CATEGORY 1

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SUBJECT: LER 96-001-00:on 960108,fuel assembly was mispositioned due to inadequate self checking & mgt directions.Changed policy & procedure.W/960207 ltr.

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TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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DUKE POWER

February 7, 1996

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

Subject: Oconee Nuclear Station
Docket Nos. 5-269, -270, -287
Licensee Event Report 269/96-01
Problem Investigation Process No.: 5-096-0040

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report 269/96-01, concerning a mispositioned fuel assembly due to inadequate self checking and management direction.

This report is being submitted in accordance with 10 CFR 50.73 (a)(2)(i)(B) and (a)(2)(ii)(A). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

J. W. Hampton

J. W. Hampton

/fts

Attachment

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IE22

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Oconee Nuclear Station, Unit One	05000 269	1 OF 17

TITLE (4) Mispositioned Fuel Assembly Due To Inadequate Self Checking and Management Direction

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									Oconee, Unit Two	05000 270
01	08	96	96	-- 01 --	00	02	07	96	Oconee, Unit Three	05000 287

OPERATING MODE (9)	N
POWER LEVEL (10)	100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i) (B)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii) (A)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
L. V. Wilkie, Safety Review Manager	(803) 885-3518

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On December 14, 1995, with all three Oconee units at 100% Full power, a fuel handling team performing a fuel assembly (FA) inspection in the Unit 1&2 spent fuel pool (SFP) inadvertently left the FA unattended and suspended inside the SFP mast. It was discovered on January 8, 1996, by fuel handling personnel during check outs for planned fuel movements. The FA was reinserted into the SFP rack. The primary safety significance of the event was the potential uncovering of the FA during a postulated event requiring actuation of the Reactor Coolant Make-up function of the Standby Shutdown Facility (SSF), which uses the SFP as a water source. An engineering analysis concluded that the fuel cladding would not be breached during an SSF event with this FA in the mast. Therefore, 10CFR100 limits would not have been exceeded and the Final Safety Analysis Report (FSAR) analysis consequences would have bounded the event. However, having an unattended FA in the mast is outside the intent of Technical Specification 3.8 on fuel handling and 3.18 on the SSF. The root causes are inadequate self checking and lack of management expectations for formality and procedure use in fuel handling. Corrective actions include policy and procedure changes.

BACKGROUND

In addition to a Spent Fuel Pool (SFP) [EIIS:ND] where spent fuel is stored in racks submerged under borated water, Oconee Nuclear Station has an Interim Spent Fuel Storage Facility on site. There spent fuel is stored in dry containers, thus the term "dry cask storage" is used.

Fuel handling activities at Oconee are performed by members of a dedicated fuel handling maintenance crew. The fuel handling supervisor is a previously licensed Senior Reactor Operator. The crew's work activities are primarily fuel handling activities and plant crane [EIIS:RN] maintenance. A significant portion of the fuel handling crew's scheduled work involves shuffling spent fuel assemblies in the SFP and support of dry cask storage activities. The minimum crew number for operating the refueling bridge [EIIS:FHB] in the SFP is one bridge operator and one spotter. Fuel Handlers are qualified to Fuel Handling activities per Employee Training Qualification Standards.

OP/0/A/1506/01 (Fuel & Component Handling) is the "HOW TO" procedure for using the fuel handling bridge. It is an "Information Use" procedure which has no sign-offs, is performed from memory, and, by management policy, is not required to be at the job location.

Normally, OP/0/A/1503/09 (Documentation of Fuel Assemblies &/or Component Shuffle Within a SF Pool) is the "WHERE TO" procedure used to make miscellaneous fuel movements. An enclosure, initiated by Reactor Engineering, designates the fuel assemblies and/or control components to be moved, the starting locations, and the ending locations. The fuel handlers sign off each move as it is made.

Technical Specification 3.8 provides required prerequisites for fuel handling in the SFP. One requirement is that the SFP filtered ventilation system [EIIS:VF] must be operable, or fuel handling must be suspended. The SFP filtered ventilation system is considered inoperable whenever the fuel receiving bay door is open.

