physics technicians found contaminated tools outside the RCA having contamination levels greater than the unconditional release limits.

   On June 18, 1996, health physics technicians removed 12 M&TE tools from the clean tool room having contamination levels up to approximately 12,500 dpm/100 cm^2 (250 net counts per minute/probe).

   On June 19, 1996, health physics technicians removed five rigging slings from the licensee's clean tool room having contamination levels from approximately 40,000 to 600,000 dpm/100 cm^2 (8,000 to 120,000 dpm/probe).

2. On June 13, 14, and 16, 1996, health physics technicians found tools in the RCA having contamination levels greater than the limits for tools and equipment utilized in the RCA.

   On June 13, 1996, health physics technicians removed nine tools from a temporary hot tool room having loose contamination levels from approximately 1,000 to 20,000 dpm/100 cm^2.

   On June 14, 1996, health physics technicians removed five wrenches from the Unit 1 hot tool room having loose contamination in the range of 1,000 to 4,000 dpm/100 cm^2.

   On June 16, 1996, health physics technicians removed numerous tools from a temporary hot tool room having loose contamination in the range of 1,000 to 30,000 dpm/100 cm^2.

   On June 16, 1996, health physics technicians removed numerous (two bags) of tools from the Unit 1 hot tool room having loose contamination in the range of 1,000 to 120,000 dpm/100 cm^2.

This is a Severity Level IV violation (Supplement IV).

Pursuant to the provisions of 10 CFR 2.201, the Florida Power & Light Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, with a copy to the Regional Administrator, Region II, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or

include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. However, if you find it necessary to include such information, you should clearly indicate the specific information that you desire not to be placed in the PDR, and provide the legal basis to support your request for withholding the information from the public.

Dated at Atlanta, Georgia
this 5 day of August 1996.

U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos: 50-335, 50-389
License Nos: DPR-67, NPF-16

Report No: 50-335/96-09, 50-389/96-09

Licensee: Florida Power & Light Co.

Facility: St. Lucie Nuclear Plant, Units 1 & 2

Location: 9250 West Flagler Street
Miami, FL 33102

Dates: June 9 - July 6, 1996

Inspectors: M. Miller, Senior Resident Inspector
W. Miller, Resident Inspector (Acting)
M. Miller, Reactor Inspector, paragraph E1.1
T. Johnson, Senior Resident Inspector, Turkey Point, paragraph 01.4
F. Wright, Reactor Inspector, paragraphs R1 and R5
J. York, Reactor Inspector, paragraph E1.2

Approved by: K. Landis, Chief, Branch 3
Division of Reactor Projects

ENCLOSURE 2

~~9608230215~~ 38PP

EXECUTIVE SUMMARY

St. Lucie Nuclear Plant, Units 1 & 2
NRC Inspection Report 50-335/96-09, 50-389/96-09

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 4-week period of resident inspection.

Operations

- Observations of two instances of reduced inventory operations indicated appropriate controls, and excellent attention to the tasks at hand (paragraphs O1.2 and O1.4).

- Operator action in response to a condition of inoperability of three channels of wide range nuclear instrumentation was prompt and appropriate (paragraph O1.3).

- The licensee's control of the use of overtime was found to be poor, with multiple examples of unapproved deviations of Technical Specification guidelines. Two violations resulted from the identified failures to properly manage employee overtime (paragraph O8.1)

Maintenance

- Maintenance activities associated with the installation of the Unit 1 reactor vessel head and Unit 2 feedwater control system were satisfactory (paragraph M1.1).

- Maintenance activities associated with the testing of both units' reactor pressure relief dampers indicated that testing and maintenance was not managed in accordance with plant procedures and that records were not maintained in accordance with the licensee's Quality Assurance Plan. Two violations resulted from activities in these areas (paragraph M1.1).

- A review of licensee-identified issues surrounding Unit 1 reactor auxiliary building floor drain isolation valve maintenance, and a failure to perform required post-maintenance testing resulted in a non-cited violation (paragraph M8.1).

Engineering

- The I&C group, with engineering support, was satisfactorily implementing the Unit 1 nuclear instrumentation modification after overcoming initial implementation and coordination problems due to the outage work load (paragraph E1.1).

- The inspectors reviewed the functions of the current engineering organization and the anticipated changes after downsizing takes place on August 1, 1996 (paragraph E1.2).

Plant Support

- The radiation protection program was adequately managed and internal and external exposure control programs were effectively implemented with all radiation exposures within 10 CFR Part 20 limits. (paragraph R1.3)
- A violation was identified concerning failure to follow procedures for the control of contaminated tools. (paragraph R1.5)
- Tours of licensee facilities showed generally good radiological housekeeping and controls. (paragraph R1.5)
- A review of the licensee's Reactor Coolant Pump Oil Collection System indicated that the system was in accordance with UFSAR commitments (paragraph F2.1)

Report Details

Summary of Plant Status

Unit 1 entered the inspection period in Mode 6 for a refueling outage. The unit entered Mode 5 on June 14 and remained in Mode 5 for the balance of the inspection period.

Unit 2 entered the inspection period in Mode 5 to support modifications to the unit's C auxiliary feedwater pump steam supply lines. The unit was taken critical on June 13 and was placed on line on June 14. The unit operated at essentially full power until June 22, when power was reduced to approximately 20 per cent to support troubleshooting and repair of the 2B main feedwater regulating valve controller. The unit was returned to full power on June 23 and operated at essentially full power for the balance of the inspection period.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. In general, the conduct of operations was professional and safety-conscious; specific events and noteworthy observations are detailed in the sections below.

O1.2 Reduced Inventory Operations (71707)

a. Scope

On June 19, at 5:15 a.m. Unit 1 entered reduced inventory to replace the pump seals for RCP 1A2. The RCS level was restored to normal level at 4:48 p.m. on June 20. While the RCS was at a reduced inventory, a number of controls and procedures were implemented to ensure the safety of the unit. Two procedures implemented were: AP 0010145, Rev 10, "Shutdown Cooling Controls" and AP 1-0410022, Rev 27, "Shutdown Cooling", Appendix A, "Instructions for Operations at Reduced Inventory or Mid-loop Conditions."

b. Findings

The inspector reviewed the following items during this evolution:

- Containment Closure Capability - Instructions were issued to accomplish containment closure. The equipment and personnel hatches were closed prior to reduction in RCS inventory. Four electrical penetrations remained open to support outage work on going within the reactor building while the RCS was in reduced inventory. The inspector reviewed the penetrations which remained open and verified that closure capability was available.

- RCS Temperature Indication - Two CETs were available on each SPDS

channel.

- RCS Level Indication - Independent RCS wide and narrow range level instruments, which indicated in the control room, were operable. An additional Tygon tube loop level indicator was installed in the containment and was visible to a dedicated operator in the control room via a television monitor.

- RCS Level Perturbations - When RCS level reduction was initiated, additional operational controls were invoked. Plant staff were advised of the reduction in RCS level and Operations took action to ensure that maintenance would not perform work that might effect RCS level or shut down cooling.

- RCS Inventory Volume Addition Capability - One HPSI pump and one charging pump were available for inventory addition, as were two trains of shutdown cooling.

- RCS Nozzle Dams - The RCS nozzle dams previously installed in each of the two steam generators were inspected for integrity prior to the reduction in RCS inventory and were reinspected for integrity every four hours.

- Vital Electrical Bus Availability - Operations did not release busses or alternate power sources for work while the unit was in a reduced inventory.

- Pressurizer Vent Path - The manway atop the pressurizer was removed to provide a vent path and Operations verified that the manway was unobstructed every four hours.

c. Conclusions

Operations exerted appropriate controls while the RCS was in reduced inventory.

O1.3 <u>Unit 1 Loss of Nuclear Instrumentation</u>

a. Scope

During the current Unit 1 refueling outage, the nuclear instrumentation system was replaced with a new system. On June 20 at 3:25 a.m., with Unit 1 in Mode 5, only one (Channel A) of the four wide range logarithmic neutron flux monitor instrumentation channels was operable. The remaining three channels were inoperable. The inspector followed the licensee's actions with regard to these failures.

b. Findings

TS Section 3.3.1.1 requires two nuclear instrumentation channels to be in service when the unit is at Modes 3, 4 or 5. With only one channel operable, the shutdown margin requirement of TS Section 3.1.1.2 is

required to be verified within one hour and at least once per 12 hours thereafter.

