NUREG-1038
Supplement No. 2

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 1985

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ABSTRACT

This report, Supplement No. 2 to the Safety Evaluation Report for the application filed by the Carolina Power and Light Company and North Carolina Eastern Municipal Power Agency (the applicants) for a 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 and Supplement No. 1.

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1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

In November 1983, the U.S. Nuclear Regulatory Commission staff (NRC staff or staff) issued a Safety Evaluation Report (SER), NUREG-1038, regarding the application by Carolina Power and Light Company and North Carolina Eastern Municipal Power Agency (the applicants) for a license to operate the Shearon Harris Nuclear Power Plant, Unit 1. Supplement No. 1 was issued in June 1984. This report is Supplement No. 2 to the SER.

This supplement provides more recent information regarding resolution of some of the open items identified in the SER and in Supplement No. 1. It also addresses one of the recommendations of the Advisory Committee on Reactor Safeguards in its report on the Shearon Harris plant, dated January 16, 1984, which was inadvertently omitted in Supplement No. 1.

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 a continuation of the chronology of NRC's principal actions related to the safety (or radiological) review of the application. Appendix B 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.

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1.7 Summary of Outstanding Issues

Section 1.7 in the SER and in Supplement No. 1 noted that certain information had not yet been provided by the applicants for several identified items. This supplement updates those items for which additional information has subsequently been provided. These items, and the sections of this supplement discussing the review conclusions, are

(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)

(6) Preservice/Inservice Inspection Program (5.2.4, 6.6)

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

(7) Reactor auxiliary cooling water system (9.2.2)

(8) Periodic testing of instrument air quality (9.3.1)

(9) Unmonitored release of condenser discharge during hogging operations (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) Item III.D.1.1 - leak reduction program (9.3.5)

In addition to the above, two issues regarding the operator training program (Section 13.2.1) and the boron injection tank concentration reduction (Section 15.1.6) are also discussed in this report.

The outstanding items in Section 1.7 of the SER are given in updated Table 1.2 with the current status of each item. For items discussed in this supplement, the specific section is identified. The resolution of the outstanding items that have not been resolved will be discussed in future supplements to the SER.

1.8 Confirmatory Issues

Section 1.8 of Supplement No. 1 to the SER stated that certain confirmatory information will be provided by the applicants. Five of these items, identified as items 24, 25, 26, 27, and 29, have been resolved in the cited sections of this supplement. The SER previously identified the issue of the Preservice/Inservice Inspection Program as Open Item 6. As a result of additional information provided by the applicants, this issue has been reclassified as a confirmatory item and is discussed in Sections 5.2.4 and 6.6 of this supplement. The confirmatory items in Section 1.8 of the SER are shown in updated Table 1.3, which gives the current status of each item.

1.9 License Conditions

Section 1.9 of the SER and Supplement No. 1 listed several probable license conditions. These conditions are given in updated Table 1.4.

Table 1.2 Outstanding items

Item	Status	Section(s)
(1) Design of retaining wall	Resolved	2.5.5
(2) Missiles outside containment	Resolved	3.5.1.1
(3) Functional capability of Class 1 auxiliary piping systems	Resolved	3.9.3
(4) Control of minimum wall thickness in ASME Class 1, 2, and 3 piping systems	Resolved	3.9.3
(5) Equipment qualification	To be resolved	3.10, 3.11
(6) Preservice/Inservice Inspection Program	Changed to Confirmatory Item 34	5.2.4, 6.6
(7) Periodic testing of instrument air quality	Resolved	9.3.1
(8) Fire protection	To be resolved	9.5.1
(9) Unmonitored release of condenser discharge during hogging operations	Resolved	10.4.2, 11.5
(10) Method of estimating noble gas activity from atmospheric steam dump valves	To be resolved	10.4.2, 11.5
(11) Monitoring of all inputs to the service water system	Resolved	11.5
(12) Emergency preparedness	Changed to Confirmatory Item 35	13.3
(13) Steam generator tube rupture isolation time	Changed to Confirmatory Item 36	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	To be resolved	13.5.1
I.C.2 Shift and relief turnover procedures	To be resolved	13.5.1
I.C.3 Shift supervisor responsibilities	To be resolved	13.5.1

Table 1.2 (Continued)