The Standby Shutdown Facility (SSF) [EIIS:NB] is designed to maintain the plant in a safe shutdown condition for a 72 hour period in the event of an Appendix R fire, a turbine building flood, a security event, a station blackout when the turbine driven emergency feedwater [EIIS:BA] pump [EIIS:P] is inoperable, or a tornado which renders the auxiliary service water and emergency feedwater systems inoperable. The SSF Reactor Coolant (RC) makeup pump [EIIS:CB] takes water from the SFP inventory in order to makeup to the Reactor Coolant System (RCS) [EIIS:AB] through the reactor coolant pump seals. In addition, SFP cooling may also be lost during an

SSF event such that boil-off of SFP water will also contribute to the loss of SFP inventory. The design basis of the SSF system will allow depletion of the SFP inventory to within one foot from the top of the SFP racks assuming no action to refill the SFP. Technical Specification 3.18.4 requires the SSF RC Makeup System be operable for each unit when the RCS is at or above 250°F.

EVENT DESCRIPTION

During Unit 1 EOC16 (End of Cycle 16) refueling outage, which started on Nov. 2, 1995 and concluded Dec 10, 1995, a fuel assembly (FA) was observed to have four intermediate spacer grids damaged. As part of the root cause evaluation, Reactor Engineer A desired to perform a visual inspection of FA NJ05T8 (FA-8), the fuel assembly which had been adjacent to the damaged assembly in the reactor core for the fuel cycle.

On December 14, 1995, at about 0900 hours, Reactor Engineer A contacted the Fuel Handling Supervisor for support in inspecting FA-8. The request was initially denied due to workload. Subsequently, one of the planned tasks was deferred several hours and the Fuel Handling Supervisor contacted Reactor Engineer A to schedule the inspection for after lunch.

Around 1300 hours, two Fuel Handlers and Reactor Engineer A entered Unit 1&2 Spent Fuel Pool (SFP) to inspect FA-8. A pre-job briefing was performed between Reactor Engineer A and Fuel Handler A but it covered only the basics of what needed to be done. Reactor Engineer A had no procedure or movement enclosure for this evolution, and, since the inspection did not involve leaving an FA in a new SFP location, Reactor Engineer A felt that he did not need one.

Fuel Handler A thought Reactor Engineer A had a procedure since he had called the control room to verify prerequisites listed in the normal fuel handling procedures. Reactor Engineer A stated that he called the control room out of habit. However, Reactor Engineer A stated that he did not inform the control room operator that fuel handling activities were about to take place.

Fuel Handler A operated the Unit 1&2 SFP bridge by memory, which is the normal practice. Fuel Handler A stated that he felt comfortable doing fuel handling steps by memory. Fuel Handler B acted as a runner for the job. Reactor Engineer A acted as a spotter, operated the video equipment, directed Fuel Handler A to SFP rack location K40, and directed mast operation (up/down) while video taping was in progress. During this

evolution, the mast and FA-8 were moved several feet east to improve the available lighting. Also, Reactor Engineer A requested Fuel Handler A to rotate the fuel mast 90 degrees and back while FA-8 extended below the mast. After some scratches were noted on FA-8's lower end fitting, FA-8 was returned to its proper location and lowered into the storage rack.

For comparison, Reactor Engineer A decided to look at another FA selected at random from the same cycle. Reactor Engineer A directed Fuel Handler A to SFP rack location L44 to pickup FA NJ06E7 (FA-7) and directed mast operation (up/down) while the FA was video taped. After observing similar scratches on FA-7, Reactor Engineer A stated that he had seen enough.

At this point neither Reactor Engineer A nor Fuel Handler A specifically stated a need to lower FA-7 prior to proceeding.

OP/0/A/1506/01, Limit and Precaution 2.27 directs personnel to not leave portable underwater lights and cameras in close proximity to irradiated fuel assemblies when not being used. Therefore, Reactor Engineer A began to raise the video camera. Due to the need to wipe down the pole and cable attached to the camera as it is raised, this task requires two people.

However, OP/0/A/1506/01, Limit and Precaution 2.22 directs personnel to turn off the Bridge hydraulic pump to prevent overheating when a Bridge is idle for 15 minutes or greater and the hoist is not engaged. In this case the hoist was engaged, but during the investigation it was learned that the Fuel Handling Supervisor has issued standing directions to turn off the pump even if the hoist is engaged. When the hydraulic pump is off, most of the control panel indications are either de-energized or go to a default state.

In accordance with these instructions, Fuel Handler A stopped the hydraulic pump, left the control console, and assisted Reactor Engineer A with pulling up and wiping down the video equipment. Once the camera was secured, Fuel Handler A returned to the control console and de-energized the bridge. During interviews, Fuel Handler A stated that he believed that he had lowered the FA back into the fuel rack and did not look at the control console indications to confirm this.