The inspector reviewed the licensee's response actions after the three channels became inoperable. Procedures OP-1-0010125A, Rev 3, "Unscheduled Surveillance Tracking Data Sheet 30" and OP-1-0110056, Rev 22, "Surveillance Requirements for Shutdown Margin Modes 2, 3, 4, and 5," were implemented. The first shutdown margin calculation was performed at 3:30 a.m. on June 20 and additional calculations were performed each 12 hours thereafter until a second channel was restored to service on June 26 at 4:42 p.m.

The inspector reviewed the calculation data sheets and found the calculations to be in compliance with the engineering and chemistry reference data in the control room.

c. Conclusion

The action by Operations when three of the four nuclear instrumentation channels became inoperable was prompt and appropriate.

O1.4 Reactor Coolant System (RCS) Mod-Loop Operations (71707)

a. Scope

On July 5, at 6:00 p.m. Unit 1 entered RCS Mid-Loop to remove the SG nozzle dams. The RCS level was restored to normal level at 4:45 a.m., July 7. While the RCS was at a reduced inventory in mid-loop (29'8" or 23" above hot leg bottom), a number of controls and procedures were implemented to ensure the safety of the unit. Procedures implemented were: AP 0010145, Rev 10, "Shutdown Cooling Controls", AP 1-0410022, Rev 27, "Shutdown Cooling", Appendix A, "Instructions for Operations at Reduced Inventory or Mid-loop Conditions," and OP 1-0120021, Rev 44, "Draining the RCS."

b. Findings

The inspector reviewed the following items during this shutdown safety significant evolution:

- NRC Generic Letter 88-17, Loss of Decay Heat Removal, and FPL responses.
- Containment Closure Capability - Instructions were issued to accomplish containment closure. The equipment and personnel hatches remained opened prior to reduction in RCS inventory. A crew was on station ready to effect closure in 30 minutes. One electrical and one mechanical penetration remained open to support outage work on going within the reactor building while the RCS was at mid-loop. The inspector reviewed the penetrations which remained open and verified that closure capability was available.

- RCS Temperature Indication - A number of CETs were availabl on each SPDS channel.

- RCS Level Indication - Independent RCS wide and narrow range level instruments, which indicated in the control room, were operable. An additional Tygon tube loop level indicator was installed with a dedicated operator in the containment and was visible to another dedicated operator in the control room via a television monitor. Alarm functions and ERDADS monitoring were also available.

- RCS Level Perturbations - When RCS level reduction was initiated, additional operational controls were invoked. Plant staff were advised of the reduction in RCS level and operations took action to ensure that maintenance would not perform work that might effect RCS level or shut down cooling. This included the use of signs in the plant and in the control room.

- RCS Inventory Volume Addition Capability - One HPSI pump and one charging pump were available for inventory addition, as were two trains of shutdown cooling using the LPSI pumps.

- RCS Vent Paths - Two vent paths were available (pressurizer manway removal and reactor head vent).

- Vital Electrical Bus Availability - Operations did not release busses or alternate power sources for work while the unit was in mid-loop. Both EDGs and all three offsite power sources were available. No switchyard work was allowed.

- Core Cooling - Both SDC trains were maintained in service, including availability of LPSI pumps, SDC heat exchangers, and ICW/CCW pumps.

- Training - All crews were briefed on the drain-down and mid-loop evolutions. In addition, selected crews were given simulator training including a loss of SDC and RCS inventory. Implementation of off-normal operating procedures was reviewed.

- Management Oversight - A dedicated SRO (in addition to the NPS, ANPS, and NWE) was assigned to provide "Mid-Loop" oversight over and above the normal crew complement. Further, Operations and plant management provided coverage/monitoring of the mid-loop activities. In addition, a Management-On-Shift representative was also providing independent oversight. Shutdown safety was also addressed by the STA at the outage meetings. Due to mid-loop, core cooling, core inventory, and containment safety functions were "yellow." This information was stressed by management during meetings.

c. Conclusions

The inspector concluded the licensee's RCS mid-loop controls were effective in assuring shutdown safety. Licensee attention, including management oversight was excellent.

O6 Operations Organization and Administration

O6.1 Site Reorganization (40500)

On June 27, the licensee announced the following planned reorganization:

- Plant System Engineering, formerly assigned under the Plant General Manager, will be moved under the Site Engineering Manager. J. West will assume the title of System Engineering Manager.

- The Operations Support and Testing organization, formerly under the Operations Manager, will be moved under the Site Engineering Manager. P. Fulford will remain as the OST Supervisor.

- Responsibility for the implementation of the maintenance rule will be moved under the Site Engineering Manager.

- The Work Control and Outage Management functions are to be combined under the Plant General Manager. C. Wood will assume the position of acting Work Control Manager.

O8 Miscellaneous Operations Issues

O8.1 Control of Overtime (71707, 40500)

a. Scope

The inspector reviewed the licensee's control of overtime for the period of May 13 through June 13. The inspector obtained gate logs for 26 individuals. The selected individuals were chosen from the licensee's maintenance, engineering, planning, and management organizations based upon their involvement in outage activities and the inspector's understanding of the activities under their cognizance. From the results obtained (which demonstrated time spent on site), the inspector reduced the inspection population to five individuals based upon indications of excessive hours. The individuals in question included supervisors and engineers with responsibilities for safety-related work.

As acceptance criteria, the inspector reviewed TS 6.2.f, which required that the hours expended by personnel performing safety-related functions be limited, with an objective that personnel work a normal 8 hour day, 40 hour week while the plant was operating. The TS observed that

substantial amounts of overtime might be required during extended periods of shutdown for refueling, and established guidelines for these periods. The TS stated"

". . . on a temporary basis the following guidelines shall be followed:

a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

b. An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time.

c. A break of at least 8 hours should be allowed between work periods, including shift turnover time . . .

. . . Any deviations from the above guidelines shall be authorized by the Plant General Manager or his deputy, or higher levels of management, in accordance with established plant procedures and with documentation of the basis for the deviation." The inspector reviewed AP 0010119, Rev 14, "Overtime Limitations for Plant Personnel," and found that the procedure appropriately implemented the TS requirements.

b. Findings

The inspector found that the licensee deviated from TS guidelines for the control of overtime without the prior (or subsequent) approval from senior plant management. Of the five individuals focused on as a result of gate logs, the following information was obtained from timesheets (violations of the requirements were cited only for excesses of requirements which had not received approval per AP 0010119):

Individual	Violations of 72 Hour Requirement	Violations of 24/48 Hour Requirement	Violations of 16 Hour Requirement
A	3	0	0
B	0	0	0
C	5	1	0
D	14	2	0
E	16	12	3
Total	38	15	3

The instances identified above, in which TS guidelines were exceeded, and for which the TS-required approvals for the deviations were not obtained, collectively represent a violation (VIO 335,389/96-09-01, "Failure to Control Overtime").

While violations were identified, the inspector also noted that significant differences existed between timesheet records, which divided time between TS and non-TS (e.g. shift turnovers) categories, and gate records, which indicated total time on site. For the 5 individuals highlighted above, numerous instances of differences between total time on site and timesheet-indicated time on site existed, with differences frequently exceeding one and two hours and, at times, exceeding several hours. The most time spent continuously on site was noted to be approximately 26 hours.

The inspector discussed the results above with the affected parties to ascertain the reasons for the excessive use of overtime and for the differences between gate logs and timesheets. Responses were mixed. Regarding the heavy use of overtime, several respondents pointed out that the project that they had been working was adversely affected by the loss of several key personnel which reduced the depth of knowledge on the associated job. Several stated that the diverse activities on both units (due to the outage on Unit 1 and the recent trip of Unit 2) had placed increased demands on their time.

In discussing the method for completing timesheets, the inspector found that a lack of uniformity existed. Some respondents treated work periods (as described on the timesheet) as any work performed on a given calendar day. By applying this approach, the potential existed for the work hours recorded for a given day to represent a composite value of two work periods if one (or more) of the work periods extended across midnight. The potential result of this type of accounting was that the true length of a work period, as referenced in TS, would not be accurately reflected on timesheets, confounding the ability to maintain an accurate count of daily, 48 hour and 7-day totals.