Item		Status	Section(s)
I.C.4	Control room access	To be resolved	13.5.1
I.C.5	Feedback of operating experience	To be resolved	13.5.1
I.C.6	Verification of correct performance of operator activities	To be resolved	13.5.1
I.D.1	Control room design review	To be resolved	18
II.E.1.1	Auxiliary feedwater system reliability evaluation	Resolved	10.4.9
II.F.2	ICC instrumentation	Resolved	4.4.6
III.A.1.2	Emergency support facilities	To be resolved	13.3.4
III.D.1.1	Leak reduction program	Resolved	9.3.5

Table 1.3 Confirmatory issues

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

Table 1.3 (Continued)

Issue	Status	Section(s)
(19) Adequacy of station electrical distribution	To be resolved	8.4.2.3
(20) Use of load sequencer with offsite power	To be resolved	8.4.7
(21) Compliance with Phase I and Phase II of NUREG-0612	To be resolved	9.1.5
(22) Pressure differential alarms	To be resolved	9.4.5.2
(23) Emergency lighting	To be resolved	9.5.3
(24) Radiation monitors for turbine building vent stack	Resolved	10.4
(25) Ability to continuously sample radioiodine and particulates (condenser vacuum pump effluent)	Resolved	10.4.2
(26) Location of high range noble gas monitors (turbine building vent)	Resolved	10.4.2, 10.4.3, 11.5
(27) Drawings for the filters handling sludge	Resolved	11.4.1
(28) Process Control Program	To be resolved	11.4.1
(29) Polymer binder system	Resolved	11.4.1
(30) Radiation protection manager	Resolved	12.5
(31) Corporate management and technical support organization	To be resolved	13.1.1.6
(32) Initial test program • 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	To be resolved	14

Table 1.3 (Continued)

Issue	Status	Section(s)
(33) TMI Action Plan (NUREG-0737)		
I.C.7 NSSS vendor review process	To be resolved	13.5.2.3
II.K.3.5 Automatic trip of RCPs during LOCA	To be resolved	15.9.9
(34) Preservice/Inservice Inspection Program	To be resolved	5.2.4, 6.6
(35) Emergency preparedness	To be resolved	13.3
(36) Steam generator tube rupture isolation time	To be resolved	15.6.3

Table 1.4 License conditions

License Condition	Section(s)
(1) Turbine stem maintenance program	3.5.1.3
(2) Turbine stem valve maintenance	3.5.1.3
(3) II.B.3 Post-accident sampling system	9.3.2
(4) Processing of filter sludge in VR system	Deleted
(5) Operating experience	13.1.2.4
(6) Security plan adherence to regulations	13.7
(7) Restriction above 90% power	15.4.3, 4.3
(8) II.F.2 Instrumentation for inadequate core cooling detection	4.4.6
(9) Physical security	13.6.3

2 SITE CHARACTERISTICS

2.5 Geology and Seismology

2.5.5 Stability of Slopes

2.5.5.2 Properties of Soil Backfill

As stated in the SER, the staff required the applicants to demonstrate that the soil parameters used in the design of the retaining wall are appropriate.

On the basis of the test results provided in the applicants' submittal dated March 7, 1984, the properties of selected impervious material used as backfill against plant structure are:

saturated unit weight, γ_s = 130 pcf
angle of internal friction, ϕ = 20°
cohesion, C = 400 psf

For simplicity of design, the applicants used ϕ = 30° and C = 0, which happen to be the same as the properties of borrow area Z material. The applicants have performed an additional analysis which indicates that the factors of safety for tie rods using soil properties ϕ = 20° and C = 400 psf are higher than those using soil properties ϕ = 30° and C = 0.

The staff, therefore, concurs with the applicants' conclusion that the design parameters used in the design are acceptable.

2.5.5.3 Stability of Retaining Wall

As stated in the SER, the staff required additional analyses to confirm the appropriateness of assumed lateral soil pressures, soil amplification effects, and the design of the deadmen for the retaining wall.

The applicants have performed additional analyses using three sets of soil properties with no surcharge load, with full groundwater hydrostatic pressure, and with and without considering corrosion of the tie rods. The factor of safety for the most critical tie rod under normal loading conditions is greater than 1.5.

The ground acceleration amplification through the soil backfill was analyzed using the computer program FLUSH. The calculated dynamic soil loads were used in the design of the retaining wall and the tie rods. The minimum factor of safety for the most critical tie rod under safe shutdown earthquake conditions is greater than 1.1. The factors of safety for the most critical tie rod under different loading conditions are presented in Table 2.1.

