NUREG-1038
Supplement No. 1

Safety Evaluation Report related to the operation of Shearon Harris Nuclear Power Plant, Unit No. 1

Docket No. STN 50-400

Carolina Power and Light Company
North Carolina Eastern Municipal Power Agency

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

June 1984

UNITED STATES NUCLEAR REGULATORY COMMISSION

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ABSTRACT

This report, Supplement No. 1 to the Safety Evaluation Report for the application filed by the Carolina Power and Light Company and North Carolina Eastern Municipal Power Agency (the applicant) for license to operate the Shearon Harris Nuclear Power Plant Unit 1 (Docket No. 50-400), has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement reports the status of certain items that had not been resolved at the time of publication of the Safety Evaluation Report.

6/28/84

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APPENDICES

1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

In November 1983, the Nuclear Regulatory Commission staff (NRC staff or staff) issued a Safety Evaluation Report, NUREG-1038, regarding the application by Carolina Power and Light Company and North Carolina Eastern Municipal Power Agency (the applicant) for a license to operate the Shearon Harris Nuclear Power Plant Unit 1. This report is the first supplement to that Safety Evaluation Report (SER).

This supplement provides more recent information regarding resolution of some of the open items identified in the SER. This supplement also provides and discusses the recommendations of the Advisory Committee on Reactor Safeguards (ACRS) in its report on the Shearon Harris Plant, dated January 16, 1984.

Each of the following sections or appendices of this supplement is numbered the same as the section or appendix of the SER that is being updated, and the discussions are supplementary to and not in lieu of the discussion in the SER unless otherwise noted. Accordingly, Appendix A is an updated bibliography,* Appendix D is a list of abbreviations used in this supplement. Appendix E is a list of principal contributors to this supplement. Appendix G is a copy of the ACRS report.

The Project Manager is Bart C. Buckley; he may be reached on (301) 492-8379.

1.7 Summary of Outstanding Issues

Section 1.7 of the SER noted that certain information had not yet been provided by the applicant for several identified items. This supplement updates those items for which additional information has subsequently been provided by the applicant. These items, and the sections of this supplement discussing the review conclusions, are

(2) Missiles Outside Containment (3.5.1.1)
(13) Emergency Preparedness (13.3)
(14) I.D.1 Control Room Design Review (18)
     II.F.2 ICC Instrumentation (4.4.6)

The remaining outstanding items and the references to applicable sections of the SER are given in Table 1.2 below.

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*Availability of all material cited is described on the inside front cover of this report.

Table 1.2 Outstanding items

Item	SER Section(s)
(1) Design of retaining wall	2.5.4
(2) Missiles outside containment	3.5.1.1
(3) Functional capability of Class 1 auxiliary piping systems	3.9.3
(4) Control of minimum wall thickness in ASME Class 1, 2, and 3 piping systems	3.9.3
(5) Equipment qualification	3.10, 3.11
(6) Preservice/inservice inspection program	5.2.4, 6.6
(7) Periodic testing of instrument air quality	9.3.1
(8) Fire protection	9.5.1
(9) Unmonitored release of condenser discharge during hogging operations	10.4.2, 11.5
(10) Method of estimating noble gas activity from atmospheric steam dump valves	10.4.2, 11.5
(11) Monitoring of all inputs to the service water system	11.5
(12) Emergency preparedness	13.3
(13) Steam generator tube rupture isolation time	15.6.3
(14) TMI Action Plan Items (NUREG-0737 and Supplement No. 1 to NUREG-0737)	
I.A.1.2 Shift supervisor administrative duties	13.5.1
I.C.2 Shift and relief turnover procedures	13.5.1
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III.D.1.1 Leak reduction program	9.3.5

1.8 Confirmatory Issues

Section 1.8 of the SER stated that certain confirmatory information will be provided by the applicant. One of these items, identified as item (30), Radiation Protection Manager (Section 12.5) in the SER, is resolved in Section 12.5.1 of this supplement. The remaining confirmatory issues, with reference to the applicable sections of the SER, are listed in Table 1.3.

Table 1.3 Confirmatory issues

Issue		SER Section(s)
(1)	Emergency plan meteorological program	2.3.3
(2)	Revision of FSAR Table 3.2.1-1	3.2.2
(3)	Turbine missiles	3.5.1.3
(4)	Design documentation of ASME components	3.9.3.1
(5)	Piping supports	3.9.2
(6)	Plant-specific submittal concerning testing of safety and relief valves	3.9.3.2
(7)	Leak rate test program for pressure isolation valves	3.9.6
(8)	Calculation of ultimate strength capacity of containment building under uniform internal pressure	3.8
(9)	Additional information on excore detectors	4.3
(10)	PORV setpoint values	5.2.2
(11)	Revised pressure-temperature curves	5.3.2
(12)	Examination of steam generators and NUREG-1014 revisions	5.4.2.2
(13)	Revision of FSAR on containment penetrations	6.2.4
(14)	Additional information on adequacy of the ECCS during shutdown and startup	6.3.5.1
(15)	Design modifications for automatic reactor trip using shunt coil trip attachment	7.2.2.4
(16)	Solid-state logic protection system test circuit	7.3.3.11
(17)	Testing for remote shutdown operation	7.4.2.2
(18)	RCS overpressure protection during low temperature operation	7.6.2.2

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Issue		SER Section(s)
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(20)	Use of load sequencer with offsite power	8.4.7
(21)	Compliance with Phase I and Phase II of NUREG-0612	9.1.5
(22)	Pressure differential alarms	9.4.5.2
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(24)	Radiation monitors for turbine building vent stack	10.4
(25)	Ability to continuously sample radioiodine and particulates (condenser vacuum pump effluent)	10.4.2
(26)	Location of high range noble gas monitors (turbine building vent)	10.4.2, 10.4.3 11.5
(27)	Drawings for the filters handling sludge	11.4.1
(28)	Process control program	11.4.1
(29)	Polymer binder system	11.4.1
(30)	Corporate management and technical support organization	13.1.1.6
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	• Additional testing to verify the capacity of the steam generator safety and relief valves	
	• Amend FSAR to incorporate additional information on AWP endurance tests	
	• Expansion of natural circulation tests to fully comply with NUREG-0737, Item I.G.1	
(32)	TMI Action Plan (NUREG-0737)	
	I.C.7 NSSS vendor review process	13.5.2.3
	II.K.3.5 Automatic trip of RCPs during LOCA	15.9.9

1.9 License Conditions

The proposed license condition related to security plan adherence to the regulations was inadvertently omitted from the Shearon Harris SER dated November 1983.

(6) Security plan adherence to the regulations (13.7)

3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.5 Missile Protection

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Outside Containment)

SER Section 3.5.1.1 contained three open issues. These issues are discussed below as Items 1, 2, and 3. During the review process, a fourth issue was raised concerning the possibility of missiles from pumps affecting safety-related systems, structures, or components. This issue, designated as Item 4, is also discussed below.

(1) Fan Blades as Missiles

The staff was concerned that fan blades could become missiles as illustrated by the event at Palo Verde in 1982. In that event, a fan blade became an internally generated missile and damaged the containment liner.

In response to the staff's concern, the applicant stated that specifications for inline, centrifugal, and axial fans require that fan housing design prevent the expulsion of any missiles generated by fans operating at maximum conditions, as adjusted in the field, into areas outside the housing. To accomplish this, vendors perform analyses and furnish calculations to show that expulsion of postulated fan-generated missiles is precluded by the fan housing design. The staff finds this acceptable.

(2) Damage from Missile Resulting from Failure of the AFW Turbine-Driven Pump

In response to this staff concern, the applicant had stated that the turbine is designed with redundant overspeed protection to prevent failure as a result of overspeed; in addition, in case of such failure, the motor-driven auxiliary feedwater (AFW) pumps are protected by barrier walls. Finally, the applicant provided the results of an analysis of AFW turbine-missile trajectories and an examination of physical plant arrangement that conclude that there will be no adverse safety consequences in the event of a turbine missile. The staff finds this acceptable.

(3) Omission of Essential Services Chilled Water System (ESCWS) and Waste Process Building Cooling Water System (WPBCWS) from List of Systems Requiring Protection from Internally Generated Missiles

The SER noted that neither the ESCWS or the WPBCWS appears on the list of structures, systems, and components requiring protection against internally generated missiles (outside containment). In response, the applicant stated that the following had been considered as potential sources of missiles that could damage the ESCWS: high pressure systems, rotating machinery, gravitational missiles, and secondary missiles (resulting from the impact of primary missiles). However, the applicant concluded that these missiles were either not credible or would not affect safety-related equipment in the ESCWS area.

The staff finds this conclusion to be unacceptable. Missiles from these potential sources are considered credible unless some deliberate element in the design or extra precaution is provided to prevent their generation. Design of equipment to appropriate codes is not a satisfactory means for preventing missile generation. Therefore, to justify the conclusion that such missiles are not credible, the applicant must show, in detail, that the design specifically considered the problem of missile generation from these potential sources, or the applicant must show that the ESCWS is protected against such missiles. Either of these approaches will satisfy the staff's concern regarding protection of the ESCWS against internally generated missiles outside containment.

As for the WPBCSW, the applicant noted that no adverse safety or radiological impact results from failure of the nonnuclear WPBCWS and, therefore, the WPBCWS need not be protected against missiles. The staff finds this acceptable.