With regard to not obtaining the appropriate deviation approvals for time worked in excess of the guidelines, several workers stated that they believed that obtaining a deviation provided a blanket authorization for overtime spent on the project for which the deviation applied. The inspector noted that the AP was not specific as to whether a deviation request was required for each planned deviation from the guidelines or whether it applied to the job which was described on the request. The inspector discussed this issue with the Plant General Manager, who stated that it was his expectation that a deviation request be filed for each planned deviation of the guidelines (the implication being that a series of work periods for which each period led to violations of one or more guidelines should each be documented on separate requests). The inspector had requested any deviation requests

associated with the personnel audited for the subject time period. Two were identified which addressed themselves to 3 of the personnel. The deviations covered by these deviation requests were not considered in the summary table above.

AP 0010119 required that department heads perform a monthly review of assigned overtime to assure that excessive overtime was not assigned. However, step 8.5 of the procedure, which directed that department heads perform a monthly review to ensure that excessive overtime hours were not assigned, was not specific as to how such a review should be performed (e.g. population size, sources of information). The inspector noted that this was the second weakness identified in the procedure (the first being a lack of specificity on when deviation requests were required).

Technical Specification 6.2.f requires, in part, that deviations from overtime guidelines be approved in accordance with established procedures and that controls be included in established procedures such that individual overtime be reviewed monthly by the Plant General Manager or his designee to assure that excessive hours have not been assigned. The inspector concluded that the failures of the subject procedure to provide an appropriate level of detail resulted in a procedure which was inadequate to satisfy the requirements of the TS. Consequently, the procedure inadequacies constitute a violation (VIO 335,389/96-09-02, "Inadequate Procedure for Managing Overtime").

Independent of this inspection (and unknown by the inspector at the time), the licensee's QA organization performed an audit of overtime usage for the period from May 5 through 18. A population of 100 plant personnel was selected at random for the audit. QA reviewed gate logs for the sample population and applied criteria which assumed a one half hour lunch break and accepted turnover periods to reach the following criteria for determining whether guidelines had been exceeded:

- No more than 17.5 hours in 1 day.
- No more than 27 hours in a 48 hour period
- No more than 82.5 hours in a 7 day period
- An 8 hour break between work periods.

QA determined that 13 percent of their population exceeded the criteria at least once and that 8 percent exceeded the criteria at least twice. QA informed management of their findings in this area on June 6. As a result, the Site Vice President and the PGM discussed the problem with plant staff at morning meetings to stress expectations for personal accountability in this area. On June 19, the PGM issued a letter to department heads restating the overtime guidelines and stressing personal accountability on the issue. The inspector noted that, with respect to immediate corrective actions, 23 examples of unapproved deviations existed in the inspector's sample from June 8 through 13.

c. Conclusions

As a result of this inspection, the inspector concluded the following:

- Overtime usage for the period May 13 through June 13 exceeded TS guidelines for a number of personnel.
- The licensee failed to effectively control overtime as required in AP 0010119, Rev 14, "Overtime Limitations for Plant Personnel," in that deviation requests were neither prepared nor approved for the majority of deviations identified.
- AP 0010119 was unclear in its expectations, both for when a deviation request was required and for how reviews of overtime usage were to be executed.
- The requirement for monthly reviews of overtime usage, detailed in AP 0010119, was ineffectively implemented.
- Personnel have, at times, worked hours which were not recorded on timesheets.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance Observations (62703)

a. Inspection Scope

The inspector reviewed the following maintenance activities performed during the inspection period:

- GMP 1-M-0015, Rev 30, "Reactor Vessel Maintenance - Sequence of Operations"

Reactor Cavity Pressure Relief Dampers - Units 1 and 2

- WO 9502884301, Lubrication and Testing of the Unit 1 Reactor Cavity Pressure Relief Dampers (1996 Refueling Outage)
- WO 9102918301, Lubrication and Testing of the Unit 1 Reactor Cavity Pressure Relief Dampers (1991 Refueling Outage)
- WO XA870271657, Lubrication and Testing of the Unit 1 Reactor Cavity Pressure Relief Dampers (1987 Refueling Outage)
- WO 9500282701, Lubrication and Testing of the Unit 2 Reactor Cavity Pressure Relief Dampers (1995 Refueling Outage)
- WO 9300990701, Lubrication and Testing of the Unit 2 Reactor

Cavity Pressure Relief Dampers (1994 Refueling Outage)

- WO 9200226001, Lubrication and Testing of the Unit 2 Reactor Cavity Pressure Relief Dampers (1992 Refueling Outage)
- WO XA900621191827, Lubrication and Testing of the Unit 2 Reactor Cavity Pressure Relief Dampers (1990 Refueling Outage)

Repairs to Main Feedwater Regulator for Steam Generator 2B

- WO 9601578201, Spurious Operation of Main Feedwater Regulator Valve for Steam Generator 2B with Main Feedwater Regulator Controller in Automatic
- WO 9601581601, Improper Operation of FCV-9021 With Controller for Main Feed Regulator Valve for Steam Generator 2B in Automatic
- WO 9601632301, Replace Steam Generator 2B Downcomer Level Controller
- WO 9601645501, Install Steam Generator 2B Steam/Feedwater Flow Controller

b. Observations and Findings

Unit 1 Reactor Head Installation

The inspector witnessed the performance of Procedure GMP 1-M-0015 for the installation of the reactor head. A thorough pre-job briefing was performed for personnel assigned to perform the task. The inspector observed lifting the reactor vessel head by the reactor building polar crane from the reactor building refueling floor and placing the head on cribbing place on the reactor flange. After placing the head on the cribbing, the reactor vessel head O-ring and seating surface were inspected, cleaned and honed as necessary to meet the specification requirements of the procedure. The head installation personnel followed the procedure and the inspector noted no discrepancies.

Reactor Pressure Relief Dampers

The reactor pressure relief dampers were described by Unit 1 UFSAR Section 6.2.1.3.3 and by Unit 2 UFSAR Section 6.2.1.2.3. These dampers are closed during normal plant operations to aid in maintaining proper reactor cavity ventilation. During a LOCA, these dampers were designed to open and provide pressure relief through two electrical tunnels to maintain the pressure rise in the reactor cavity below the design safety limits.

The inspector reviewed the previous maintenance and tests performed on these pressure relief dampers. The vendor recommended that maintenance and testing of the Unit 1 dampers be performed annually and once each 18 months for the Unit 2 dampers. The maintenance and testing of the Unit

1 dampers were on a 54 month frequency and the Unit 2 dampers were on an 18 month frequency. The licensee did not have an evaluation to justify deviating from the vendor's recommended maintenance and test frequency for the Unit 1 dampers. Procedure QI 5-PR/PSL-1, Rev 71, "Preparation, Revision, Review/Approval of Procedures," Section 5.11, stated that maintenance and preventative maintenance requirements specified in vendor technical manuals shall be considered when writing maintenance procedures. Per the QI, deviations from these recommendations required justification by a technical review performed by the respective maintenance engineering group. The failure to provide a technical review justifying changes to the maintenance and testing of the Unit 1 pressure relief dampers from annually to once per 54 months is identified as an example of Violation 50-335,389/96-09-03, "Failure to Test the Reactor Cavity Pressure Relief Dampers In Accordance With The Vendor's Technical Manual." Additionally, the inspector noted that the subject QI was not up to date, in that it referenced a maintenance engineering group which has not existed on site (due to reorganization) for some time.

The work orders for the maintenance and testing of these dampers referenced the vendor technical manuals as the document providing the means of testing the dampers and the minimum acceptance requirements. The vendor manual for the Unit 2 dampers identified the torque value set points for each pressure relief damper blade. During a review of the vendor's manual stored in the maintenance planning library, the inspector noted an internal memorandum in the manual for the Unit 2 dampers which contained alternate acceptance criteria for the torque values required for the damper blades. This engineering memorandum changed the vendor's recommended set points of the damper blades from plus or minus 5 lbs. to plus or minus 30 lbs. for blades Nos. 1 through 6 and plus or minus 26.5 lbs. for blade No. 7. There were no records to indicate that this change to the vendor document had received the required safety review prior to implementation. These revised values had apparently been in use for Unit 2 since November 13, 1987. The vendor manual in the document control vault did not contain this change. Therefore, the inspectors concluded that these changes had not been reviewed and approved as required by Procedure QI 5-PR/PSL-1, Section 5.11.5. This procedures stated that preventative maintenance requirements specified in technical manuals shall be considered when writing maintenance procedures and that the vendor recommendations for preventative maintenance activities or frequencies contained in the vendor technical manuals may be deviated from, provided a technical review was performed by the respective maintenance engineering group. The use of technical reference data which had not received a technical review and approval is identified as another example of Violation 50-335,389/96-09-03, "Failure to Test the Reactor Cavity Pressure Relief Dampers In Accordance With The Vendor's Technical Manual."