The deadmen for the retaining wall were designed in accordance with the procedures described in NAVFAC DM-7.2 (Department of the Navy, 1980). The lateral soil pressures and soil amplification effects were properly accounted for.

Therefore, the staff concurs with the applicants' conclusion that the retaining wall and the tie rod design are acceptable.

2.5.5.5 Settlement and Monitoring Program

As stated in the SER, settlement analyses are necessary to determine the effect of differential settlement on the design and performance of the retaining wall.

The applicants, on the basis of the observation of settlement at cooling tower #1, predicted that the settlement of the retaining wall would not be greater than 1 in. and the differential settlement would not have any signficant effect on the performance of the retaining wall. To ensure the performance of the retaining wall, the applicants proposed to monitor the settlement and horizontal movements of the retaining wall during and after construction. The frequency of monitoring will be in accordance with Position C.4 of Regulatory Guide (RG) 1.127 (Rev. 1). The applicants will also install three specimens of tie rods in the same soil environment for future retrieval and testing to evaluate corrosion of the tie rods.

The staff concurs with the applicants' conclusion that the proposed program is adequate to ensure the performance of the retaining wall and is, therefore, acceptable.

2.5.5.6 Conclusions

On the basis of the information submitted in letters dated March 7 and April 19, 1984, the staff finds that the designed stability and the monitoring program for the retaining wall are adequate and acceptable.

Table 2.1 Soil properties and factors of safety for the most critical component (tie rods at elevation 232.83 ft) according to type of fill

Soil property and load case	Selected impervious fill: Estimated value	Selected impervious fill: Design parameter	Random or modified random fill
Soil property			
Saturated unit weight, γ_s (pcf)	130	135	140
Angle of internal friction, $\phi(°)$	20	30	25
Coehesion, C (psf)	400	0	120
Load case	Factor of safety		
(1) Without hydrostatic pressure			
(a) Normal load condition	2.50	2.45	2.15
(b) Design-basis event (DBE) load condition	1.77	1.75	1.59
(2) With hydrostatic pressure			
(a) Without corrosion			
1. Normal load condition	1.87	1.70	1.60
2. DBE load condition	1.42	1.33	1.27
(b) With corrosion (1/8-in. reduction in diameter)			
1. Normal load condition	1.67	1.52	1.43
2. DBE load condition	1.27	1.19	1.13

Notes:

(1) The minimum acceptable factor of safety for the normal load condition is 1.5, and that for the DBE load condition is 1.1 (Final Safety Analysis Report (FSAR) Section 3.8.5.5).

(2) The factors of safety are based on 90% of the minimum yield strength of the tie rod material. The corresponding factor of safety, for all values listed above, based on minimum yield strength of the tie rod material would be increased by the ratio of 1.00:0.9; for example, for load case (2)(b)1, random or modified random fill, the factor of safety of 1.43 would be increased to 1.59.

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)

(1) Effect Of Gravitational and Secondary Missiles on Essential Services Chilled Water System (ESCWS)

In the SER, the staff stated that the applicants had omitted the ESCWS from the list of systems that required protection against missiles. The applicants, in a submittal dated October 25, 1984, agreed to include the ESCWS in FSAR Table 3.5.1-1 as a system required for safe shutdown and as one that has to be protected against internally generated missiles. The applicants made this correction in Amendment 17, dated November 1, 1984.

In Supplement No. 1 to the SER, the staff noted that the applicants had considered that missiles from high pressure systems, rotating machinery, gravitational missiles, and secondary missiles were not credible or would not affect the ESCWS. The staff found this unacceptable. The applicants then reviewed the spectra of possible missiles, which included a study of the potential high energy systems, and determined that the only missiles possible from such systems would derive from instrument wells. The applicants concluded that such missiles would not affect the ESCWS (FSAR Table 3.5.1-17) and that the ESCWS is protected from the effects of internally generated missiles from pumps.

Further, the applicants reviewed the effects of gravitational missiles and secondary missiles on the ESCWS. However, the staff was concerned that, in this review, the applicants had not property complied with its position regarding the effect of missiles from safety-related and non-safety-related components.

The applicants' most recent response to the staff's concern is reviewed below:

(a) Gravitational Missiles

The applicants stated that equipment, systems, components, and/or structures in the vicinity of the ESCWS are either seismically designed to ensure their integrity in the event of a safe shutdown earthquake or a determination has been made that postulated failure of components nonseismically designed will not cause either redundant ESCWS train to become inoperable. The staff finds the applicants' response acceptable and considers this issue closed.