At 1342 hours, Fuel Handlers A and B exited the Unit 1&2 SFP with Reactor Engineer A. This left FA-7 suspended and unattended in the mast.

No fuel handling tasks in the Unit 1&2 SFP occurred over the next several weeks.

On January 8, 1996, at approximately 1030 hours, Fuel Handlers A and C entered the Unit 1&2 SFP to start preparation for loading a dry cask later in the week. When Fuel Handler A energized the bridge and started the hydraulic pump, he observed the control console indications and realized that a FA was in the mast. Fuel Handlers A and C initially assumed that the FA had been left in the mast recently by other members of the crew. Fuel Handlers A and C made the decision to lower the FA in the open rack at location L44 to allow an identification of the FA in order to determine where it should be and to trace the last known movement to determine who was responsible.

While Fuel Handler A lowered FA-7 into the storage rack, Fuel Handler C called the Fuel Handling Supervisor and informed him of the discovery and that Fuel Handler A had lowered the FA in the empty rack at L44. Fuel Handlers A and C identified the FA as NJ06E7 at Unit 1&2 SFP rack location L44.

At 1130 hours, the Fuel Handling Supervisor called the Rotating Equipment Manager and Reactor Engineering to report the event. It was verified that FA-7 was the last FA moved in Unit 1&2 SFP.

At 1230 hours, a meeting was held to discuss the event. The video tape from 12/14/95 was reviewed to see if the tape had shown the FA being put back down in the pool. The personnel present concluded that FA-7 had been in the fuel mast from 12/14/95 until 1/8/96. All three Oconee units were at 100 % full power throughout this period.

The design basis of the SSF system will allow depletion of the SFP inventory to within one foot from the top of the SFP racks assuming no action to refill the SFP. A concern was raised that FA-7 could have been uncovered by an SSF event, with the potential for heating to clad failure with resultant release of fission products. However, no analysis existed to determine if clad failure would occur or if the severity of the releases would exceed limits from either the FSAR analysis or 10CFR100. Thus there was a concern that the SSF might have been unable to perform its intended function and would need to be considered past inoperable. Therefore, one action item from the 1230 meeting was to start an operability evaluation which would include calculation of expected clad temperatures and potential releases.

The Maintenance Superintendent (who was acting as the Station Manager) discussed the event during the Station Manager's staff meeting at 1330 hours. The Operations Superintendent was at the meeting and assumed the control room knew of the event.

At 1500 hours, after the staff meeting, the Maintenance Superintendent, the Rotating Equipment Manager and Fuel Handling Supervisor went to inform the ONS NRC Resident Inspectors of the event.

After briefing the senior resident, the Maintenance Superintendent, Rotating Equipment Manager, and Fuel Handling Supervisor discussed the situation and decided not to continue with fuel handling until procedures were revised to prevent this event from reoccurring.

At about 1800 hours, the Senior VP of Nuclear Generation and the Site VP discussed the event and decided to initiate a Significant Event Investigation Team (SEIT).

Throughout this period, the control room was not informed of the discovery of the FA in the mast. On 1/9/96, at about 0630 hours, an NRC resident asked control room operators about the log entry for the event. This was the first time Operations shift had heard about the event.

At 0800 hours, this event was discussed in the daily site direction meeting. Site management present discussed issues related to past operability and reportability. The information available at that time was insufficient to reach a conclusion.

At 1414 hours, a log entry was made in Unit 1 Log about the event. Notes were added on Reactor Operator (RO), Control Room Senior Reactor Operator (SRO), and Unit Shift Supervisor's turnover sheets not to move fuel in 1&2 and/or 3 SFP until after the SEIT investigation was completed.

Discussions of operability and reportability issues continued. Issues discussed included compliance with Technical Specifications (TS) and FSAR analyses of fuel damage and resultant releases. TS that potentially apply in this case are 3.8, Fuel Movement and Storage in the Spent Fuel Pool, and 3.18, Standby Shutdown Facility.

TS 3.8 was initially not considered to apply, based on an interpretation that FA-7 was not moving while left in the mast. By that interpretation, fuel handling was not in progress and, therefore, the TS was not exceeded.