The records for the maintenance and testing performed by the Wos identified the torque values measured for each damper during the maintenance and test activities, except for WO 9502884301. This WO was for the Unit 1 dampers and did not list the measured torque values. The

maintenance craft records only indicated that the dampers met the vendor's requirements. Discussions with mechanical maintenance personnel indicated that this information was obtained but the data was left in the Unit 1 reactor building. The licensee obtained a statement from the maintenance craft which indicated that the work was accomplished and that acceptance criteria were met.

The inspector noted that WO 9502884301 used the vendor's technical manual as a reference document which did not require review by the Facility Review Group (FRG) and approval by the Plant General Manager. However, the work order did not contain sufficient detail instructions to perform the required maintenance, lubrication and testing required for the dampers. For example, the work order did not provide a description of the test procedure and did not provide the acceptable test values, (e.g. torque required on each damper blade). The work order stated that the referenced vendor manual was not required to be present at the job site.

The inspector concluded that the vendor manuals were, in fact, required to be at the job site in order to perform the required maintenance and test on these dampers. The work order did not provide sufficiently detailed information to perform the maintenance or to conduct the required test activity. Procedure QI 5-PR/PSL-1, Section 5.11.2, stipulated that vendor manuals could be used to supplement an invoked plant approved procedure/guideline or work scope/instructions without FRG review and Plant General Manager approval. However, to meet this requirement, the work order must have provided sufficient information to accomplish the task. For this work order, the information in the vendor's technical manual was required to perform the maintenance and test. This item is identified as another example of Violation 50-335, 389/96-09-03, "Failure to Test the Reactor Cavity Pressure Relief Dampers In Accordance With The Vendor's Technical Manual."

Maintenance inspection, testing and lubrication data for the Unit 1 reactor cavity pressure relief dampers from 1987 and from 1990 for Unit 2 were evaluated. The inspector noted a number of significant deficiencies. These included:

- Unit 1 dampers worked by WO 91102918301, completed December 7, 1991, did not provide the proper torque values for the dampers. The work order indicated that the torque adjustments for 4 of the 7 damper blades to reactor cavity damper 1 were left out of tolerance and the torque values for 3 of the 7 damper blades to reactor cavity damper 2 were left out of tolerance.

- Unit 2 dampers worked by WO XA900621191827, completed October 24, 1990, did not provide the proper torque values for dampers. The torque adjustments for 4 of 7 of the damper blades for damper 1 were out of tolerance and the tolerance for 5 of the 7 damper blades for damper 2 were found out of tolerance using the vendor technical information. However, it appears that unapproved alternate acceptance criteria may have been used. As discussed

above, the use of this unapproved data is identified as a violation.

- Unit 2 dampers worked by WO 9200226001, completed May 21, 1992, did not provide the proper torque values for reactor cavity damper 2. The torque adjustments for all 7 of the damper blades were found out of tolerance. The work data did not clearly indicate that these dampers were properly readjusted. No retests results were provided in the work package. Damper 1 was found satisfactory, adjusted and retested.

- Unit 2 dampers worked by WO 9300990701 only tested one of the two dampers. Damper 2 was not tested. The test for damper 1 did not use the acceptance criteria of the vendor technical manual but used alternate values. This issue is discussed above and is identified as a violation. The licensee reviewed this issue and confirmed that only one of the two dampers had been tested.

Based on these findings, the licensee issued Condition Report 96-1524 to address these items. These discrepancies are considered a failure to conform to the test requirements of 10 CFR 50 Appendix B Criterion XI, Test Control, and are identified as Violation 50-335,389/96-09-04, "Failure To Perform Adequate Testing On the Reactor Cavity Pressure Relief Dampers."

As regards safety significance, the licensee presented information that indicated that the blast dampers' classification as safety-related was conservative. An NRC Safety Evaluation Report, dated March 5, 1993, concluded that dynamic effects associated with postulated pipe breaks in the RCS could be excluded from the design and licensing bases for St. Lucie. That determination was based upon the licensee demonstrating that they were bounded by the Combustion Engineering Owner's Group's leak-before-break analysis and that leakage detection systems for the RCS were capable of identifying sufficiently low leak rates (allowing time for operator action before leaks manifested themselves as larger breaks). Consequently, the blast dampers, which had been designed to accommodate dynamic effects, were not necessary in a safety-related context.

Notwithstanding the low safety significance (given the results of the SER) of the blast damper deficiencies identified by the inspectors, the fact that the licensee had treated them as safety-related indicated a poor application of QA plan requirements to the maintenance and testing of the components. Consequently, the violations identified above remain cited, as they indicate a programmatic weakness with regard to the testing of (designated) safety-related components.

Repairs to Main Feedwater Regulator Valve and Controller for Steam Generator 2B

On June 13, while returning Unit 2 on line following the manual trip on June 6, operation of main feedwater regulator valve FCV-9021 to Steam Generator 2B was erratic and spurious when the main feedwater regulator valve controller was placed in the automatic mode. Therefore, Operations placed the controller for this valve in manual and initiated work orders to investigate the problem and perform the appropriate repairs. The main feedwater controller for Steam Generator 2B was maintained in manual and the Unit was maintained at approximately 98 percent power until June 23 when these investigations and repairs were completed. During this time, the inspector monitored the testing and repairs performed on the main steam/feedwater controller and flow control valve FCV-9021 to Steam Generator 2B. The completed work orders for these tasks were also reviewed.

The licensee's investigation found that main feedwater regulating valve controller FIC 9021-1 for Steam Generator 2B would not control properly. This controller was replaced with a controller from Unit 1. Several problems were identified and corrected, including the replacement of a broken lens for the auto transfer push button, correction of wiring label discrepancies and replacing the positioner for valve FCV 9021 with a new positioner. Post maintenance testing was satisfactorily performed on the replacement controller and other components which were replaced or worked.

The inspector did not identify any discrepancies during observation of the work and testing activities or in the completed work order packages.

c. Conclusion

The inspectors concluded the following with respect to maintenance activities observed during the period:

- Maintenance activities were satisfactorily performed with respect to reactor head installation and the Unit 2 feedwater regulating system

- Significant problems were identified with the maintenance and testing associated with the reactor cavity ventilation dampers. Two violations were identified for these items, involving the adequacy of testing (periodicity justification, acceptance criteria, and the quality of work instructions) and the recording of as-found and as-left data.

M3 Maintenance Procedures and Documentation

M3.1 Plant/Transmission and Distribution Interface (62703)

a. Scope

The inspector reviewed the licensee's procedures for the establishment and control of the interface between Operations and Transmission and Distribution personnel. The inspector reviewed the following procedures:

- AP 0010532, revision 6, "Relay Work Orders"
- Transmission and Distribution (T&D) Procedure 2650, "Interconnected System - Division of Responsibility"

b. Findings

The inspector found that the subject procedures established clear lines of demarkation between T&D and the onsite organization. With respect to AP 0010532:

- Appropriate controls were established for relay work performed by Protection and Control personnel through the Relay Work Order process.

- The procedure required that the ANPS for an affected unit be notified and that permission be received prior to the commencement of work.

- Controls over the use of vendor technical manuals were consistent with site practice.

With respect to T&D procedure 2650:

- Lines of responsibility and authority were described

- Specific instructions were described for work involving nuclear plants, with requirements for communication with the NPS/ANPS specified.

- Additional instructions were included to accommodate reduced-inventory and mid-loop conditions.

The inspector discussed these controls with Operations personnel and found that, in general, the interface between Operations and T&D had been controlled appropriately. One example of a failure to communicate across organizational boundaries, involving relay work performed on a tagged-out component, was identified. The licensee stated that the issue was resolved through discussions and counseling.

c. Conclusion

The inspector concluded that the licensee had appropriately established controls between activities performed by site personnel and those conducted by T&D.