(4) Internally Generated Missiles from Pumps

During the staff review, a concern arose regarding the possibility of internally generated missiles resulting from pump failure. The applicant was made aware of this concern, and, in response, the applicant noted that missiles from pumps within the nuclear steam supply system (NSSS) scope that are outside of containment have been designed so that their maximum no-load speed is equivalent to their operating speed. Thus, a sudden loss of load (resulting, for example, from a line break) will not result in the generation of missiles. Further, the FSAR states that the balance-of-plant (BOP) is designed so that missiles from internal sources will not damage engineered safety features in a way that would jeopardize the minimum required safety functions.

This staff finds this unacceptable. Although the pumps in the NSSS scope are prevented from overspeeding, the staff concern relates to the possibility of missile generation from well-designed pumps operating normally (see Item 3, above). In addition, the applicant must provide detailed information to explain how safety-related structures, systems, and components are protected against internally generated missiles generated from pumps within the BOP scope.

(5) Conclusion

The staff finds the applicant's resolution of Items 1 and 2 acceptable. However, the applicant must provide further information regarding protection of the ESCWS against internally generated missiles outside of containment (Item 3) and the protection of safety-related structures, systems, and components against the effects of missiles generated from all pumps outside of containment (Item 4).

4 REACTOR

4.4 Thermal-Hydraulic Design

4.4.6 NUREG-0737 Item II.F.2, Instrumentation for Inadequate Core Cooling Detection

NUREG-0737 Item II.F.2 clarifies the requirements for inadequate core cooling instrumentation (ICCI) that is to be installed and operational before fuel load. On November 4, 1982, the Commission determined that an instrumentation system for detection of inadequate core cooling (ICC) consisting of an upgraded subcooling margin monitor, core exit thermocouples, and a reactor coolant inventory tracking system is required for the operation of pressurized water reactor (PWR) facilities.

(1) ICCI Design

In response, the applicant described the ICC system proposed for Shearon Harris in the following letters from M. A. McDuffie (CP&L) to H. R. Denton (NRC):

- "Draft Safety Evaluation Report Responses," August 11, 1983
- "Responses to Requests for Additional Information," November 4, 1983
- "Draft Safety Evaluation Report - Core Performance Branch," December 6, 1983

The staff's review of the information in those letters follows.

The applicant has selected an ICCI system that consists of three subsystems: (1) subcooling margin monitor (SMM), (2) incore thermocouple system, and (3) reactor vessel level instrumentation system (RVLIS). The design uses a computer-based processing system (the emergency response facility information system, ERFIS) for primary display of incore exit thermocouple and margin of subcooling data.

Although the ERFIS is not Class 1E, it receives qualified pressure and temperature signals through an accessible isolator, and is powered from a highly reliable power source that is backed up by a battery. The ERFIS computer margin of subcooling data and the incore exit thermocouple temperature are displayed on the safety parameter display system (SPDS) cathode ray tube (CRT) that is on the main control board (MCB). The ERFIS has two redundant channels and is single failure proof. The Shearon Harris emergency operating procedures (EOPs) and functional restoration procedures (FRPs) will be based on the Westinghouse owners group emergency response guidelines and functional restoration guidelines. These procedures employ ICCI (RVLIS, the core exit thermocouples, and the subcooling data) and other post-accident monitoring capabilities (reactor coolant system pressure, reactor coolant pump status, and safety injection flow).

Thus, the Shearon Harris ICCI will be used in accordance with the emergency response guidelines developed by the Westinghouse owners group.

(2) Subcooling Margin Monitor

An SMM will be installed and operational before fuel load. The ERFIS processes ıe wide-range reactor coolant system (RCS) pressure indicators and the incore :it thermocouples, both of which span the necessary range to preclude the need r overlapping instrumentation. The ERFIS computes the margin of subcooling ta which is displayed, as well as the incore exit thermocouple temperature on ə SPDS CRT on the MCB.

case both primary channels fail to determine the margin of subcooling, the licant proposes to use the fully qualified wide-range RCS pressure indica-s, in conjunction with the fully qualified incore exit thermocouple tempera-əs and the American Society of Mechanical Engineers steam tables.

Core Exit Thermocouple System

core exit thermocouple system was designed to meet the criteria in Attach-1 to Item II.F.2.

ıocouples used for the core exit for each core quadrant (in conjunction core inlet temperature data) are sufficient to provide indication of radi-stribution of the coolant enthalpy (temperature) rise across representa-regions of the core.

rimary display includes the following:

\ spatially oriented core map indicating the temperature or temperature lifference across the core (at each thermocouple location) is displayed on he CRT.

selective reading of core exit temperature, which is consistent with parameters pertinent to operator actions in connection with plant-specific ICC procedures, will be continuous on demand.

Direct readout and hard copy capabilities are available for all thermo-couple temperatures. The range extends from 200°F to 2300°F. Hard copy will be provided by computer printout.

Trend capability showing the temperature-time history of representative core exit temperatures is available on demand.

Alarms are provided in the control room. The alarm setpoints will be consistent with the decision points iı the emergency operating procedures.

The CRT interface will be located in accordance with human factors design principles to facilitate rapid access to requested displays. CP&L provided its human factors methodology for the MCB in a letter to the staff dated June 1, 1983. This submittal identified the methodologies and human engineering specifications that apply to items such as ICCI that were not defined when the detailed control room design review was performed.

The backup display, which is in the control room, that has the capability for selective reading of each of the operable thermocouples. The range extends from 200°F to 2300°F.

The types and locations of displays and alarms will consider

- the use of this information by an operator during both normal and abnormal plant conditions
- integration into emergency procedures
- integration into operator training
- other alarms that are activated during an emergency and the need for prioritization of alarms

This system will be fully operational by fuel load.

(4) Reactor Vessel Level Measurement

Information utilized to give the operator an advance warning of the approach to ICC and to monitor the recovery from ICC, if it occurs, is obtained through a qualified instrumentation package. The information is obtained by the use of the RVLIS and incore exit thermocouples.

The Westinghouse RVLIS being installed at Shearon Harris represents the most recent Westinghouse design. It is a fully qualified, redundant system for monitoring water inventory in the reactor vessel. Each of the two channels includes differential pressure cells and transmitters for narrow- and wide-range monitoring over the full length of the vessel, with the reactor coolant pumps off (natural circulation) and on, respectively. Additionally, narrow-range monitoring is provided for each channel of the upper plenum during natural circulation. The microprocessor in each chanel utilizes these differential pressure signals in conjunction with other inputs (such as RCS pressure and temperature, loop resistance temperature detectors or incore thermocouples, and RVLIS reference leg temperature sensors) to compensate for density changes in the system legs and to provide direct water level readers for the operator.

Qualified incore thermocouples are used to determine core exit temperature. These 51 thermocouples (26 channel A, 25 channel B) are inputs to and are processed by the RVLIS microprocessors. Both RVLIS water level readings and incore exit thermocouple data will be data-linked to the ERFIS computer for primary display on the SPDS CRT on the MCB. The data link is supplied by an isolated non-Class 1E output from the qualified RVLIS microprocessors. Although ERFIS is not Class 1E, it is powered by a highly reliable source. The isolation device cabinets and ERFIS are readily accessible and adjacent to the main control room.

Additionally, qualified microprocessor outputs (RVLIS water level and thermocouple data) will be transmitted to dedicated redundant backup displays. These backup displays are alphanumeric and qualified (Class 1E), and are located in the control room. The primary and backup displays have a selective capability for providing RVLIS water level, thermocouple data, and temperature mapping functions. A technical description of the system is in the Westinghouse manual entitled "RVLIS-Summary Report, December 1980."

This system will be fully operational by fuel load.

(5) Staff Evaluation

On the basis of its review of the information submitted by the applicant, the staff finds that the ICC detection system is in conformance with these requirements of NUREG-0737, Item II.F.2 and II.F.2, Attachment 1. However, the staff will ensure that Technical Specifications relating to the final ICCI system are submitted and approved before fuel load.

In addition the Unit 1 license will be conditioned as follows:

1) ICCI will be installed and preoperational tests will be completed before fuel load. Startup tests and calibrations for which the core must be in place will be completed before operation above 5% of full power.

2) Before the plant exceeds 5% power, an implementation letter report must be provided for staff review.

) Before criticality, the modified emergency procedures that incorporate the generic Westinghouse RVLIS system for Shearon Harris must conform to generic EOP guidelines relating to the use of the RVLIS, or deviations must be identified and explained.

10 STEAM AND POWER CONVERSION SYSTEM

10.4 Other Features

10.4.9 Auxiliary Feedwater System

The staff and its contractor, Brookhaven National Laboratory (BNL), have reviewed the reliability analysis provided by the applicant relative to the unavailability of the AFW system to be able to provide the required amount of water to the steam generators in the event of loss of main feedwater (LMFW), loss of offsite power, and loss of all ac power. The event consisting of loss of main feedwater with offsite power available imposes a greater demand on the AFW system than does the event consisting of loss of main feedwater without offsite power. Therefore, this review considers only the case with offsite power available.

Table 10.1 compares the applicant's results with the BNL results. Both the applicant and BNL show satisfactory results when comparing the reliability of the AFW system when only one of the three AFW pumps (two motor-driven, one turbine-driven with the turbine-driven pump having twice the capacity of each motor-driven pump) is needed with the criterion of no more than 1×10^{-4} to 1×10^{-5} failures per demand for loss of main feedwater (LMFW) and loss of offsite power (LOOP). However, when two pumps are needed for an LMFW, the AFW system unreliability (4.6×10^{-4}) exceeds the criterion of 1×10^{-4}. For loss of all alternating current (LOAC), there is no criterion for numerical unavailability. The results of the study are utilized to ensure AFW availability independent of ac power.