(b) Secondary Missiles

The applicants stated that, in the event of secondary missile generation, one of the following interactions would occur:

- Safety-related equipment would not be within the strike zone of the secondary missiles generated.
- The impact energy from secondary missiles on safety-related equipment is negligible and would not cause any significant damage.
- Barriers and compartmentalization of safety-related equipment would confine secondary missiles to a finite area so that missiles from this equipment would be incapable of negating the necessary functioning of the ESCWS.

The staff finds the applicants' response acceptable and considers this issue closed.

(2) Internally Generated Missiles From Pumps

In Supplement No. 1 to the SER, the staff noted that, in regard to pump missiles, the applicants had stated that, for pumps within the nuclear steam supply system, the no-load speed and operating speeds are the same. The applicants also had said that the balance of plant (BOP) is designed so that missiles from internal sources cannot damage safety features so as to jeopardize minimum required safety function.

The staff found both answers unacceptable and asked the applicants to consider missile production, even from well-designed pumps, and to provide details regarding the protection of safety-related systems, structures, and components (SSC) from missiles from pumps within the BOP scope.

In response the applicants reviewed the pumps outside containment in accordance with the following five criteria to ensure protection against missile damage by pumps:

(a) Safety-related pumps that, because of their location, cannot affect multiple power or component trains by generated missiles shall not be considered as a potential danger to the safety of the plant.

(b) Safety-related pumps that are isolated in cubicles and cannot damage other essential equipment by generated missiles shall not be considered a potential danger to the safety of the plant.

(c) Non-safety-related pumps that are located in non-safety-related areas or are isolated in cubicles and cannot damage essential equipment by generated missiles shall not be considered a potential danger to the safety of the plant.

(d) Pumps for which vendor calculations or certifications are available stating that the pumps cannot generate missiles that penetrate the pump casing shall not be considered a potential danger to the safety of the plant.

(e) Any pumps that are intended for use only during cold shutdown shall not be considered a potential danger to the safety of the plant.

All pumps met one of these first four criteria with only one exception, the steam generator wet layup recirculation pump, which met Criterion e.

The staff believes that pumps that comply with Criteria a-d will not damage safety-related SSC; however, it was not satisfied with Criterion e. It noted that Section 3.5.1.1 of the Standard Review Plan (SRP, NUREG-0800), "Internally Generated Missiles (Outside Containment)," states that SSC "necessary to perform functions required for attaining and maintaining a safe shutdown condition - must be assured adequate protection from internally generated missiles."

The staff then asked the applicants to discuss how all SSC necessary to maintain shutdown were adequately protected from missiles from the steam generator layup recirculation pump.

The applicants provided a revised Criterion e, which states: "Any pumps which are used only after cold shutdown and cannot damage any structures, systems, and components necessary to maintain shutdown shall not be considered a potential danger to the safety of the plant."

Because this criterion applies only to the steam generator layup recirculation pump, which is intended to be used only during cold shutdown, and because the criterion states that SSC necessary to maintain shutdown cannot be damaged by pumps included under this criterion, the staff considers this matter resolved.

In view of the discussion under Items 1 and 2 above, the staff considers the issue of internally generated missiles outside containment closed.

3.9 Mechanical Systems and Components

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

In Section 3.9.3.1 of the SER, the staff identified an open issue regarding the control of minimum wall thickness in American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Class 1, 2, and 3 piping systems. In the piping specification (CAR-SH-M-30), a minimum wall thickness is specified (1) at weld end preparations, (2) after machining or grinding the pipe wall, (3) in trimming the pipe inside diameter, and (4) after counterboring the pipe ends. The applicants have stated that these minimum wall thicknesses are based on equations per ASME Code, Section III, Subparagraphs NB/NC/ND-3641.1. However, the use of the nominal wall thickness in the piping stress analyses (rather than the minimum wall thickness) and the use of a stress intensification factor of 1.0 for girth butt welds are not acceptable because these factors can result in an underprediction of piping stresses at those girth butt welds where the piping wall thickness is less than the nominal wall thickness.