M8 Miscellaneous Maintenance Issues

M8.1 Violation 335/95-15-03 Issues Revisited (62703)

a. Scope

IR 96-04 closed violation 335/95-15-03, which was issued in response to operator failures to log valve position deviations properly. The issue involved floor drain valves HCV-25-1/7, which were closed and not logged. The valves' closure complicated a loss of RCS inventory event on August 10, 1995, when water issuing from an open relief valve collected in the Unit 1 pipe tunnel, rather than draining through the floor drain system (see IR 95-20).

In closing the subject violation, the inspector noted that the valves had been left closed after difficulties encountered while stroking them in preparation for Hurricane Erin. The inspector reviewed the work packages generated in the repair of the valves and reported that all work appeared to have been performed properly and that appropriate post-maintenance testing had been performed.

During the current inspection period, the inspector reviewed CR 96-1183, which was issued by the licensee's QA organization to document the fact that the subject valves may not have been properly tested after rework. The issue involved the fact that the valves' actuators had been removed in support of correcting the sticking conditions described above. QA indicated, in the CR, that proper post maintenance testing may not have been performed on the part of I&C personnel (the inspector's reviews, referred to above, were of Mechanical Maintenance activities). Specifically, PWO 63/3836 required that HCV-25-1 be retested using Appendix C-3 of QI 11-4 (which required a calibration of the valve), but no such record existed in the archived work package.

The CR referred to PWO 63/3836, which reported that valve HCV-25-1 was sticking in the open position. The inspector reviewed the subject PWO, as well as PWOs 63/2537 (which addressed repairing sticking on HCV-25-4) and 63/4171 (which directed the replacement of the solenoid valve on HCV-25-4). The inspector found that the CR had correctly identified the fact that the retest specified in step 4 of the subject PWO (valve calibration) had not been performed. Rather, records of a functional test (which, basically, cycled the valve to verify proper operation) were included in the package. As corrective action, the licensee specified that the calibrations of HCV-25-1/7 were to be completed satisfactorily (PWO 96011562 and PMAI 96-070112). The inspector noted that AP 0010432, revision 84 (in affect at the time of the work),

"Nuclear Plant Work Orders," step 8.7.7, included a requirement that stated "[i]f the NPWO instructions need to be changed, use the scope change process." The inspector found that the licensee's failure to obtain a scope change which redefined the required post-maintenance testing requirement for HCV-25-1 (prior to performing a different post-maintenance test) was a violation of the licensee's procedure and was, thus, a violation of NRC requirements for procedural adherence. This licensee identified and corrected violation is being treated as a Non-Cited Violation consistent with Section VII.B.1 of the NRC Enforcement Policy (NCV 335/96-09-05, "Failure to Perform Proper Post-Maintenance Testing of HCV-25-1").

The inspector's review of PWOs 63/2537 and 63/4171 indicated that an inconsistent approach to post-maintenance testing had been applied to HCV-25-01 and HCV-25-04 (similar valves for which similar work had been performed). In the case of HCV-25-01, post-maintenance testing was to have included a calibration of the subject valve (although, as noted above, only a functional test was performed). In the case of HCV-25-04, the PWO simply stated that retesting was to be performed in accordance with QI 11-4 (the licensee's procedure for post-maintenance testing on I&C equipment). The valve received a functional test, as opposed to a calibration.

QI 11-PR/PSL-4, revision 28 (the revision in affect at the time of the work), "Instrumentation and Control Test Control," specified that post-maintenance testing was to be performed as specified in Appendix B, "Required Testing Matrix Lists," of the procedure. The inspector discussed the matrix with the licensee and was told that the reason for the inconsistent treatment of the two valves was that, while providing post-maintenance test instructions for valves which had an actuator repaired or replaced, the matrix included no instructions for actuators which were simply removed in support of other maintenance. As a result of the subject CR, a PMAI (96-07-113) was generated to require calibrations for cases in which a valve actuator was uncoupled from a valve for any reason. The inspector found the licensee's actions to be appropriate to the circumstances.

c. Conclusions

The inspector concluded the following with respect to the issues raised in CR 96-1183:

- I&C personnel failed to comply with procedural requirements when post-maintenance testing was performed on HCV-25-1 on September 9, 1995.

- The licensee took appropriate corrective actions for a weakness identified in the Required Testing Matrix Lists of QI 11-4.

III. Engineering

E1 Conduct of Engineering

E1.1 Nuclear Instrumentation Modification

a. Inspection Scope (37550, 37751)

The inspector reviewed the concerns listed in six Condition Reports (CRs) related to the implementation of plant change/modification PC/M 009-195 for the Nuclear Instrumentation System.

Background

One specific report, CR 96-1358, "Possible Loss of Design Control", for the NI System, dated June 12, 1996, indicated there were significant problems. CR 96-1358 stated there have been problems encountered during implementation which have required design changes, numerous deviations from the approved test procedure, and key individuals were no longer on the project. In addition, the work package had become voluminous and unwieldy with trouble shooting activities. There were 13 work package scope changes and approximately 40 deviations to the test procedure. Trouble shooting was further complicated by not having vendor engineering support on-site. CR 96-1358 is discussed below in the section Conditions Reports.

The inspector's work scope included an examination of all aspects of the NI System modification including a review of the design package, design changes, deviations, testing procedures, test procedure changes, logs, drawings, work orders, equipment specifications, memoranda, design procedures, administrative procedures, UFSAR Chapter 7.2.1, and Technical Specification 3.3.1.1, and 3.9.2. The inspector initiated walkdowns and observed ongoing work. In addition, discussions were held with vendor engineering personnel concerning the ongoing problems. The items in the work scope were performed to verify that the modification was being performed within the requirements of the licensee's program and NRC requirements.

b. Observations and Findings

Plant Change Modification (PC/M)009 195

Nuclear Engineering completed the design as REA/Project # SLN-94-025-11, "RPS NI Drawer Replacement", dated February 27, 1996. It was released as plant change/modification PC/M 009-195, Rev 0, "Replacement Of The Neutron Flux Monitoring And Protective System (NI Drawers) For The RPS System."

The PC/M's purpose was to upgrade the Unit 1 NI System with similar instrumentation that was installed in Unit 2 NI System during the last Unit 2 outage. The design change included the following modification for the four NI channels A, B, C, and D:

1) Replace the four existing Gamma-Metric wide range excore detectors and cables with improved dual fission chamber assemblies.

2) Replace the four existing amplifiers located in containment with new assemblies.

3) Replace the eight existing RPS NI drawers located in the Control Room with the new NI instrumentation.

PC/M 009-195 was being implemented under work order WO No. 96007751. The work order package consisted of PC\M 09-195 and the Pre-Operation (Pre-Op) Test Procedure Number 1-1400280, "Functional Testing of PC/M 009-195 Safety Channel." The inspector verified that the WO was being implemented under administrative procedure ADM-0010432, Rev 3, "Control of Plant Work Orders". In addition, PC/M 009-195 met the requirements in the following Quality Instructions (procedures) for Nuclear Engineering and the equipment specification:

1) ENG-QI 1.0, Rev 3, "Design Control"

2) ENG-QI 1.1, Rev 0, "Engineering Packages"

3) ENG-QI 1.2, Rev 1, "Minor Engineering Package"

4) ENG-QI 1.3, Rev 1, "Drawing Change Requests"

5) SPEC-IC-004, Rev 0, "Equipment Specification For The RPS Nuclear Instrumentation System"

The inspector reviewed and verified that the factory acceptance test procedure, GAMMA-METRICS Test Procedure RMSP Assy No. 201663, agreed with Pre-Op 1-1400280 Test Procedure. Drawings JPN-009-195-001 to 017, "Out-of-Core Neutron Detectors" and "Nuclear Instrumentation & Reactor Protection System, were reviewed and verified that the modification was being implemented according to design drawings.

<u>Scoping Changes</u>

Thirteen scoping changes were reviewed to determine if the changes were controlled and within the requirements of the modification program and procedures. The scoping changes [Change Request Notice (CRN)] reviewed are listed below:

1) CRN-6100 - Enhanced installation instructions and details for using existing cables

2) CRN-6155 - Corrected a vendor drawing error.

3) CRN-6170 - Vendor approved voltage adjustment change from 15 VDC to 15.5 VDC for 18 gauge cable.