Table 10.1 Unavailability of the Shearon Harris AFW system

	Applicant's Results	BNL Results	
Transient	Mission Success A*	Mission Success A*	Mission Success B**
LMFW	6.6E-6	9.2E-6	4.6E-4
LOOP	6.1E-5	4.9E-5	--
LOAC	1.9E-2	2.5E-2	--

*Mission Success A refers to the cases in which only one pump is necessary for success.

**Mission Success B refers to the case (LMFW only) in which flow from both motor-driven pumps (or the turbine-driven pump) is required.

The applicant stated that mitigation of LMFW with offsite power available required the AFW system to supply water to the steam generators at a greater rate than was required for the other two events (LOOP and LOAC). The applicant provided a certified AFW pump test curve to show that one motor-driven pump could supply more than the amount required to the steam generators for an LMFW, LOOP, and LOAC event. However, this did not take into account the recirculation line, which diverted part of the water back to the condensate storage tank. This left insufficient water for the steam generators in the event of an LMFW. Thus, LOOP and LOAC can be accommodated by one motor-driven pump, but LMFW (without offsite power) cannot if the recirculation line remains open. This accounts for the need for two pumps and the unavailability value of 4.6×10^{-4} for the LMFW event.

When asked about the ability of a motor-driven pump to deliver the flow rate required to mitigate the LMFW event, the applicant noted that if one of the two motor-driven pumps is unavailable because of loss of voltage, the recirculation line on the other pump (the operating pump) would close to allow sufficient water to the steam generators. The valve on each recirculation line may also be closed by an operator, either from the control room or from the auxiliary control panel, thus permitting a sufficient flow of water to the steam generator from one operating motor-driven AFW pump.

This information was relayed to BNL when it was used to perform a sensitivity study based on the results of the reliability analysis that assumed that only one motor-driven AFW pump is required to mitigate the LMFW event with offsite power available (resulting in the unavailability value of 9.2×10^{-6}). The original analysis was modified by multiplying the cutsets containing random failure unavailabilities of either or both motor-driven pumps by conservative factors. This conservatively calculated result showed an unavailability of approximately 2×10^{-5}. Therefore, the staff finds the unavailability of the Shearon Harris AFW system for the LMFW event also acceptable because the unavailability value is less than 1×10^{-4}.

12 RADIATION PROTECTION

12.5 Operational Radiation Protection Program

12.5.1 Organization

In Section 12.5.1 of the SER, the staff stated that the person to fill the position of Environmental and Radiation Control Manager had not been selected. The person now selected has qualifications that meet the provisions of RG 1.8. The staff considers this matter resolved.

13 CONDUCT OF OPERATIONS

13.3 Emergency Planning

13.3.1 Introduction

The staff's evaluation of the applicant's emergency preparedness plan is provided in SER Section 13.3. The deficiencies identified in that evaluation have been addressed by the applicant in (1) Revision 2 to the Shearon Harris Emergency Plan, February 1984, and (2) revisions to the Corporate Emergency Plan and Implementing Procedures, February 1984.

The revised sections of the plans and procedures were reviewed against (1) the appropriate planning standards in 10 CFR 50.47, (2) the requirements of Appendix E to 10 CFR 50, and (3) the specific guidance criteria of NUREG-0654/FEMA-REP-1, Revision 2, entitled "Criteria of Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," November 1980, which has been endorsed by RG 1.101 (Rev. 2).

Section 13.3.3 contains the conclusions of the staff, based on the previous review and this review.

13.3.2 Evaluation of the Emergency Plan

13.3.2.1 Assignment of Responsibility (Organization Control)

The SER identified four open issues. The applicant's response to these issues is as follows:

(1) The applicant has provided specific information listing the response organizations, the contact (and alternate), the location for the response, the response time, and the agent for initial notification in Table G.1-1 of Appendix G of the plan.

(2) The applicant has shown the interrelationships of the emergency response organizations (a) before the activation of the technical support center (TSC) and the emergency operations facility (EOF), (b) after the activation of the TSC, and (c) after the activation of both the TSC and EOF in Figures G-1, G-2, and G-3 of Appendix G of the plan.

(3) The applicant has described the duties and responsibilities of the Emergency Response Manager and members of the manager's staff in the revised plan. The EOF staff organization is shown in Figure 2.4-1 of the plan and described in detail in the Corporate Emergency Plan.

(4) The applicant has provided signed agreements with 17 offsite response and support organizations, including local fire, rescue, and medical organizations; county and state emergency management organizations; the Department of Energy; and the Institute of Nuclear Power Operations in Revision 2 of the plan.

The staff has reviewed and evaluated the additional information provided by the applicant and finds that the resolution of staff concerns on these four issues is acceptable.

13.3.2.2 Onsite Emergency Organization

The SER identified three open issues. The applicant's response to these issues is as follows:

(1) The applicant has clearly identified the conditions for transfer of authority from Shift Foreman (Site Emergency Coordinator-Control Room) to the Site Emergency Coordinator (TSC), and from the Site Emergency Coordinator (TSC) to the Emergency Response Manager (EOF) in Revision 2 of the plan. These conditions are

   (a) The facility (TSC or EOF) is ready to be activated and assume emergency functions.

   (b) The Site Emergency Coordinator (TSC) or the Emergency Response Manager (EOF) has received a briefing on the status of the emergency.

(2) The applicant has revised Tables 2.2-1 and 2.2-2 to correspond with Table B-1 of NUREG-0654 so that they specify the minimum onshift staffing available for emergencies and the capability for augmentation in 30 to 45 minutes with additional augmentation in 60 to 75 minutes. The variations in times of arrival are determined by weather conditions. The minimum staffing and the augmentation of the emergency staff follow the guidelines of NUREG-0654 and Supplement 1 of NUREG-0737 and are adequate.

(3) The applicant has shown the relationship and interfaces between the various onsite and offsite response and support agencies in the revised plan. These relationships are given in Table 4.0-1 and illustrated in Figures 2.2-1 and 2.4-1 and Figures G-1, G-2, and G-3 of Appendix G. The table shows the organizations with the primary responsibility and the secondary or support responsibility for each of 17 major emergency functions.

The staff finds that the applicant's response to the above three issues is satisfactory and that the staff's concerns have been fully resolved.

13.3.2.4 Emergency Classification System

The SER identified two open issues. The applicant's response to these issues is as follows:

(1) The applicant plans to submit the Plant Emergency Procedures by September 28, 1984. The staff will evaluate the Emergency Action Level (EAL) sets at that time.

(2) The revised plan clearly states the role of the judgment of the Shift Foreman (or Site Emergency Coordinator) in assessing the status of the plant, which is that he will declare any one of the four emergency classes where EALs have been exceeded, or in his judgment, the status of the plant warrants such a declaration. The staff considers this matter resolved.

13.3.2.7 Public Information

The SER identified one open issue. In response to this concern, the applicant plans to submit the public information brochure in June 1984 for staff review.

13.3.2.8 Emergency Facilities and Equipment

The SER identified three open issues. The applicant's response to these issues is as follows:

(1) The applicant has described three seismic monitoring systems for the Shearon Harris site in Section 3.9.2 of the revised plan. These consist of one system on site, a second system off site to provide remote reading of seismic activity at the Corporate Headquarters in Raleigh, and a third system consisting of two independent detectors with tape recorders located at points on site. The recorded signals can be played back either in the control room or at Corporate Headquarters.

(2) The applicant has stated in Section 3.4.1 of the revised plan that approximately 2650 ft^2 in the lunch room are available for use by approximately 200 people. Table 3.1-1 has been revised to include battery-powered lanterns and a Polaroid camera in the operational support center emergency supplies.

(3) The applicant has described the onsite laboratory facilities and specified the onsite laboratory as the central point for receipt and analysis of all onsite samples in Section 3.9.7 of the revised plan. The applicant also has described the mobile environmental monitoring laboratory, based at Shearon Harris Energy and Environmental Center (SHEEC), and designated the mobile laboratory as the central point for receipt and analysis of all offsite radiological samples.

The staff finds the applicant's response to these three issues acceptable and, therefore, considers them closed.

13.3.2.9 Accident Assessment

The SER identified four open issues. The applicant's response to these issues is as follows:

(1) The applicant has discussed the emergency response facilities information system (ERFIS) and the safety parameter display system (SPDS) in Section 3.9 of the revised plan. The ERFIS is a computer system (with backup units for critical hardware) that samples flows, pressures, temperatures, fluid levels, radiation levels, radiological monitoring system (RMS) equipment, and valve status. The SPDS is a software subsystem of the ERFIS. The applicant also has described the RMS with process radiological monitors and effluent radiological monitors, the postaccident sampling system (PASS), the meteorological monitoring system, and field monitoring capability. A detailed description of this equipment and resources is contained in the FSAR.

(2) The applicant has identified the meteorological data input to the ERFIS and the access to the ERFIS computer by the corporate meteorological

center in the revised plan. The corporate center meteorologists analyze national and local weather data and provide localized weather forecasts for the CP&L system or Shearon Harris as required. NRC and State agencies contact the corporate center for appropriately formatted and verified data, both current and forecast.

(3) The applicant has provided a discussion of the techniques used in dose projection by the ERFIS if any indicators are off scale or otherwise tagged as suspect or false. The computer data bank contains the radionuclide-mix assumptions based on the accident source terms of FSAR Table 15.0.9-1. These data are used as default values until actual sample data can be substituted.

(4) The applicant has provided a discussion of the radiation monitoring system in Section 3.9.3 of the revised plan. This system, described in detail in FSAR Sections 11 and 12, can provide the means for relating various measured parameters and activity measurements to the key isotopes listed in Table 3 of NUREG-0654.

Based on its evaluation of additional information provided by the applicant, the staff concludes that these issues are fully resolved.