In a letter dated February 7, 1984, the applicants responded to the staff's concern. Subsequently, on February 16, 1984, the staff met with the applicants to discuss the details of this issue. The staff found that the pipe fabricator (Southwest Fabricating and Welding Company, Inc.) uses 87 1/2% of nominal wall thickness as the minimum wall thickness when the piping line list specifies a nominal wall thickness. When machining to the counterbore diameter (as specified in the Ebasco design guide) reduces the wall thickness in the counterbored area to less that 87 1/2% of the specified nominal thickness, the piping fabricator will build up the area with weld metal to ensure that the thickness is not less than 87 1/2% of nominal thickness. Thus, the piping fabricator appeared to provide assurance that the minimum wall thickness permitted in the design specification based on ASME Code equations in Subparagraphs NB/NC/ND-3641.1 would not occur. However, to provide the staff with added assurance that the condition permitted in the design specification did not actually occur, the staff requested the applicants to examine, on a sampling basis, the actual wall thicknesses of several girth butt welds on those piping systems most susceptible to this condition. In a letter dated November 7, 1984, the applicants provided the staff with the results of in situ examinations of girth butt welds for piping in the residual heat removal and component cooling water systems. Of the 20 welds examined, none were found to have a wall thickness less than the minimum wall thickness used by the piping fabricator (87 1/2% of nominal wall thickness).

On the basis of its review of the criteria used by the piping fabricator to ensure that the minimum wall thickness will not result in a thickness below the minimum specified (i.e., 87 1/2% of nominal wall thickness) and the sample of welds actually measured in the plant, the staff concludes that the use of nominal wall thickness in the piping stress analyses (rather than the minimum wall thickness permitted in the design specification) and the use of a stress intensification factor of 1.0 for girth butt welds are acceptable. Thus, the staff considers this issue closed.

In Section 3.9.3.1 of the SER, the staff identified an open item regarding the use of a stress limit of 3.0 Sm, as stated in Appendix F of ASME Code, Section III, to ensure the functional capability of ASME Code, Class 1 auxiliary piping systems. The faulted limit given in Appendix F is intended to ensure the structural integrity of piping systems; however, the staff had not previously accepted the 3.0-Sm limit to ensure the functionality of piping systems. The applicants committed to provide the justification that would demonstrate that the 3.0-Sm stress limit could not cause a collapse or gross deformation in the piping to the extent that the functionality of the piping could be impaired.

In a letter dated September 6, 1984, the applicants submitted their final report to the staff. The report presents the methodology and evaluation criteria that demonstrate the functional capability of ASME Code, Class 1 auxiliary piping systems. The criteria presented in the report, which utilizes the equations and definitions given in the ASME Code (1977 Edition up to and including the Summer 1979 Addenda), are principally based on inelastic analysis techniques and establish deformation limits so that small reductions in pipe cross-sectional area are ensured. Finite element analyses of three-dimensional modeled pipe elbows (shell elements) with elasto-plastic strain hardening material properties and with large deformation considerations were performed. The functional capability of the piping was evaluated by computing the degree of ovalization and resulting percentage change in flow area for different values of applied moment loadings up to failure. The report concluded that, for the specific cases of relatively thick piping (schedule 160) found in the ASME Code, Class 1 portions of pressurized water reactors and for the Shearon Harris facility in particular, the small deformation limit is bounded by a stress of 3.0 Sm calculated by a linear elastic analysis and, thus, represents an acceptable limit to ensure the functional capability of the piping.

On the basis of its review of the applicants' analyses, the staff finds that the applicants have provided an acceptable technical basis for the use of the 3.0-Sm stress limit to ensure the functionality of ASME Code, Class 1 auxiliary piping systems at the Shearon Harris facility. Thus, the staff considers this item closed.

5 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

This section was prepared with the technical assistance of Department of Energy (DOE) contractors from the Idaho National Engineering Laboratory. This evaluation supplements conclusions in the SER that addressed the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g). The SER describes Open Item 6 concerning the Preservice Inspection (PSI) Program. The applicants have provided additional information on this issue that sufficiently resolves Open Item 6 so that the subject can be considered a confirmatory issue.

5.2.4.3 Evaluation of Compliance With 10 CFR 50.55a(g) for Unit 1

Review has been completed on the information presented in the FSAR through Amendment 19 dated January 1985; Revision 2 of the Reactor Pressure Vessel (RPV) Preservice Inspection Program submitted March 14, 1983; Revision 1 of ISI-201 ASME Preservice Inspection Program Plan (except RPV) submitted March 5, 1985; the inspection isometric drawings submitted June 15, 1984; and Region II Inspection Report No. 50-400/84-40. This report describes the results of a meeting with the applicants on November 2, 1984, concerning the ultrasonic examination of cast stainless steel pipe welds.