4) CRN-6193 - Changed relay operating position.

5) CRN-6104 - Minor installation change to eliminate interference between new drawers.

6) CRN-6094 - Changed length of rope used for pulling detector cables.

7) CRN-6132 - Revised connector clamp for triaxial cable. Superseded CRN-6130.

8) CRN-6130 - Provided tolerance for connector clamp.

9) CRN-6091 - Modified mounting tabs for amplifier boxes. (Note - The inspector verified welder qualification for this change).

10) CRN-6254 - Provided additional wide range calibration data.

11) CRN-6196 - Provided new data to change recorder's scale and UFSAR Table 7.2-1.

12) CRN-6078 - Removed hold points to allow implementation of work order.

13) CRN-6342 - Provided instructions to install "C" channel drawer connector J6.

The inspector did not have any safety concerns with the scoping changes and did not consider the number excessive.

Deviations From Pre-Op Test Procedure

The inspector reviewed 52 "deviations" from the pre-operational test procedure. The inspector verified all 52 deviations were approved by the Facility Review Group (FRG) which was the licensee's independent safety evaluation board. Most of the deviations were minor in nature and had no safety significance. All the deviations concerning design had been approved by Gamma-Metrics or Engineering. More than several deviations resulted as the consequence of a minor modification where "B and D" detectors were connected to "A and C" NI drawers to facilitate fuel loading. This minor modification was implemented at the request of Fuel Engineering. The inspector did not consider the problems encountered or the number of deviations to be excessive in this area as long as they were reviewed by FRG. However, the inspector determined the method of implementing deviations was cumbersome.

Wide Range NI Temporary System Alternation

The inspector reviewed the minor modification package and safety evaluation JPN-PSL-SEIS-96-028, "Wide Range NI Temporary System Alteration" approved May 24, 1996. The purpose of this minor modification was to provide for the temporary installation of coaxial jumpers cables to allow the connections of wide range NI detectors No. 2 (channel B) and No. 4 (channel D) to the preamplifiers inputs for channels A and B respectively.

This modification accommodated the fuel loading analysis which required that the fuel locations adjacent to detectors "B" and "D" be loaded first for fuel shuffling and to meet Technical Specifications (TS).

TS 3.3.1.1. provides the requirement for reactor protective instrumentation and TS 3.9.2 provides the requirement for refueling. This minor modification required several "deviations" to the Pre-Op testing. The inspector reviewed this minor modification to verify it met the requirements in procedure ENG-QI 1.2, "Minor Engineering Package" (modification) including the safety evaluation. No concerns were identified in this area.

Condition Reports

Six Condition Reports (CR) were initiated by Quality Assurance identifying concerns, problems, and non-conforming conditions with the Unit 1 NI Instrumentation System modification. The seventh CR was initiated as the result of it being identified by the inspector during a walkdown of the control room NI cabinets and work observation. Each CR was reviewed by the inspector to determine if the concern or condition was properly evaluated and appropriate corrective action initiated. The seven CRs are listed as follows:

1) CR 96-1358 - This is discussed above in E1.1.a "Work Scope." It is also discussed below in detail since it was considered significant.

2) CR 96-711 - The amplifiers boxes had their mounting tabs welded without removing the electronics. Gamma-Metrics stated no damage was expected due to the design of mounting the printed circuit board on standoffs.

3) CR 96-1443 - The 120 VAC power wire was damaged during removal of the "C" channel wide range drawer. Wire needs to be repaired.

4) CR 96-1480 - The triaxial cable for connector J6 on "C" channel drawer is very stiff and causes connector to break. The connector cable assembly was rebuilt using CRN 96-6342 twice. The work for the first rebuild not satisfactory and the connector had to be redone.

5) CR 96-1434 - Several instances in which the requirements in specification SPEC-IC-004 had not been met. The preamplifier power cable (J3) was 18 gage instead of 16 gauge. This required the actions in CRN 96-6170 (listed below). The existing wide range signal cables between the amplifiers and containment penetrations was listed as coaxial cable on drawing 8770-6390. Instead it was found to be triaxial for the original installation. CRN 96-6342 was initiated to install new connectors.

6) CR 96-1464 - Data from ECAD concerning the "C" channel source range detector indicated low impedance during testing. Gamma-Metrics was consulted for an evaluation and solution for corrective action.

7) CR 96-1481 - The inspector identified that cable support clamps were missing from four connector on each NI drawer in the control room. A long term solution for corrective action was being initiated by the licensee.

The inspector reviewed each CR and discussed its concern with engineering for their evaluation and the corrective action. The inspector verified that problems did exit. However, the problems were not considered significant and no safety concerns were identified.

<u>CR 96-1358</u>

The conditions listed in CR 96-1358 (CR) and the inspector findings (IF) are listed below:

1) CR - There was possible loss of design control.

   IF - The inspector did not identify any loss of design control. However, coordination between design engineering and I&C was initially weak. The initial design engineer assigned to the NI System project left the site. He was later replaced with a design engineering manager to coordinate the project.

2) CR - Key individuals who had been involved with the design and implementation are no longer available.

   IF - Two of the three individuals initially assigned to the NI project were no longer available. However, both engineering and I&C had a sufficient number of qualified engineers capable of completing the modification. The problem was not considered a loss of key personnel, it was more a lack of coordination since the I&C supervisors were performing more than one function. The I&C group in the Maintenance Department performed several functions that are unique to it. First level supervisor's duties included being a craft foreman, system engineer, installation engineer, test engineer, trouble shooter, etc. In several instances, the I&C supervisors were overwhelmed with work due to the refueling outage.

3) CR - There have been problems encountered during implementation which have required changes to "design" as well as numerous "deviations" from the FRG approved Pre-Op Test Procedure and vendor technical specifications manual.

IF - The inspector verified there had been problems encountered during implementation and testing. The problems encountered were mostly with cables and connectors during work implementation. The "deviations" were reviewed by the inspector and approved by FRG (safety review group) for the Pre-Op Testing.

Many of the "deviation" and "scoping changes" were written to incorporate corrective action and post maintenance testing after trouble shooting installation problems. This was identified by the inspector as a work control implementation weakness. This problem was corrected later by the licensee when new work orders (WO) were written with the appropriate post maintenance testing. For example, instead of initiating a "scope change" and "deviation" for trouble shooting, implementing corrective action, and testing, three new WOs 96016391, 96016395, and 96016397, all dated June 21, 1996, were written to "Repair MB NI Drawer". All three WOs had one specific task with appropriate post maintenance testing. All three WOs were approved by FRG. This method of using additional WOs improved control and implementation of the modification.

4) CR - The work package had become voluminous and unwieldy.

IF - The inspector agreed. The licensee initiated corrective action by implementing an "engineering implementation/test log" and using additional WOs to control the work as discussed above.

5) CR - Trouble shooting was further complicated by not having vendor engineering support on-site.

IF - A Gamma-Metric's design engineer and a field engineer were on-site to support the modification. The field engineer was scheduled to remain on-site until the modification was completed and turned over to Operations.

c. Conclusion

The inspector concluded that there was justification for the comments stated in CR 96-1358. I&C personnel were sometimes overwhelmed with problems and work. However, there was no loss of design control. The I&C supervisors and technicians were knowledgeable and technically capable of implementing the NI modification. The inspector considered the capabilities of I&C personnel to be a strength in the Maintenance Department.

The inspector verified that FRG reviewed all changes and deviations to ensure plant safety. Design engineering was made aware that I&C needed

additional support and was in the process of providing it. Program weaknesses for implementing changes and trouble shooting were identified and corrected using new work orders. The vendor, Gamma-Metrics, provided good engineering and field support with knowledgeable personnel. The inspector concluded the NI System modification was being completed in a satisfactory manner to ensure plant safety.