TS 3.18.4 requires the SSF RC Makeup System be operable for each unit at or above 250°F in the RCS. During an SSF event the SSF RC makeup pump takes suction from the SFP and can allow depletion of the SFP inventory such that FA-7 would be uncovered. Preliminary engineering calculations indicated possible heating to clad failure with resultant release of fission products. This could result in dose consequences beyond the licensing basis.

At 1700 hours, a decision was made to make a 1 hour NRC Emergency Notification System call, based on management's conclusion that these consequences represented an unanalyzed condition that could significantly compromise plant safety. The notification was made at 1755 hours.

On 1/10/96, the SEIT arrived on site and began an investigation. On 1/12/96, the SEIT presented its preliminary findings in a formal exit with site management and the Senior VP of Nuclear Generation.

One concern raised by the SEIT was the interpretation that leaving a FA in the mast met the requirement to suspend fuel handling. A survey of industry practices revealed that all of the other sites contacted defined fuel handling to include any time an assembly was supported by the fuel handling bridge or crane. These other sites interpreted "suspension of fuel movement" to mean that fuel movement should be continued until any FA in a raised position could be moved to a safe location and lowered.

Applying this more conservative interpretation of "fuel handling" resulted in the conclusion that TS 3.8 should be applied the entire time FA-7 was in the fuel mast. Since the fuel receiving bay door was opened at various times during the period, making the filtered ventilation system inoperable, the new interpretation would mean that the intent of TS 3.8.12 was not met.

The operability calculations and analysis were completed and the results are discussed in more detail in the "Safety Analysis" section of this report. The analysis showed that FA-7 would not be damaged and would not result in off site releases exceeding 10CFR100 limits. However, another FA with a higher decay heat potentially could. Therefore, management concluded that the condition of a FA being located within the SFP mast during an SSF event is not in compliance with the intent of TS 3.18.

Therefore, in addition to being reportable as an unanalyzed condition that could significantly compromise plant safety, this event would also be reportable as a condition outside the intent of Technical Specifications.

In response to the SEIT preliminary concerns, "Short Term" actions were initiated to enhance programs, policies, and procedures to address the SEIT recommendations and observations. These were primarily aimed at those items needed to resume limited fuel shuffles in preparation for dry cask storage and new fuel receipt prior to a refueling outage on Unit 2, currently scheduled for late March, 1996.

On Feb. 1, 1996, the SEIT issued its final report. The root causes identified are the same as the root causes listed below.

CONCLUSIONS

The root causes of this event are related to inadequate barriers intended to minimize the potential for this type of error. Two root causes for the event have been determined:

The first root cause of this event is the failure of Fuel Handler A to self-check his actions. This was a skill based error resulting from a momentary memory lapse while performing routine actions using an Information Use procedure.

The second root cause to this event is the lack of management expectations for formality in all aspects of the fuel handling process. The lack of formality was exhibited in the following actions, which were in accordance with management's expectations at the time for this type of work in the spent fuel pool, leading up to the leaving of the FA in the mast:

1. The failure to write and process a work request for the conduct of this activity.

2. The perception that no task specific procedure was required to conduct this activity.

3. OP/0/A/1506/01 (Fuel & Component Handling) was being performed from memory because it was an Information Use procedure and was not required to be at the job location. Performing procedures from memory will increase the risk of human error. Requirements of OP/0/A/1506/01 were not met in that:

   a) The Control Room was not specifically notified that fuel handling was in progress in the Spent Fuel Pool (SFP).

   b) Fuel Handler A rotated the mast 90 degrees and back at the request of Reactor Engineer A. This was performed while the FA was not "full up" in the mast.

   c) Steps to lower a FA and disconnect from the fuel grapple are included in the procedure but the omission of those steps resulted in FA-7 being left suspended inside the fuel mast.

d) Limit and Precaution 2.22 directs that "When a Bridge is idle for 15 minutes or greater and the hoist is not engaged, turn off the Bridge hydraulic pump to prevent overheating." This condition was not met when Fuel Handler A secured the hydraulic pump because the hoist was engaged. Due to workarounds with the hydraulic pump and instruction from the Fuel Handling Supervisor, this had become a common fuel handling practice.

4. Inadequacy of OP/0/A/1506/01 (Fuel & Component Handling) in that it did not provide steps for the fuel handler to verify that the fuel bridge mast was empty prior to shutting down the bridge.