13.3.2.10 Protective Response

The SER identified six open issues. The applicant's response to these issues is as follows:

(1) The applicant has provided a description of the plant public address (PA) system and the areas that are covered by this system, including the controlled area outside the protected area, in the revised plan. Security personnel with portable loudspeakers may be used to augment the PA system.

(2) The applicant has described the method by which continuous accountability will be maintained by the designated team leaders and managers in the revised plan. These methods are discussed in detail in Plant Emergency Procedure 0-PEP-382.

(3) The applicant has provided a summary of evacuation time estimates, condensed from the Evacuation Time Study received in January 1984, in Table G.8-1 of the plan. The staff is reviewing the study and will provide its evaluation in a future supplement to the SER.

(4) The applicant has shown the radiological monitoring sampling locations in Figure 3.9-1 of the plan. These locations are discussed in detail in the Plant Emergency Procedures.

(5) The applicant has shown the locations of public shelter areas in Figure G-4 of the plan. The exact location of the areas and other details are discussed in Parts 2.E, 3.E, 4.F, and 5.E of the state plan.

(6) The applicant has discussed the shelter protection provided by local housing in Section 4.5.2 of the plan.

On the basis of its review of the applicant's responses, the staff concludes that the applicant's protective response capability is acceptable subject to the staff's confirmation of the evacuation time study (item (3) above), which will be addressed in a future supplement to the SER.

13.3.2.11 Radiological Exposure Control

The SER identified three open issues. The applicant's response to these issues is as follows:

(1) The applicant has provided for the decontamination of supplies, instruments, and equipment according to plant procedures or their disposal as radwaste.

(2) The applicant has described the measures to control access to drinking water and food supplies on site in Section 4.6.3.8 of the plan. The potable water clear well will be manually isolated until samples can be analyzed. If the well water is contaminated, potable water will be shipped into the plant. Most of the food on site is packaged food in vending machines that can be disabled or isolated until samples are analyzed.

(3) The applicant has provided for the storage of decontamination supplies and clean protective clothing at personnel decontamination facilities located in the plant, in the administration building, and the Shearon Harris Energy and Environmental Center.

On the basis of its review of the additional information provided by the applicant on these three issues, the staff concludes that the radiological exposure control program is acceptable and, therefore, considers these issues closed.

13.3.2.13 Recovery and Reentry Planning and Postaccident Operations

The SER identified one open issue. In response to the staff's concern, the applicant has revised Section 6.4 of the plan to specify that notification of onsite and offsite organizations concerning the activation of the recovery organization will be initiated by the Emergency Response Manager and will follow plant emergency notification procedures. Notification procedures will also list the new positions of the recovery organization. The staff finds the applicant's response acceptable and considers this matter closed.

13.3.2.14 Exercises and Drills

The SER identified one open issue. In response to the staff's concern, the applicant has described the policy on exercises and drills in Section 5.3 of the revised plan. Detailed procedures for the conduct of exercises and drills are described in Corporate Emergency Plan Implementation Procedure CEPIP-18. The revised plan states that exercises will be conducted under various weather conditions and that some will be unannounced. It also provides for "free play" decisionmaking. The Plant Emergency Coordinator, with the Director of Emergency Preparedness, will determine corrective actions and will follow up on implementation. On the basis of its evaluation of Section 5.3 of the revised plan, the staff finds the applicant's policy acceptable.

13.3.2.15 Radiological Emergency Response Training

The SER identified two open issues. The applicant's response to these issues is as follows:

(1) The applicant has provided details of the training to be provided to specific categories of emergency response personnel in the revised plan. The plan specifies that initial training and annual retraining will be provided. It also provides assistance with the training of offsite emergency response personnel according to the radiological emergency plan, and supplemental training as related to Shearon Harris. The Emergency Plan Training Program is described in detail in Plant Emergency Procedure PEP-403 and in the Plant Emergency Plan Training Program Lesson Plans.

(2) The applicant has specifically addressed the training of Damage Control Team personnel in Section 5.2.3.7 of the revised plan.

On the basis of its evaluation of the applicant's response, the staff finds the applicant's training program acceptable.

13.3.3 Conclusions

On the basis of its review of the Shearon Harris Nuclear Power Plant Emergency Plan in the SER and the review of the plan revisions, the staff concludes that, on satisfactory completion of those items identified in Sections 13.3.2.4 (1) and 13.3.2.7 of this report, the Emergency Plan will provide an adequate planning basis for an acceptable state of emergency preparedness.

After the staff has reviewed the findings and determinations made by the Federal Emergency Management Agency on the adequacy of state and local emergency response plans and after it has reviewed the revisions to the applicant's Emergency Plan, the staff will provide, in a supplement to the SER, its overall conclusions as to whether the state of onsite and offsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.

13.6 Physical Security

13.6.3 Proposed License Condition

The following proposed license condition was inadvertently omitted from the SER:

> The licensee shall fully implement and maintain in effect all provisions of the Commission approved physical security, guard training and qualification and safeguards contingency plans, including amendments made pursuant to the authority of 10 CFR 50.54(p). The approved plans, which contain Safeguards Information as described in 10 CFR 73.21, are entitled, "Shearon Harris Nuclear Power Plant Security Plan," Revision 1, dated July 5, 1983; "Shearon Harris Nuclear Power Plant Safeguards Contingency Plan," Revision 1, dated July 5, 1983; "Shearon Harris Nuclear Power Plant Security Personnel Training and Qualification Plan," Revision 1, dated July 5, 1983.

18 HUMAN FACTORS ENGINEERING

As discussed in the SER, Supplement 1 to NUREG-0737 requires the applicant to complete a detailed control room design review (DCRDR) before an OL can be issued for Shearon Harris. The applicant departed from the NRC recommendation to provide a program plan early in their review process, followed by a summary report at the completion of the DCRDR.

The applicant submitted a combined program plan and summary report by letter dated December 7, 1982. These were reviewed against the requirements of Supplement 1 to NUREG-0737. Consultants from Lawrence Livermore National Laboratory (LLNL) assisted the staff in the review and prepared the program plan comments and the in-progress audit report. A human factors engineering in-progress audit of the control room design review was performed at the site August 15 through 19, 1983.

18.1 General

The applicant's submittal included elements of a program plan that was developed and implemented in January 1981 and revised in September 1981 before the DCRDR requirements of NUREG-0737 Supplement 1 and the guidelines of NUREG-0700 were issued. The submittal included a description of tasks that have been completed and a summary report of the control room design review that noted the findings and corrective actions taken to date.

Before, during, and after the audit the applicant provided a number of supporting documents. These included

- sample human engineering discrepancy (HED) sheets
- a sample HED status information report
- a sample HED listing
- the Shearon Harris Unit 1 control room design evaluation records file index
- a representative set of interim main control boards (MCB) panel drawings
- the applicant's letter LAP-83-426, dated September 27, 1983, which contained additional information requested by the NRC audit team

Available at the audit site were the control board simulator, which is significantly different from the MCB, which is still under construction. The applicant's operating and engineering personnel and the applicant's human factors consultant from the Essex Corporation were available on a daily basis during the audit.

The audit included briefings, discussions, document reviews, and a brief review of the simulator and the incomplete MCB. The emphasis was on evaluating the organization and processes of the applicant's DCRDR.

18.2 Planning Phase

The applicant's submittal contains elements of a program plan that meets most of the basic requirements of NUREG-0737, Supplement 1, in addition to describing the related activities completed at the time it was submitted. However, specific areas of the work should be described in greater detail and additional documentation should be made available for audit. The additional documentation should clarify methodology, task procedures, and objectives to ensure that the applicant understands the requirements and processes of a DCRDR.

18.3 Qualifications and Structure of the DCRDR Team

The applicant established a multidisciplinary review team for the DCRDR that included applicant and Essex Corporation personnel, representing a cross section of the required disciplines. The qualifications of the applicant's review team members are in Appendix A of the applicant's supplemental summary report dated September 27, 1983. A description of the review team composition was included in Attachment 1 and Appendix A of the applicant's letter of June 1, 1983.

The applicant's review team was divided into three groups

(1) human factors evaluation
(2) human factors/operations support
(3) project management/nuclear operations/plant engineering and design group

The Human Factors Evaluation Group was composed of six intermediate- and junior-level human factors specialists from the Essex Corporation. The group's responsibility was to collect and reduce data and perform preliminary data analysis. However, except for the group supervisor, the extent of each group member's participation is not clear. The group supervisor's responsibility was to coordinate the evaluation activities, interface with the applicant's project manager, and represent the human factors position at HED review meetings.

The Human Factors/Operations Support Group was composed of Essex Corporation home office personnel. This group consisted of five senior-level human factors specialists, a nuclear engineer, two reactor operators, three junior-level human factors specialists, a procedure specialist, and a photographer. The overall responsibility of this group was to review: in-depth analysis; discrepancy definition; recommendations for resolution of discrepancies; data collection support; and operational and engineering analysis. The group leader was not identified, and the extent of each member's participation is unclear.

The Project Management/Nuclear Operations/Plant Engineering and Design Group was composed of applicant personnel from various disciplines, representing the Nuclear Plant Engineering Department, System Planning and Coordination Department, Nuclear Operations Department, and Nuclear Operations Department Training. The group leader coordinated the activities of Essex, the applicant, EBASCO (the architect/engineer), and Westinghouse (the nuclear steam supply system vendor). The group leader also reviewed the overall progress of the DCRDR. EBASCO and Westinghouse personnel provided any required review and comments on design philosophies, discrepancy analysis, etc.

On the basis of its review, the staff finds that the review team, as described above, satisfies Supplement 1 to NUREG-0737.