E1.2 Engineering Downsizing (37550)

a. Inspection Scope

The inspectors reviewed the functions of the current engineering organization and the anticipated changes following proposed downsizing, scheduled to take place on August 1, 1996.

b. Observations and Findings

The inspectors discussed with engineering management the current engineering organization, the functions of each of the groups, and the approximate number of people in each group. The licensee then discussed the changes in groups, functions, and numbers that would take place on August 1, 1996. Currently, the St. Lucie site is supported by an additional engineering organization located within Corporate Engineering to Juno Beach, Florida. Most of these Corporate Engineering functions are being transferred to the St. Lucie site (and to the Turkey Point site) and Corporate Engineering is being downsized. As a basis for future monitoring, the inspectors discussed current engineering work load, i.e., open Modification Packages, Condition Reports, etc.

c. Conclusion

To follow and evaluate any effect of the downsizing on the engineering function at St. Lucie, the inspectors will, on a periodic basis, use inspection procedure IP 37550, "Engineering." The objectives of this procedure are to "Evaluate the licensee's engineering activities, particularly the effectiveness of the engineering organization to perform routine and reactive site activities including the identification and resolution of technical issues and problems."

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Occupational Radiation Internal and External Exposure Control (83750)

a. Inspection Scope

The inspectors reviewed the personnel exposure records to verify radiation exposures were within regulatory limits and the licensee was implementing proper internal and external exposure control measures.

b. Observations and Findings

The inspectors found all internal and external personnel exposures were below regulatory limits. Tours of the Radiation Control Areas (RCAs) were made to verify that radiological areas were properly posted and controlled. Locked high radiation areas were found properly secured. The inspectors reviewed select licensee radiation surveys and made independent radiation surveys in those areas to verify radiological conditions were properly identified and posted.

The inspectors observed good use of engineering controls and work processes to control airborne radioactive contamination.

c. Conclusions

In general, the licensee appeared to be implementing effective radiological controls to minimize personnel exposures to internal and external radiation sources. No concerns with the licensee's internal or external exposure control programs were identified.

R1.2 Control of Radioactive Materials and Contamination (83750)

a. Inspection Scope

The inspectors reviewed licensee procedures for control of contaminated tools, discussed controls with tool room staff and radiation protection personnel, reviewed licensee radiation surveys of tool rooms, and made independent radiation surveys in tool rooms.

HP-2, "Florida Power and Light (FPL) Health Physics Manual," Rev 10, Dated August 24, 1995, described the radiation protection program at FPL's nuclear power plants. The FPL contamination guidelines were summarized in Table 4.2, "Contamination Guidelines."

The licensee's contamination limits for materials, tools, equipment and solid waste unconditionally released from the RCA were:

1,000 dpm/100 cm^2 for loose beta and gamma contamination and

5,000 dpm/100 cm^2 for fixed beta and gamma contamination (direct measurement)

The licensee's contamination limits for tools and equipment used in the RCA were:

1,000 dpm/100 cm^2 for loose beta and gamma contamination and

10 mrem/hr for fixed beta and gamma contamination

b. Observations and Findings

The inspectors noted that the licensee had made some positive changes in tool controls since the previous inspection. The inspectors observed the hot tool room located in the Unit 1 Reactor Auxiliary Building (RAB) was manned during the inspection and secured when unattended. Additionally, the licensee had combined two clean tool rooms located outside the RCA and the licensee was improving tool tracking and inventory capabilities. The licensee was also able to obtain enough temporary tool room personnel to staff tool rooms at all times during the peak outage period.

The licensee planned to reduce the total number of issued tools. A backlog of contaminated tools from previous outages had accumulated in storage locations within the RCA. Staff reductions in decontamination personnel had resulted in decreased tool decontamination efforts and increased levels of contaminated tools in storage.

As permitted by licensee procedures, some tools were designated for use within the RCA and were referred to as hot tools. The hot tools had specific contamination limits which were greater than unconditional release limits. These tools were identifiable with purple paint. The inspectors toured shops and warehouses and examined vehicles and "gang boxes" outside the RCA for hot tools. No hot tools were found outside the RCA. The inspectors also made radiation and contamination surveys in clean and hot tool rooms. No tools exceeding limits for clean or hot tools were identified by the inspectors.

In discussions with tool room personnel, the inspectors found many were unaware of the specific radiation and contamination limits for clean or hot tools. The temporary tool room personnel were generally less knowledgeable of the tool contamination limits. However, tool room personnel were not responsible for determining contamination levels of tools.

The inspectors reviewed routine and special surveys of licensee tool rooms. The licensee spent approximately 162 hours surveying tools during the period of June 13-16, 1996. The announced radiation protection inspection began June 17, 1996. The licensee also spent another 116 hours surveying tool rooms during the first three days of the inspection (June 17-19, 1996). The licensee's survey efforts in tool rooms during this seven day period were significant and not typical of routine monitoring.

During the licensee's surveys numerous tools were found outside the RCA having contamination levels in excess of the limits for use in clean areas. The licensee also identified numerous tools for use in the RCA having contamination limits in excess of the limits for hot tools.

The licensee's tool room surveys during the period of June 13-19, 1996, identified the following examples where contaminated tools were found outside the RCA:

On June 18, 1996, health physics technicians (HPTs) removed 12 M&TE tools from the licensee's clean tool room having contamination levels up to approximately 12,500 dpm/100 cm^2 (250 net counts per minute/probe).

On June 19, 1996, HPTs removed five rigging slings from the licensee's clean tool room having contamination levels from approximately 40,000 to 600,000 dpm/100 cm^2 (8,000 to 120,000 dpm/probe).

The licensee's tool room surveys during the period of June 13-19, 1996, identified the following examples where tools were found within the RCA having contamination levels in excess of the licensee's contamination limits for hot tools:

On June 13, 1996, HPTs removed nine tools from a temporary hot tool room having loose contamination levels from approximately 1,000 to 20,000 dpm/100 cm^2.

On June 14, 1996, HPTs removed five wrenches, from the Unit 1 hot tool room having loose contamination in the range of 1,000 to 4,000 dpm/100 cm^2.

On June 16, 1996, HPTs removed numerous tools from a temporary hot tool room having loose contamination levels from approximately 1,000 to 30,000 dpm/100 cm^2.

On June 16, 1996, HPTs removed numerous tools (licensee identified as two bags), from the Unit 1 hot tool room having loose contamination in the range of 1,000 to 120,000 dpm/100 cm^2.

In the February 1996, radiation protection program inspection the inspectors found a few contaminated tools in the hot tool room that were slightly above the licensee's limits. In response to the inspector's findings, the licensee secured the hot tool room when unattended for better control. A Non-Cited Violation (NCV) concerning the control of contaminated tools was identified at that time. The licensee identified all of the recent examples of tools having contamination levels in excess of the licensee's contaminated limits. However, these were additional examples of tools having contamination in excess of contamination limits previously identified by the inspectors in the February 1996, radiation protection inspection. Corrective measures implemented by the licensee following the NCV were inadequate to prevent the additional violations identified in the recent and extensive tool room surveys. The failure to control contaminated tools in accordance with licensee procedures is identified as a violation (VIO 50-335/96-09-06, "Failure to Control Contaminated Tools In Accordance

with Licensee Procedures"). The licensee opened a condition report for the purpose of identifying the cause of the contaminated tool violations and to cause appropriate corrective actions.

During tours of the licensee's facilities the inspectors found housekeeping was generally good. However, numerous drums containing low level contamination were still stored in the yard area within the RCA that were exposed to environmental conditions and could present problems during severe winds.

c. Conclusions

While the licensee was making progress in achieving controls for tools in general, the licensee's controls had not been effective in preventing contaminated tools from leaving the RCA or ensuring tools for use inside the RCA had contamination levels below the licensee's contamination limits.

R1.3 Maintaining Occupational Exposure ALARA (83750)

a. Inspection Scope

The inspectors reviewed the status of the licensee's collective dose for 1996 and the implementation of the person-rem budget program.

b. Observations and Findings

The inspectors attended an ALARA Review Board meeting held during the inspection. During the meeting, the inspectors noted the new ALARA dose budget program appeared to have strong management support and to have directly involved site department managers in the dose reduction process. Department managers were accountable for collective doses and required to take corrective actions to minimize collective dose for their departments. Managers were encouraged to utilize the corrective action program to capture successful activities into procedures and to document unsuccessful activities for appropriate corrective actions.

The collective doses for specific work activities were reviewed with ALARA personnel and the inspector inquired about specific tasks exceeding expected collective dose. The effects of recent staff reductions on site collective dose were also discussed with licensee personnel. Recent staff reductions had resulted in additional temporary and less experience personnel performing certain activities including shielding, insulation removal and decontamination during outages. It appeared that the use of temporary and less experienced personnel could reduce efficiency and therefore increase collective doses. The licensee had not quantified collective dose differences of experienced versus less experienced laborers for task and the inspector was unable to measure the impact that temporary personnel were having on collective dose. However, no significant collective dose problems were identified during the inspector's reviews.