5. The failure to provide an adequate pre-job briefing for the evolution.

The pre-job briefing did not address roles and responsibilities of the individuals involved. During most of the activities, Fuel Handler A was acting under the direction of Reactor Engineer A. This potentially led to an expectation on the part of Fuel Handler A for Reactor Engineer A to instruct him to lower the FA. Reactor Engineer A felt it was not his responsibility to ensure that FA-7 was lowered back into the SFP racks.

Past industry and site experience was reviewed to determine if this event is recurring. It was concluded that industry operating experience has not been used effectively at Oconee to prevent fuel handling events. SER 91-15, as an example, identified fuel mispositioning events that occurred within the industry due in part to inadequate independent verification and self-verification techniques. Oconee reviewed the SER, revised refueling procedures, enhanced methods of fuel handlers communication, and evaluated training in response to this SER. However, these corrective actions were ineffective in preventing four fuel mispositioning events that occurred in 1992 through 1994.

An operating experience review was performed using the Oconee Problem Investigation Process (PIP) data base in the area of fuel handling activities to look for similar events with root causes similar to this event. Attachment A to this report summarizes past fuel handling events and the related NRC violations.

The first root cause (self-verification as it relates to fuel handling work practices) has contributed to four events resulting in three NRC violations at Oconee during the period of 1992 through 1995.

The second root cause (lack of management expectations for formality in all aspects of the fuel handling process) has also contributed to fuel handling events at Oconee (particularly PIP 1-094-0707 and the associated NRC violation of August 2, 1994).

Therefore, it is concluded that this event is recurring with respect to both root causes. The repetitive nature of these fuel handling events demonstrate the lack of full use of lessons learned from previous events and application of too narrow a scope for corrective actions.

There were no radioactive releases, personnel injuries or over exposures, or NPRDS reportable equipment problems associated with this event.

CORRECTIVE ACTIONS

Immediate

1. Fuel Handlers lowered the fuel assembly into a Spent Fuel Pool (SFP) storage rack location.

2. Mechanical Maintenance management suspended fuel handling activities pending procedure changes.

Subsequent

1. Engineering calculations were performed and this event was analyzed with respect to the potential for exceeding design basis releases.

Planned

1. Step by step procedures will be required for all fuel movements.

2. A procedure checklist will be provided to assure that the fuel mast is returned to a proper end state at the conclusion of fuel handling.

3. Formalized pre-job briefings for all fuel related activities in the SFP will be implemented.

4. Appropriate personnel corrective actions will be taken in accordance with Duke Power policies.

5. A Self Initiated Technical Audit (SITA) will be performed to provide a broader review of fuel handling and other SFP activities and work processes.

Planned corrective actions 1 through 5 are considered Commitments to the NRC. They are the only items included in this report intended to be NRC Commitments.

SAFETY ANALYSIS

The consequences of the failure of a fuel assembly (FA) in the spent fuel pool (SFP) are analyzed in the Final Safety Analysis Report (FSAR), Section 15.11.2.1, "Single Fuel Assembly Handling Accidents". The FSAR accident scenario is a radioactive release from all 208 fuel rods. This accident is assumed to occur under at least 9 feet of water for iodine retention. The dose calculation with the FSAR initial condition assumptions of release inventories and conditions yields a dose of .66 rem whole body and 174 rem thyroid at the site boundary.

During an event requiring the Standby Shutdown Facility (SSF) Reactor Coolant (RC) makeup pump, FA NJ06E7 (FA-7) would have been uncovered by the decreasing inventory of the SFP. A heat up calculation of air cooling of the FA has been performed using the actual decay time after shutdown assuming only radial free convection and radiation. Results indicate a maximum cladding wall temperature at the top of the FA of 1022 degrees F. Potential damage mechanisms and the applicable limiting temperatures are:

cladding creep out (ballooning) and rupture	1150 deg F.
accelerated oxidation	1600 deg F.
metal water reaction	2200 deg F.
enhanced fission gas release from the UO2 pellet matrix	2450 deg F.
zircaloy melting	3400 deg F.

This calculation shows that cladding integrity would be maintained and no effluent radiation release occurs. Therefore, the existing analysis in Section 15.11.2.1 is still bounding.

An estimation was also performed for the most limiting decay heat load possible. In this case a high powered assembly, only 72 hours after subcriticality, was assumed in the mast and cooled by air and radiation. This analysis determined a maximum cladding temperature of 2000 degrees F. In this scenario, damage to the cladding would occur, and there would be no iodine retention in water, so the release of radiation from the assembly would be significant.