18.4 Review Phase

18.4.1 System Function and Task Analysis

The applicant performed its task analysis, using NUREG-1580 guidelines, which differ from the NUREG-0700 guidelines. (When NUREG-0700 was issued, the applicant's analysis was already in progress.) The task analysis guidelines in NUREG-1580 are based on analyzing procedures to document operator actions, information requirements, and controls and displays used in executing procedures. The NUREG-0700 task analysis guidelines are based on a systematic top-down function/task analysis to determine information and control capability requirements that can be objectively compared to the actual instrumentation and controls available in the control room.

In response to an NRC audit team request, the applicant stated in a letter dated September 27, 1983, that a generic top-down task analysis--based on the identification of event sequences, plant systems, and operator functions and tasks as recommended in NUREG-0700--was performed on the High Pressure Basic Version of the Westinghouse Owners Group (WOG) emergency response guidelines (ERGs). The staff review of the WOG guidelines indicates that they constitute a satisfactory start in identifying and describing tasks, but they are generic and not plant specific. The guidelines do not identify information and control requirements in sufficient detail to allow a comparison to the control room instrumentation and control room inventory.

There was no substantiating documentation on the details or completeness of the methodology used in the task analysis.

On the basis of its review, the staff finds that the applicant's task analysis process described in the summary report may not identify instrument and control requirements independent of existing instrumentation in the control room and may not ensure that the man-machine interface is complete. Thus, the staff finds that the applicant must provide a more complete rationale and justification of the method for conducting task analysis for the staff to determine whether the requirements of Supplement 1 to NUREG-0737 have been satisfied.

18.4.2 Control Room Inventory

During the in-progress audit, the applicant stated that the EBASCO panel component list had been substituted for a control room inventory. This list was used to check the existence of controls and displays on the MCB panel drawings to confirm the expected panel content versus need. (The instrument and control needs were determined by EBASCO and the applicant utilizing their expertise and operating experience.) Because of the list used and the fact that the WOG generic function and task analysis does not provide a detailed list of control and display characteristics, an accurate comparison of control and display requirements with the inventory is not possible.

18.4.3 Control Room Survey

The Shearon Harris control room survey was carried out through reviews of plant design documents, vendor documentation, and layout drawings. The Shearon Harris simulator was used where applicable. A to-scale paper mockup was also used as a basis for identifying HEDs and as a starting point from which to relocate controls and diaplays.

The control room survey process described by the applicant in the summary re- ɔort and during the onsite audit satisfies the requirement of Supplement 1 to łUREG-0737.

8.5 Assessment and Implementation Phase

3.5.1 Assessment of HEDs

ıe applicant's review team did not perform a formal, documented assessment of :Ds, as required by NUREG-0737, Supplement 1. Instead, the assessment was e of several, parallel, iterative processes executed during what the appli- nt called the redesign of the MCB.

ȝ initial HED identification process was completed on each panel in the con- ɔl room. HEDs were screened by the onsite human factors consultant in con- ıction with the applicant's review Project Manager. HEDs that required cor- ·tion were then discussed (assessed) in batches in meetings attended by the licant's review team (which included the human factors consultant, the licant's review Project Manager, applicant operating and engineering person- , and EBASCO and Westinghouse personnel, as appropriate). The applicant's ject Manager and the Essex human factors consultant attended all meetings. ȝr members of the applicant's review team were present when their expertise needed.

Many redesign review meetings were held during which the control room was repeatedly and iteratively redesigned. The applicant stated that this process included discussion of many alternate solutions before the final solution to correct each HED was selected.

The applicant's policy was to correct every HED by designing it out of the system, with the objective of achieving an HED-free board. If successful, the execution of this philosophy obviates the necessity to formally assess each HED for importance, potential safety consequences, cumulative impact of minor HEDs, determination of priorities, and setting implementation schedules. However, for those instances in which an HED with safety significance cannot be designed out of the system, and a decision is made by the applicant not to correct the HED or to only partially correct it, the applicant must provide justification for the action taken.

The staff finds that the assessment methodology used by the applicant meets the requirements of Supplement 1 to NUREG-0737.

18.5.2 Selection of Design Improvements

Alternate design improvements for each HED were considered during the applicant's redesign review, which was done at the same time as the redesign of the MCB. The review team used NUREG-1580 and human engineering requirements specifications (HERS) developed for the applicant by consultants as the bases for the HED resolutions. As described in the applicant's summary report, the redesign effort appears to have been similar to the effort involved in the determination of a new design.

The applicant stated that the MCB redesign review resulted in the removal of about 200 controls and the relocation of a significant number of unnecessary items from front-to-rear panels. EBASCO and Westinghouse concurred with these changes. Appendix D of the applicant's report "Recommended Control Room Equipment Arrangement" describes how the recommended panel arrangements were determined; factors considered included desirability of equipment location, readability of displays and labels from various MCB locations, visual angles, adverse effect on the operator's short-term memory because of distances operators had to move between equipment, discrimination problems caused by viewing distances, etc. In addition, a quarter-scale mockup was used to study and confirm rearrangement of the control board instrumentation. The mockup was also used to verify and confirm labeling, demarcation, and annunciator arrangement. The redesign effort included ongoing discussions with operators to verify that the MCB redesign details and HED corrective actions were compatible with operational needs and left no discrepancies uncorrected. This included verification that any equipment found missing would be installed before the redesign was considered complete.

On the basis of its review, the staff finds that the methods used for selection of design improvements to correct HEDs described in the applicant's summary report and during the onsite audit satisfy the requirements of Supplement 1 to NUREG-0737. However, the applicant must submit for staff evaluation information concerning any HEDs identified during the verification process, their resolution, and an acceptable time schedule for implementing corrective actions.

18.5.3 Coordination of Control Room Improvements with Other Programs

The Shearon Harris emergency operating procedures (EOPs) are being developed using modified procedures from the H. B. Robinson plant and the Shearon Harris simulator. The WOG ERGs will be considered in the development of the Shearon Harris EOPs, and the applicant has stated that the EOPs will be subject to task analysis before they are implemented. The results of the task analysis will be integrated back into the EOPs and into the DCRDR. The applicant then plans to complete the system function and task analysis and verify and validate the EOPs after they are implemented and the MCB is fully functional. The applicant stated that any subsequent MCB modifications will be made following the same processes used in the MCB redesign.

Although the applicant described this coordination effort, the applicant did not provide information on coordinating the DCRDR activity with the SPDS, RG 1.97, and the emergency response facilities.

18.5.4 Proposed Schedule for Implementing Design Changes

The applicant's report does not include information on implementation of design changes because the applicant's policy of correcting all HEDs during the redesign process before operation obviates the need for implementing design changes.

In letter LAP-83-156 (dated June 3, 1983) the applicant states that the verification and validation of the control room redesign after the WOG task analysis and generic guidelines are available, and the completion of the Shearon Harris EOPs verification and validation will reveal few, if any, MCB human factors concerns. If any HEDs are identified, they will be corrected via the continuation of the applicant's redesign corrective action policy.

18.5.5 Justification for HEDs with Safety Significance that Are Left Uncorrected or Partially Corrected

For the reasons discussed above, the applicant's policy of correcting all HEDs identified during redesign of the MCB essentially satisfies this requirement of Supplement 1 to NUREG-0737.

18.6 Conclusions

The applicant is continuing its DCRDR and improving the redesigned MCB. On the basis of its review of the applicant's reports and the results of the in-progress onsite audit, the staff has determined that the applicant is conducting a DCRDR that will substantially meet the guideines of NUREG-0700 and the requirements of Supplement 1 to NUREG-0737, except that the applicant must

(1) Provide a detailed description of the Shearon Harris system function and task analysis.

(2) Describe the process used to compare display and control requirements as determined by the function and task analysis, with the control room inventory.

(3) Describe the process to be used to verify that the corrective actions achieved the desired improvement without introducing new HEDs into the control room.

(4) Provide a supplementary summary report that addresses items not included in the summary report because they are not ready for review. These items are identified below.

The NRC Audit Team reviewed the partially completed MCB and independently verified many of the HEDs identified by the applicant; however, the applicant's HEDs and those found by the applicant were not matched on a one-to-one basis. For each of the HEDs listed in part A of the in-progress audit report, the applicant should give the staff the status and proposed resolutions, as well as a schedule for implementing corrective action. This should be submitted at least 6 months before fuel load. Any HEDs found must be corrected before fuel load or on a schedule approved by the staff.

The items listed below were not included in the applicant's report because they were not ready for review. For the DCRDR to be complete, the applicant must provide the results of the evaluation of the following items so the staff can determine whether the requirements of Supplement 1 to NUREG-0737 have been satisfied:

(1) workspace
(2) communications
(3) remote shutdown panel
(4) recorder panel
(5) CRTs
(6) process computer and peripherals
(7) annunciator systems

The applicant should submit the additional information required and results of the evaluation of the items discussed above at least 6 months before the Shearon Harris licensing date so the staff has time to evaluate the submittal.

19 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

During its 285th meeting, January 12-14, 1984, the Advisory Committee on Reactor Safeguards (ACRS) reviewed the application of Carolina Power and Light Company and the North Carolina Eastern Municipal Power Agency for a license to operate the Shearon Harris Nuclear Power Plant, Unit 1. This application was also considered at the subcommittee meeting held on January 3-4, 1984, in Apex, North Carolina, at which time members of the subcommittee toured the plant. Transcripts of each of these meetings are available at the Wake County Public Library, 104 Fayetteville Street, Raleigh, NC 27601. Copies of these transcripts are also available for review at the NRC Public Document Room at 1717 H Street, NW, Washington, DC. The ACRS issued a letter report January 16, 1984, a copy of which is attached as Appendix G to this report.