The licensee's 1996 annual collective dose goal of 326 person-rem was based on routine Refueling Outage (RFO) activities and was one of the most challenging for the site. However, the work scope expansion for the Unit 1 Steam Generators (SGs) was significant enough to threaten achievement of the 1996 goal. The licensee had approximately 297 person-rem through June 19, 1996.

c. Conclusions

Management support for the ALARA program was good with increased management involvement in dose reduction activities. The dose budget program has increased site participation in reducing collective dose. Upper managements encouragement to document ALARA successes and failures in the corrective action program indicate understanding and willingness to implement quality control processes in ALARA activities. The unexpected SG work had significantly impacted the licensee's ability to achieve the challenging 1996 collective dose goals.

R5 Staff Training and Qualification in Radiation Protection and Chemistry (83750)

a. Inspection Scope

The inspectors reviewed the qualifications of certain site and vendor HPTs on site for the Unit 1 RFO. Licensee Technical Specifications 6.3.1 required that staff exceed the minimum qualification requirements specified in ANSI/ANS-3.1-1978, "American National Standard for Selection and Training of Nuclear Power Plant Personnel."

b. Observations and Findings

The inspectors requested a review of vendor HP resumes for technicians working in the on-going Unit 1 RFO. The inspectors also reviewed the qualifications of all site HPTs having less than five years of experience in FPL radiation protection programs.

The inspectors were able to review experience records for a portion of vendor HPTs hired for the on-going RFO. Vendor HP resumes were reviewed by the licensee to determine experience levels for meeting ANSI qualification requirements.

c. Conclusions

The inspectors determined that the licensee had not lowered qualification requirements for site and vendor HPTs. All site and vendor HPTs qualification records reviewed by the inspectors documented compliance with the applicable qualification requirements. No violations or deviations were identified.

F.2 Status of Fire Protection Facilities and Equipment

F.2.1 Oil Collection System for Unit 1 Reactor Coolant Pumps (71750)

a. Inspection Scope

The inspector evaluated the oil collection system for the Unit 1 reactor coolant pumps (RCPs) for compliance with the requirements of 10 CFR 50 Appendix R Section III.O.

b. Observations and Findings

The oil collection system for the Unit 1 RCPs was described by UFSAR Sections 2.5 (Item III.O) and 4.K. The system consisted of enclosures at each RCP to collect lubrication oil from all pressurized and unpressurized leakage points in the Raps. The inspector performed a walkdown inspection of the oil collection system for RCP 1B, associated drain piping and oil collection tank. The system was designed to collect the total quantity of oil from one RCP (190 gallons). The oil collection tank was provided with a VAREC Series 52 flame arrestor manufactured by the James M. Clontz Associates, Inc.

The inspector noted that the installed system met the description of the system described in the UFSAR. However, the system did not meet the requirements of 10 CFR 50 Appendix R Section III.O in that the tank was not sized to collect the total quantity of oil from all four RCPs. The UFSAR described this as an Appendix R Exemption and provided a justification for this exemption.

c. Conclusions

The oil collection system for the Unit 1 RCPs met the UFSAR commitments.

V. Management Meetings and Other Areas

X1 Review of UFSAR Commitments

A recent discovery of a licensee operating their facility in a manner contrary to the UFSAR description highlighted the need for additional verification that licensees were complying with the UFSAR commitments. While performing the inspections which are discussed in this report the inspectors reviewed applicable portions of the UFSAR that related to the areas inspected. The inspectors verified that the UFSAR wording was consistent with the observed plant practices, procedures, and parameters.

X2 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on July 9. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

W. Bladow, Site Quality Manager
H. Buchanan, Health Physics Supervisor
C. Burton, Site Services Manager
R. Dawson, Business Manager
D. Denver, Site Engineering Manager
R. Frechette, Chemistry Supervisor
P. Fulford, Operations Support and Testing Supervisor
C. Marple, Operations Supervisor
K. Heffelfinger, Protection Services Supervisor
J. Holt, Information Services Supervisor
H. Johnson, Operations Manager
T. Kreinberg, Nuclear Material Management Superintendent
J. Marchese, Maintenance Manager
C. O'Farrel, Reactor Engineering Supervisor
R. Olson, Instrument and Control Maintenance Supervisor
C. Pell, Outage Manager
J. Scarola, St. Lucie Plant General Manager
A. Stall, Site Vice President
E. Weinkam, Licensing Manager
C. Wood, System and Component Engineering Manager
W. White, Security Supervisor

Other licensee employees contacted included office, operations, engineering, maintenance, chemistry/radiation, and corporate personnel.

INSPECTION PROCEDURES USED

IP 37550:	Engineering
IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 62703:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 83750:	Occupational Radiation Exposure

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-335,389/96-09-01	VIO	"Failure to Control Overtime"
50-335,389/96-09-02	VIO	"Inadequate Procedure for Managing Overtime"
50-335,389/96-09-03	VIO	"Failure to Test the Reactor Cavity Pressure Relief Dampers In Accordance With The Vendor's Technical Manual"
50-335,389/96-09-04	VIO	"Failure To Perform Adequate Testing On The Reactor Cavity Pressure Relief Dampers"
50-335/96-09-06	VIO	"Failure to Control Contaminated Tools In Accordance with Licensee Procedures"

Closed

50-335/96-09-05	NCV	"Failure to Perform Proper Post-Maintenance Testing of HCV-25-1"

Discussed

50-335/95-015-03	VIO	"Failure to Follow Procedures and Document Abnormal Valve Position"

LIST OF ACRONYMS USED

ADM	Administrative Procedure
ALARA	As Low as Reasonably Achievable (radiation exposure)
ANPS	Assistant Nuclear Plant Supervisor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AP	Administrative Procedure
ATTN	Attention
cc	Cubic Centimeter
CCW	Component Cooling Water
CET	Core Exit Thermocouple
CFR	Code of Federal Regulations
cm	Centimeter
CR	Condition Report
CRN	Change Request Notice
dpm	Disintegration Per Minute
DPR	Demonstration Power Reactor (A type of operating license)
EDG	Emergency Diesel Generator
ENG	Engineering
ERDADS	Emergency Response Data Acquisition Display System
FCV	Flow Control Valve
FIC	Flow Indicating Controller
FPL	The Florida Power & Light Company
FR	Federal Regulation
FRG	Facility Review Group
FSAR	Final Safety Analysis Report
GMP	General Maintenance Procedure
HCV	Hydraulic Control Valve
HPSI	High Pressure Safety Injection (system)
HPT	Health Physics Technician
I&C	Instrumentation and Control
ICW	Intake Cooling Water
IF	Inspector Findings
IP	Inspection Procedure
JPN	(Juno Beach) Nuclear Engineering
LOCA	Loss of Coolant Accident
LPSI	Low Pressure Safety Injection (system)
M&TE	Measuring & Test Equipment
mrem	millirem
NCV	NonCited Violation (of NRC requirements)
NI	Nuclear Instrument
NPF	Nuclear Production Facility (a type of operating license)
NPS	Nuclear Plant Supervisor
NRC	Nuclear Regulatory Commission
NWE	Nuclear Watch Engineer
OP	Operating Procedure
OST	Operations Support and Testing
PC/M	Plant Change/Modification
PDR	NRC Public Document Room
PGM	Plant General Manager
PSL	Plant St. Lucie

PWO	Plant Work Order
QA	Quality Assurance
QI	Quality Instruction
RAB	Reactor Auxiliary Building
RCA	Radiation Control Area
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
REA	Request for Engineering Assistance
Rev	Revision
RFO	Refueling Outage
RII	Region II - Atlanta, Georgia (NRC)
RPS	Reactor Protection System
SDC	Shut Down Cooling
SER	Safety Evaluation Report
SG	Steam Generator
SPDS	Safety Parameter Display System
SRO	Senior Reactor [licensed] Operator
St.	Saint
TS	Technical Specification(s)
UFSAR	Updated Final Safety Analysis Report
USNRC	United States Nuclear Regulatory Commission
VAC	Volts Alternating Current
VDC	Volts Direct Current
VIO	Violation (of NRC requirements)
WO	Work Order