Depletion of the SFP inventory removes the majority of the shielding from the spent fuel assemblies such that direct radiation shine from the spent fuel will become significant. However, the SFP walls provide lateral shielding so the direct radiation shine is primarily in a vertical direction. Since the top of FA-7 was approximately 9 feet below the SFP grade, this will only add a small amount of additional direct radiation to either the on-site or off-site dose rate.

Since the SFP inventory must be eventually replenished remotely, having FA-7 in the fuel mast does not impose any additional restrictions to the operability of the SSF RC makeup system.

During the time period of interest, no spent fuel was moved in the SFP. Since the fuel mast provides a positive mechanical lock for the spent FA and the SFP bridge is seismically designed, no additional potential for a fuel handling accident existed.

Using the updated Oconee PRA model, the annual frequency of an event relying on the standby shutdown facility for core damage mitigation is 3.3 E-04. For the 25 day period FA-7 was in the fuel mast, the probability becomes 2.3 E-05. Furthermore, typical PRA calculations utilize a 24 hour minimum time for the system relied upon to mitigate the accident. In this case a time in the range of 36-40 hours would have been available before the SFP inventory is depleted to a level exposing a portion of the FA.

In conclusion, during the period from Dec 14, 1995 to Jan. 8, 1996, when FA-7 was suspended in the fuel mast, FA-7 was in a static, stable position such that the probability of fuel damage by another mechanism (collision, dropped object, seismic event, etc.) was remote. No SSF event occurred during this period. FA-7 was not damaged and did not release any radioactive materials to the public. In the unlikely happenstance that a SSF event actually did occur, an extensive period of up to 36 to 40 hours would have been available for compensatory actions to be taken prior to uncovering FA-7. Additionally, calculations show that FA-7 would have been adequately air cooled and no damage would be expected. Therefore, the health and safety of the public was not affected by this event.

ATTACHMENT A

OPERATING EXPERIENCE REVIEW

Oconee Fuel Placement Events

PIP# *	Description
1-O92-0723	Wrong fuel assembly (FA) was placed into Unit 1 Reactor Core during refueling activities as a result of inadequate self-check and independent verification. Changes to the refueling procedure were implemented as corrective actions to prevent recurrence.
1-O92-0724	Wrong FA was placed into Unit 1 Reactor Core during refueling activities as a result of inadequate self-check and independent verification. Changes to the refueling procedure were implemented as corrective actions to prevent recurrence
1-O94-0707	Refueling sequence was altered at the request of reactor engineers to observe nuclear instrumentation response without proper documentation and procedural control. This was a non-conservative decision made by the SRO in charge of fuel handling, Reactor Engineer, and the Fuel Handling Supervisor. Corrective actions to prevent recurrence involved a change in the refueling procedure to prohibit sequence deviations without the use of a procedure change or test procedure.
1-O94-0714	A wrong FA was placed into Unit 1 Reactor Core during refueling activities as a result of inadequate self-check and independent verification. Corrective actions to prevent recurrence involved changes to procedures and methods of independent verification.

* PIP = PROBLEM INVESTIGATION PROCESS

Oconee Fuel Movement Events/Concerns

PIP#	Description
2-092-0024	A FA and control rod was damaged while the FA was being positioned for repair. The procedure was not reviewed prior to the move and the control rod and FA were damaged. Corrective actions to prevent recurrence involved procedure changes and pedestal modifications.
3-092-0470	Bent spider assemblies causes delay in removal of burnable poison rods from two fuel assemblies. It could not be determined whether the damage occurred as a result of previous fuel handling activities by Duke or by the fuel vendor. Corrective actions involved manufacturing a component sizing template to be used by Quality Assurance during the component inspection performed upon unloading of the new fuel assemblies.
2-093-0431	An intermediate grid strap became torn and separated from its FA during refueling operation. This type of damage is caused when the grid straps of adjacent assemblies snag each other during fuel movements made in the core. Corrective actions to prevent recurrence involved changes to the refueling procedure to provide new guidance to prevent FA grid strap damage.
3-094-0204	A dummy control rod assembly located in the deep end of the fuel transfer canal was struck while transporting the core support assembly. This was a result of inadequate self-check of clearances. Crane control and water clarity problems contributed to the problem. Transport had to be halted to perform inspections of the core support assembly, the transfer canal liner plate, and the fuel storage racks. Corrective actions to prevent recurrence involved procedure changes to incorporate preventive measures.
1-095-1429	During reactor defueling activities, Spent Fuel Pool (SFP) bridge hoist and grapple operation was hampered several times due to unexpected interference with consolidated fuel canisters. This interference problems in disengaging from fuel assemblies. Corrective actions involved moving the consolidated fuel canisters to an area of the SFP that is outside of the off-load area.
1-095-1462	FA NJO776 was found to have significant structural damage on four consecutive intermediate spacer grids on the southwest corner. No fuel rod damage was found or suspected.