The ACRS concluded that if due regard is given to items identified in its January 16, 1984, letter report (which are described below), and subject to resolution of the outstanding and confirmatory issues identified in the Shearon Harris SER and satisfactory completion of construction, staffing and preoperational testing, there is reasonable assurance that the Shearon Harris Nuclear Power Plant can be operated at core power levels up to 2775 MWt without undue risk to the health and safety of the public.

These items are

(1) <u>Control Room Emergency Air Recirculation System</u>

The Committee expressed its desire to be kept informed about the applicant's operational test of the control room emergency air recirculation system, including control room habitability during the recirculation mode. The applicant has scheduled this test for January 1985. The test procedure and test results will be made available to the ACRS.

(2) <u>Westinghouse D-4 Steam Generators</u>

The Committee asked that it be kept informed about the operating experience of the Westinghouse D-4 steam generators that are being used at Shearon Harris, Unit 1 and other nuclear facilities. The ACRS will be kept informed of the operating experience with these steam generators. Commanche Peak, Unit 1 is the lead plant with D-4 steam generators. An inspection of the steam generators at Commanche Peak, Unit 1 will be performed before it exceeds 1 year of full-power operation. The results of the inspection will be made available to the ACRS.

(3) <u>Systematic Assessment of Licensee Performance (SALP)</u>

The Committee requested that the two scheduled SALP reports should be provided to the ACRS before full-power operation to support the reported improvement in the applicant's management. The staff will comply with this request.

(4) Allegations

The Committee received a letter from a member of the public that makes several allegations concerning quality assurance and other issues and asked that the staff investigate the allegations and provide a written report to the Committee. The staff is preparing a response to the allegations that will be issued in the near future.

(5) Seismic Design

The Committee recommended that specific attention be given to assurance of adequate seismic capability of the emergency ac power supplies, the dc power supplies and small components such as actuators and instrument lines that are important to the accomplishment of safe shutdown and decay heat removal. The committee has also requested that specific attention be given to the adequacy of clearances between adjacent buildings. The staff will, as part of its seismic audit, pay specific attention to the items above as recommended by the ACRS. The staff will also address the adequacy of the clearances between adjacent buildings and report its findings in a future supplement to the SER.

(6) Essential Chilled Water System

As requested by the Committee, the staff will provide a detailed discussion of the essential chilled water system in a future supplement to the SER.

APPENDIX A

CONTINUATION OF CHRONOLOGY OF NRC STAFF
RADIOLOGICAL REVIEW OF
SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

November 22, 1983	Letter from applicant transmitting response to draft SER open item regarding service water sampling.
November 23, 1983	Letter from applicant transmitting responses to draft SER open items.
November 28, 1983	Letter from applicant transmitting revised response to draft SER open item 172.
December 1, 1983	Letter from applicant forwarding support information for response to draft SER open item 275.
December 2, 1983	Letter from applicant transmitting response to request for additional information.
December 2, 1983	Issuance of Generic Letter 83-32, NRC Staff Recommendations Regarding Operator Action for Reactor Trip and ATWS.
December 6, 1983	Letter from applicant forwarding revised response to draft SER open item 31.
December 9, 1983	Letter to applicant regarding facility staffing survey.
December 9, 1983	Letter from Transamerica Delaval regarding diesel generators installed in nuclear power plants.
December 13, 1983	Issuance of Safety Evaluation Report.
December 15, 1983	Letter from Transamerica Delaval regarding users' group meeting held November 30, 1983.
December 16, 1983	Letter from applicant forwarding response to facility staffing survey.
December 19, 1983	Issuance of Generic Letter 83-43 -- Reporting Requirements of 10 CFR Part 50, Sections 50.72 and 50.73, and Standard Technical Specifications.
December 19, 1983	Issuance of Generic Letter 83-42 -- Clarification to Generic Letter 81-07 Regarding Response to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

December 20, 1983	Issuance of Generic Letter 83-44 -- Availability of NUREG-1021, "Operator Licensing Examiner Standards."
December 21, 1983	Letter from applicant counsel to ASLB advising of Unit 2 cancellation.
December 29, 1983	Letter from applicant forwarding "Evacuation Time Estimates for Plume Exposure Pathway Emergency Planning Zone."
January 3-4, 1984	ACRS Subcommittee meeting with staff and applicant.
January 5, 1984	Issuance of Generic Letter 84-01 -- NRC Use of the Terms, "Important to Safety" and "Safety Related."
January 5, 1984	Board Notification 84-004 -- Environmental Qualification Briefing of Chairman by Sandia.
January 6, 1984	Issuance of Generic Letter 84-02 -- Notice of Meeting Regarding Facility Staffing.
January 10, 1984	Letter from applicant transmitting revised "Management Capability Report."
January 13, 1984	Issuance of Generic Letter 84-03 -- Availability of NUREG-0933 on Generic Safety Issues.
January 16, 1984	Letter from Advisory Committee on Reactor Safeguards.
January 17, 1984	Letter from applicant transmitting Amendment No. 11 to FSAR.
January 19, 1984	Meeting with applicant to discuss its design of the alternate shutdown system in the event control room is unavailable.
January 23, 1984	Letter from applicant forwarding "Quarterly Data Report -- Seismic Monitoring Program, October - December 1983."
January 24, 1984	Letter from applicant to IE regarding radiological aerial survey.
February 1, 1984	Issuance of Generic Letter 84-04 -- Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops.
February 6, 1984	Letter to applicant requesting additional information regarding Transamerica Delaval emergency diesel generators.
February 7, 1984	Meeting with applicant to discuss functional capability of Class 1 auxiliary piping systems.

February 7, 1984	Letter from applicant transmitting additional information on minimum wall thickness of Class 1, 2, and 3 piping.
February 13, 1984	Board Notification 84-020 -- Report of Meeting of Representatives of the Transamerica Delaval, Inc. (TDI) Emergency Diesel Generators Owners' Group.
February 14, 1984	Letter to applicant regarding deletion of home telephone numbers, unlisted utility numbers, etc. from emergency plans.
February 16, 1984	Meeting with applicant to discuss auxiliary piping systems.
February 20, 1984	Letter from applicant forwarding response to request for additional information from Materials Engineering Branch on low pressure turbine rotors.
February 27, 1984	Letter from Long Island Lighting Company forwarding "Investigation of Types AF & AE Piston Skirts" related to diesel review.
February 28, 1984	Letter from Long Island Lighting Company forwarding task descriptions for 16 known problems on diesel generators.
March 2, 1984	Letter from Long Island Lighting Company forwarding TDI Diesel Generator Owners' Group Program Plan.
March 6, 1984	Letter from applicant forwarding shift experience information.
March 7, 1984	Letter from applicant forwarding information on fuel handling building retaining wall design.
March 7, 1984	Letter from applicant transmitting preservice inspection program implementing procedure and flow diagrams.
March 8, 1984	Letter from applicant transmitting Revision 2 to Emergency Plan.
March 12, 1984	Letter from Long Island Lighting Company transmitting TDI Diesel Generator Owners' Group Design Review Report on Connecting Rod Bearing Shells for Shoreham and Grand Gulf.
March 12, 1984	Letter from applicant transmitting "North Carolina Emergency Response Plan."
March 13, 1984	Letter from Long Island Lighting Company transmitting TDI Diesel Generator Rocker Arm Capscrew Stress Analysis Report.

March 16, 1984	Letter from Transamerica Delaval responding to requests for information on R-4 series engines.
March 16, 1984	Letter from applicant requesting amendment to Construction Permit No. CPPR-158 to extend construction completion date to March 1, 1986.
March 21, 1984	Meeting with applicant to discuss equipment qualification program, including both environmental and seismic considerations.
March 22, 1984	Letter from Transamerica Delaval forwarding copies of test results on first engine produced in each model line.
March 22, 1984	Letter from applicant forwarding information on monitoring of service water system.
March 23, 1984	Letter from Long Island Lighting Company transmitting TDI Diesel Generator Air Start Valve Capscrew Dimensional and Stress Analysis Report.
March 29, 1984	Letter from applicant transmitting Amendment No. 12 to FSAR.
March 30, 1984	Letter from Long Island Lighting Company transmitting TDI Diesel Generator Cylinder Head Stud Stress Analysis Report.
April 4, 1984	Meeting with applicant to discuss design of the retaining wall with the staff and the applicant.
April 4, 1984	Board Notification 84-072 -- Transamerica Delaval, Inc. (TDI) Owners Group/NRC Meeting Transcript and Additional TDI Owners Group Information Submitted.
April 9, 1984	Letter from Long Island Lighting Company providing additional information concerning Owners Group Report on Types AF and AE Piston Skirts.
April 9, 1984	Letter from applicant forwarding revised response to Draft SER Open Item 136 concerning loss of component cooling water.
April 10, 1984	Letter from applicant regarding the requested construction permit extension.
April 11, 1984	Letter to J. P. McGaughy, Mississippi Power & Light, regarding preliminary assessment of two reports submitted to the NRC by the Transamerica Delaval, Inc. (TDI) Owners Group.

April 11, 1984	Letter to applicant forwarding request for additional information.
April 12, 1984	Letter from Long Island Lighting Company forwarding fatigue data for modular cast iron.
April 13, 1984	Letter from applicant forwarding seismic report for first quarter of 1984.
April 13, 1984	Letter from Long Island Lighting Company forwarding TDI Diesel Generator Supplement to Cylinder Head Stud Stress Analysis and Supplement to the Air Start Valve Capscrew Dimension and Stress Analysis.
April 13, 1984	Letter from Long Island Lighting Company forwarding calculations related to Rocker Arm Capscrew and Cylinder Head Stud Reports.
April 16, 1984	Letter from Long Island Lighting Company forwarding TDI Diesel Generator Report on Engine Driven Jacket Water Pump Design Review.
April 18, 1984	Letter to applicant forwarding request for additional information.
April 19, 1984	Letter from Long Island Lighting Company forwarding copies of TDI Diesel Generator Report on Push Rods.