INDUSTRY OPERATING EXPERIENCE

PIP#	Description
1-C95-0184	During set up of the B&W Fuel Reconstitution, elevator parts sheared/fell from the elevator into the cask area. The elevator part apparently sheared when it contacted the cask area wall. Elevator design deficiency and worker attention to detail contributed to this event.
1-M93-0055	Several new fuel assemblies were received and placed in storage cells that were not in accordance with procedure. Root causes were failure to follow procedure and inattention to detail. The fuel components were not adversely affected.
1-M93-0414	A control rod was inserted in the wrong FA. Corrective actions involved procedure changes and personnel training to prevent recurrence.
2-M93-0676	A contractor personnel failed to follow procedural requirements for handling fuel rods during reconstitution activities, which resulted in severely bent fuel rod and subsequent challenge to the fuel cladding integrity. This resulted in a NRC level IV violation (PIP 2-M93-0917) for the failure of contractor personnel to follow procedural requirements.
1-M96-0002	A sequence within the FA-Insert shuffle procedure was performed incorrectly resulting in the misposition of a thimble plug in the SFP. The verification process identified and corrected this discrepancy. No corrective actions to prevent recurrence were identified.
SER 91-15	This report describes six industry fuel mispositioning events during refueling and defueling activities as a result of inadequacies in procedures, independent verification, and training. Oconee's review of this event resulted in changes to refueling procedure changes and methods of communication.
SER 94-4	This report describes six specific industry events that involve human performance deficiencies while handling reactor core components that resulted in actual FA or other core component damage, damage to refueling equipment, and/or increased potential for damage to fuel or other core components.

Oconee incorporated these industry events and lessons learned into the operations fuel handling lesson plan.

IN 94-13 This report describes potential problems resulting from inadequate oversight of refueling operations and inadequate performance on the part of refueling personnel based on four industry events.

Oconee's review of this report resulted in no recommended actions based on actions taken with SER 94-4.

IN 94-13, Sup. 1 This report describes an industry event involving unauthorized movement of a defective spent fuel rod.

Oconee's review of this report resulted in no recommended actions.

OCONEE NRC INSPECTIONS

NRC Level IV Violation (November 21, 1990)

One example of a failure to adequately implement a refueling procedure that resulted in a FA being placed in the wrong location in the core. Root causes were operator error and poor visibility in the SFP. Corrective actions to prevent recurrence involved counseling the bridge operator.

NRC Level IV Violation (September 17, 1991)

One example of failure to adequately implement a refueling procedure that resulted in a FA being placed in the wrong spent fuel location. Root causes were insufficient attention to detail, insufficient procedure detail and communication errors. Corrective actions to prevent recurrence involved procedural changes and fuel handling training

NRC Level IV Violation (February 22, 1993)

Two examples of failure to implement refueling procedures that resulted in two fuel assemblies being placed in the wrong location in the core. Root cause was inadequate self-checking. Corrective actions to prevent recurrence involved procedural changes. (Covered by PIPs 1-O92-0723 and 1-O92-0724)

NRC Level IV Violation (August 2, 1994)

Refueling sequence was altered to observe nuclear instrumentation response without proper documentation and procedural control. This was performed at the request of Reactor Engineering personnel. (Covered by PIP 1-O94-0707)

NRC Level IV Violation with Civil Penalty (August 2, 1994)

A FA retrieved from the wrong spent fuel location and placed in the reactor core. Root causes were inadequate self-check and independent verification. This was the fourth occurrence of failure to identify and adequately verify FA locations. Corrective actions to prevent recurrence involved procedural changes and personnel training. (Covered by PIP 1-O94-0714 and PIP 1-O94-0707)