APPENDIX B

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APPENDIX D

ACRONYMS AND INITIALISMS

AFW	auxiliary feedwater
BOP	balance of plant
CRT	cathode ray tube
EOP	emergency operating procedures
ERFIS	emergency response factors information system
ERG	emergency response guidelines
ESCWS	essential services chilled water system
FRP	functional restoration procedures
FSAR	Final Safety Analysis Report
ICC	inadequate core cooling
ICCI	inadequate core cooling instrumentation
LMFW	loss of main feedwater
LOAC	loss of alternating current
LOOP	loss of offsite power
MCB	main control board
NSSS	nuclear steam supply system
PWR	pressurized water reactor
RVLIS	reactor vessel level instrumentation system
SER	safety evaluation report
SMM	subcooling margin monitor
SPDS	safety parameter display system
WOG	Westinghouse Owners Group
WPBCWS	waste processing building cooling water system

APPENDIX E

PRINCIPAL STAFF CONTRIBUTORS

This supplement is a product of the NRC staff. The staff members listed below were principal contributors to this report.

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Marthe E. Harwell	Section Chief	Editorial Section
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APPENDIX G

ACRS REPORT

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

January 16, 1984

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE SHEARON HARRIS NUCLEAR POWER PLANT

During its 285th meeting, January 12-14, 1984, the Advisory Committee on Reactor Safeguards reviewed the application of Carolina Power & Light Company (CP&L) and the North Carolina Eastern Municipal Power Agency (the Applicants) for an operating license for the Shearon Harris Nuclear Power Plant. The Shearon Harris Nuclear Power Plant will be operated by CP&L which also operates three other nuclear units. The project was considered during an ACRS Subcommittee meeting in Apex, North Carolina on January 3-4, 1984. Members of the Subcommittee toured the facility on January 3, 1984. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicants, Westinghouse Electric Corporation, Ebasco Services, Inc., the NRC Staff, and a member of the public. The Committee also had the benefit of the documents referenced. The Committee commented on the application for a permit to construct the Shearon Harris Plant in reports dated March 8, 1972, January 17, 1973, and August 19, 1977. On October 11, 1977 the Committee provided a response to an inquiry regarding the resolution of ACRS Generic Items related to the Shearon Harris Nuclear Power Plant.

The Shearon Harris Nuclear Power Plant is located in Wake County, North Carolina, approximately 16 miles southwest of the nearest boundary of Raleigh, North Carolina. Originally the Shearon Harris Nuclear Power Plant was to comprise four units. However, only Unit 1 will be completed, with an estimated fuel load date of June 1985. Units 3 and 4 were cancelled on December 18, 1981 and Unit 2 was cancelled on December 21, 1983.

The Shearon Harris Nuclear Power Plant uses a three-loop Westinghouse nuclear steam supply system with a rated core power of 2775 MWt. The containment is a large, dry, reinforced concrete structure.

During the Committee's consideration of this plant, the control room design was reviewed. The Applicants informed us that they intend to perform an operational test of the control room emergency air recirculation system. As a part of this exercise, control room habitability during the recirculation mode will be evaluated. We wish to be kept informed.

The Shearon Harris Nuclear Power Plant uses Westinghouse D-4 steam generators. Steam generators of this design have experienced tube degradation related to flow-induced vibrations in the preheater region. Internal modifications have been developed by Westinghouse which include expanding some steam generator tubes and directing some of the main feedwater flow through the auxiliary feedwater nozzle. We expect to be kept informed regarding the operating experience of these steam generators.

The NRC Staff has previously identified management deficiences in CP&L's nuclear program. These deficiencies are enumerated in the report (May 1983) of the most recent Systematic Assessment of Licensee Performance (SALP) conducted by the NRC Staff to assess CP&L's nuclear operations for the period January 1982 - January 1983. CP&L has taken measures to improve management function and capability. These include restructuring of the corporate organization which will eventually result in a consolidation of CP&L's nuclear organization under one senior manager. The restructuring also provides for a corporate level executive to be located onsite, as a member of involved site management, to ensure greater access to resources and to enhance the ability to initiate new programs from the site. These efforts are expected to correct the past deficiencies. Members of the Region II Staff reported orally during the meeting that significant improvement in performance has been observed since the last SALP inspection. The Committee believes that written evidence of an improvement in CP&L's nuclear operations, which could, for example, be reported in the two scheduled SALP reviews prior to fuel load should be available prior to full power operation. We wish to be kept informed.

Subsequent to the meeting with the Applicants, we have received a letter from a member of the public which makes several allegations concerning quality assurance and other issues. We request that the NRC Staff investigate these allegations and provide a written report to the Committee.

The ACRS has on several occasions recommended that evaluations be made of the capability of light water nuclear power plants to be shut down safely in the event of an earthquake of greater severity and lower likelihood than the safe shutdown earthquake. In a letter dated January 11, 1983, the ACRS made recommendations concerning a possible broad approach to deal generically with the question of seismic margins. In the meantime, for the Shearon Harris Nuclear Power Plant, we recommend that, in addition to items already considered, specific attention be given to assurance of adequate seismic capability of the emergency AC power supplies, the DC power supplies, and small components such as actuators and instrument lines that are important to the accomplishment of safe shutdown and decay heat removal. We suggest also that specific attention be given to the adequacy of clearances between adjacent buildings.

During this review there was discussion of the reliability and the fracture resistance of the chilled water system. The Applicants and the NRC Staff reported orally that the system is satisfactory in these respects. The ACRS would like to receive a detailed discussion of the chilled water system in a supplement to the Safety Evaluation Report.

One of the confirmatory issues concerning this application is "turbine missiles." Because of the nonoptimum orientation of the turbine relative to vital components in this plant, we recommend that a structured test program for evaluating overspeed protection of the turbine be prepared and submitted to the NRC Staff for review and approval before full power operation.

A number of items have been identified by the NRC Staff as Outstanding Issues. There is also a set of Confirmatory Issues that awaits additional documentation. We found no reason to believe that any of these issues will be especially difficult to resolve. We recommend that they be resolved in a manner satisfactory to the NRC Staff.

The ACRS believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that the Shearon Harris Nuclear Power Plant can be operated at core power levels up to 2775 MWt without undue risk to the health and safety of the public.

Sincerely,

Jesse C. Ebersole

Jesse C. Ebersole
Chairman

References:

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2. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Shearon Harris Nuclear Power Plant, Units 1 and 2," USNRC Report NUREG-1038, dated November 1983
3. Letter from Wells Eddleman, Intervenor, Subject: Comments on the Shearon Harris Nuclear Power Plant to the Advisory Committee on Reactor Safeguards, dated January 13, 1984

to
The Advisory Committee on Reactor Safeguards
1.13.84

by Wells Eddleman, intervenor

Many things can compromise nuclear plant safety. One is plant or corporate management. Another is, in Mark Twain's words, "what you know, that ain't so". In 1979 (construction permit remand hearing on management capability, NRC Docket 50-400), the NRC staff "knew" that although CP&L had had problems at its Brunswick plant, e.g. high staff turnover, lack of supervisory attention, so many problems that they couldn't all be given attention, large numbers of LERs and long times to repair, things were going to be OK from now on because CP&L had straightened out. Although the Atomic Safety and Licensing Board in that remand expressed some doubts about some CP&L testimony and actions, it bought that judgment overall.

From 1979-83, Brunswick has continued to have high employee turnover, numerous problems, large numbers of LERs, long times to repair, etc. In addition, numerous meltdown precursors at Brunswick were identified including the loss of 3 out of 4 RHR heat exchangers due to biofouling in 1981. Most recently, CP&L refused to shut Brunswick down to check for pipe cracks in the summer of 1983. On shutdown, numerous cracks were found up to 18" in length, and overlay welds (a "band-aid") were used to put a unit back into service.

The better operating record CP&L claims for Brunswick recently does include the longest run without either unit shutting down, in the entire history of the plant: 6 weeks in fall 1983. Brunswick 1 and 2 have respectively the 2d worst and worst design electrical rating capacity factors (DER CF) of any 2 BWRs in the USA, below even that of Browns Ferry 1 which was disabled by fire in 1975.

CP&L has lots of charts and lists about management, but one test is results, and their results are poor enough that some explanation should be sought beyond NRC Staff's reassurances. One indication that's fairly obvious is that between the ACRS subcommittee meetings Jan 3-4 and the Jan 12 presentation, the CP&L presentations were not updated directly. Statistics on total person years of experience which the Subcommittee folk had called "meaningless" were still included. You can't expect the Company to do other than try to look good, but safety requires some deeper examination. CMP, for example, who audited CP&L management and gave a rather favorable report, had adopted "unusually effective approaches" to the construction of its Seabrook plant. That plant is now recognized as a model of waste in construction, using nearly 8000 workers to try to build a 2-unit plant. The CMP auditors relied mostly on CP&L-supplied statistics in auditing CP&L, and PSNH may have fooled them about Seabrook. As a Carolina consumer said, the CMP report shows CP&L "is well managed in all areas except results". The CMP report also fails to assign responsibility for CP&L's repeated foulups at Brunswick.

Brunswick was rushed on-line in 1974 for financial reasons. J.A. Jones, a senior CP&L executive in 1979, testified in the CP remand that CP&L made an all-out effort to get Brunswick 2 running by 12-31-74 in order to get tax benefits. (You know TMI-2 was handled similarly). Shearon Harris is now being "rushed", 3 shifts if necessary, to meet a 1986 commercial operation date, tho NRC Staff believed in a recent check of the schedule that they would be 6 months behind schedule in going on-line.

The Harris pipe hanger situation does not reflect well on CP&L. In September 1980, NRC inspector Maxwell noticed a bad pipe hanger. A sample of 400 were then reinspected (all having been previously approved), and 95% were found defective. Then about 1700 pipe hangers were reinspected and of these, about 1/3 were found OK, tho numbers of them had waivers (I have reviewed thousands of pages of CP&L pipe hanger inspection reports without yet finding a request for field change or waivers that was rejected). About 1/3 were also found defective. The remaining fraction were such that from the blueprints (due to errors or lack of clarity) it could not be told whether the hangers were welded to specification or not.

These weld designs are made, as I understand it, by Bergen-Paterson, reviewdd by Ebasco, then reviewed by CP&L, then sent to Daniel International and welded. At each of these stages, unclear symbols should have been rejected, but were not for hundreds of pipe hangers that were welded (and approved by inspectors) prior to this reinspection.

More pipe hanger problems and support problems have been identified and CP&L now anticipates finishing its 100% reinspection by October 1984, i.e. more than 4 years after the first problem was found by the NRC inspector.

At Apex, the NRC Staff told the Subcommittee that although CP&L had numerous QA/QC violations, they had no safety significance because such violations happen at other plants. Zimmer? Wolf Creek?

The unfortunate history of NRC Staff coverups of defects at plants like Zimmer doesn't just undermine those NRC staffers who are dedicated to safety and quality. It also means that it is not safe to rely on staff reassurances, expecially if safety significance is determined by whether other plants also have such violations, in which case there is no such safety significance according to NRC Staff. I'd like to see them try that argument on a traffic cop who pulled one of them for speeding.

There are about 12000 pipe hangers in the Harris design. Not all appear to be in place. When CP&L retrained their hanger welders, over 10% failed a test immediately after the retraining, by welding a defective hanger yet again. NRC Staff has only inspected about 50 pipe hangers at Harris, according to discovery documents. More information is surely needed on this issue to sustain a recommendation that the plant be found safe. If built by idiots, no ingenious design or operation can save a plant from trouble. Pipe hangers are very important to plant safe shutdown in earthquakes and various transients. They should not be left to such sloppy practice as CP&L has evidenced to date. Such wide-ranging QA failures as CP&L has had on pipe hangers indicate a breakdown of QA, and voluntarily stopping while they try to fix it is about as meritorious as stopping one's car for repairs after a wheel falls off.

There are other Harris problems. For example, the "fix" for its defective steam generators doesn't just include expanding lots of steam generator tubes (which, by NRC rule, are not allowed to leak except for one rupture per steam generator or low volume leakage. Those tubes at Ginna who violated this rule have not been prosecuted to my knowledge). It also involve a jury-rig feedwater arrangement, about 18% through the AFW nozzle, permanently. The safety significance of events like the Maine Yankee water hammer of early 1983 on such jury-rigged arrangements does not appear to have been adequately analyzed. CP&L's "plan" to find voids in pipes where water hammers could start consists of having walkthrough people look for leaks. What x-ray vision they may use to see inside pipes is not clear. Nor does CP&L appear to have identified which piping and valves should be checked for leaks because of possible water hammer events those leaks might indicate are possible (e.g. due to voids in pipes).

CP&L's track record of being wrong is well illustrated by the 3 holes deep down to bedrock that surround Harris #1 -- the sites of cancelled units 2,3 and 4. There's a fault under the Unit 1 containment -- indctive for 2.5 million years or more, CP&L's consultant says, but it still weakens the rock there. NRC seismick analysis appears to be for acceleration only, not for fissures under the plant or parts of the plant moving 2 ways at once. And tho the unit 3 and 4 holes (and perhaps unit 2's will also) be filled in with dirt, they still arent as strong as rock or the basemats and buildings planned for those locations.

Earthquakes are also a very logical source of common-mode failure of all 7 transmission lines to Harris, leading to total loss of offsite power (LOOP). Terrorist attack [illegible], a huge ice storm, a wind storm or hurricane can also lead to such common mode failures of these lines. CP&L only gives a probability for tornado-caused common mode failure.

What about an earthquake-induced crack in the cooling water supply lines? Even were the plant shut down, it still would need to be cooled, and without offsite power and water that would be very difficult.

The spent fuel pool, lying on a ridge between the 2 holes farthest from Unit 1, and along a "ledge" by unit 1 to another hole, is in shaky shape even if the holes are filled with soil. Has CP&L or NRC staff adequotely addressed the possibility of the pool cracking in an earthquake, losing cooling water due to cracked piping or loss of source in a quake, and so perhaps boiling dry? What about criticality as fuel bundlexs swing in a quake? The Diablo Canyon experience shows that undiscovered geological features or design errors can greatly raise the actual risk from earthquakes compared to initial analysis.

Concerning security, the essential water chillers need to be protected. I won't say more on this, not wanting to give any terrorists advice. I do note that we intervenors instructed our experts in security who reviewed the Harris security plan to only make a contention if a serious problem in the plan were found. Minor matters were to be suggestions for improvement only. I understand that 6 security contentions were filed in secret, and that a list of suggestions was also given to CP&L, which rejected all of them, perhaps not noting that they were suggestions. This situation deserves a closer look by the ACRS.

CP&L has been fined repeatedly by NRC for poor practices. Their Brunswick plant has drawn very low SALP ratings. Indeed, at the time of the TMI-2 accident, TMI was rated higher in virtually every area evaluated by NRC, than was Brunswick. This says 2 things: don't trust a high rating, and look into Brunswick. (Data in Board Exhibit-8 of the 1979 remand on CP&L management capability.) CP&L has dragged their feet on fire protection at Brunswick and taken 16 months to put an indicator light from a vital water-tight door protecting ECCS components from flooding, to the control room at Brunswick. The Robinson plant has a very low "site stringency" or level of safety requirements (Bd. Exh. 8, op cit) and thus isn't such a good performer as its record might seem to indicate — it has looser requirements which lead to fewer violations. Good management.

CP&L's control room design leaves much to be desired at Harris. the moved cabinets actually are in each other's way (there is a contention admitted on this), and the SPDS is inadequate (contentions have been filedon this and CP&L has been ordered to produce more SPDS information to me in February. So far, CP&L has not been forthcoming with information like its detailed Human Engineering Requirements Specifications for lights and noise. One SPDS analysis CP&L has submitted has the RCS light GREEN ("safe") during a large LOCA, which doesn't appear very bright at all.

The CP&L management capability contention will go to trial in late summer or early fall as now scheduled. The emergency plan is expected out in mid-February (delayed from December) and will likely be the subject of numerous contentions. All these matters (and those above) merit review. I hope this little information is of some use to you, though it covers only a few of the issues the ACRS must deal with about Shearon Harris 1.

In closing, I'd like to thank the ACRS for its review of Harris. ACRS has the reputation of being the most knowledgeable group associated with the NRC, but of being rather shy to state problems specifically in its letters on plants. We members of the public depend on you as a line of defense of our safety. Though most of you may be "1000 miles away" like the proverbial nuclear construction worker when and if Harris 1 starts up, many people will be closer and will have to live with the plant and CP&L's running of it. Your investigatic and your comments can be of great help to these people. I believe CP&L went through its whole presentation with but minor mention of people around the plant. Those people are real, and they depend on the ACRS to catch problems CP&L and NRC Staff do not. Wells Eddleman

NRC FORM 335 (6 83)

U S NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1 REPORT NUMBER (Assigned by TIDC, add Vol No., if any)

NUREG-1038
Supplement No. 1

2 Leave blank

3 TITLE AND SUBTITLE

Safety Evaluation Report Related to the Operation of Shearon Harris Nuclear Power Plant, Unit 1

4 RECIPIENT'S ACCESSION NUMBER

5 DATE REPORT COMPLETED

MONTH	YEAR
June	1984

6. AUTHOR(S)

7 DATE REPORT ISSUED

MONTH	YEAR
June	1984

8. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

9. PROJECT/TASK/WORK UNIT NUMBER

10 FIN NUMBER

11 SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Same as 8. above

12a TYPE OF REPORT

Safety Evaluation Report

12b PERIOD COVERED (Inclusive dates)

13 SUPPLEMENTARY NOTES

Docket No. 50-400

14 ABSTRACT (200 words or less)

Supplement No. 1 to the Safety Evaluation Report for the application filed by Carolina Power and Light Company and North Carolina Eastern Municipal Power Agency for a license to operate the Shearon Harris Nuclear Power Plant, Unit 1 (Docket No. 50-400), located in Wake and Chatham Counties, North Carolina, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement provides more recent information regarding resolution of some of the open items identified in the Safety Evaluation Report. This supplement also provides and discusses the recommendations of the Advisory Committee on Reactor Safeguards in its report on Shearon Harris, dated January 16, 1984.

15a KEY WORDS AND DOCUMENT ANALYSIS

15b. DESCRIPTORS

16 AVAILABILITY STATEMENT

Unlimited

17 SECURITY CLASSIFICATION (This report)
Unclassified

18 NUMBER OF PAGES

19 SECURITY CLASSIFICATION (This page)
Unclassified

20 PRICE
$

*U.S. GOVERNMENT PRINTING OFFICE : 1984 0-421-297/4021