In accordance with the requirements set forth by 10 CFR 50.55a, the PSI Program must meet the requirements of the 1974 Edition of Section XI, including Addenda through Summer 1975. However, the applicants have elected to meet the requirements of a more recent Code and Addenda and to comply with the 1980 Edition of Section XI, including Addenda through Winter 1981 and ASME Code Case N-335. The extent of examination of Class 1 and Class 2 piping welds has been determined by the requirements of the 1974 Edition of Section XI including Addenda through Summer 1975 as permitted by 10 CFR 50.55a(b)(2). The applicants have responded to issues identified by the staff in this section of the SER by providing additional information.

Revision 1 of ISI-201 states that the line numbers identified in the line lists with the corresponding required examination method are applicable for PSI examinations on all circumferential welds located on that portion of pipe. This approach is conservative for weld selection on Class 2 piping and consequently will provide enhanced baseline information for future inservice examinations. The applicants also committed to submit all requests for relief from specific areas where the PSI Code requirements cannot be met and include a supporting technical justification. This submittal will be made at least 6 months before anticipated fuel loading.

On November 2, 1984, the applicants met with NRC Region II representatives to demonstrate the capability of their ultrasonic examination procedure and instrumentation to detect actual flaws and artificial reflectors in the region's centrifugally cast stainless steel test specimens. The applicants successfully demonstrated that the equipment and procedures could detect actual flaws in the volume subject to examination. These test specimens represented examination conditions that are more difficult than will be experienced during examination of the wrought stainless steel reactor coolant piping at Shearon Harris. ASME Code Case N-335 includes provisions that supplement and improve the requirements for the ultrasonic testing of pipe welds. Therefore, the staff concludes that the applicants' equipment and procedures have demonstrated conservative detection capability for preservice examination and this issue is resolved.

On the basis of its review of the applicants' submittals and the results of the November 2, 1984, meeting with the applicants, the staff has determined that the Shearon Harris Preservice Inspection Program for the reactor coolant pressure boundary is acceptable and the review is a confirmatory issue contingent upon the applicants submitting all requests for relief from impractical examination requirements with a supporting technical justification. The staff will report this evaluation in a future supplement to the SER.

The initial Inservice Inspection Program has not been submitted by the applicants. This program will be evaluated after the applicable ASME Code edition and addenda can be determined based on 10 CFR 50.55a(b), but before inservice inspection commences during the first refueling outage.

6 ENGINEERED SAFETY FEATURES

6.4 Control Room Habitability

Amendment 19 (dated January 31, 1985) to the FSAR included revised data on the amount and location of onsite chlorine. The staff has evaluated the revised data and concludes that they do not affect its earlier conclusion in the SER on the acceptability of the control room habitability during toxic gas release accidents. This section supersedes Section 6.4 of the SER in its entirety.

The requirements for the protection of control room personnel under accident conditions are specified in General Design Criterion (GDC) 19, 10 CFR 50, Appendix A. The applicants have proposed to meet these requirements by incorporating shielding and emergency ventilation systems in the control room and by having an adequate supply of self-contained breathing apparatus in the control room for the emergency team. The applicants have stated in the FSAR that the control room habitability system is designed as seismic Category I and is designed to function properly following any single active failure.

The control room will be isolated - by the closure of the isolation dampers - on receipt of a containment isolation actuation signal or following detection at the outside air intake of significant quantities of radiation, smoke, or chlorine. In addition, the control room will be isolated on detection of chlorine by remote chlorine detectors located at the storage site. In the event of a high radiation signal, the operator will override the intake radiation detector isolation signal and/or the containment isolation actuation signal and open the least contaminated intake for pressurization of the air in the control room. The operator can also manually initiate the smoke purge system following an event such as a smoke alarm in the control room area.

Each emergency filter train in the control room heating, ventilation, and air conditioning system supplies approximately 4000 cfm of filtered air, of which at least 3200 cfm is return air recirculation, and up to 800 cfm is outside air for pressurizing the control room. The emergency filtration units utilize 99% efficient, seismic Category I, atmospheric cleanup units for removal of radioactive iodine.

The staff has evaluated the control room doses following a postulated design-basis loss-of-coolant accident in accordance with SRP Section 6.4. The calculated whole-body and thyroid doses are within the guidelines of SRP Section 6.4.

The staff has also evaluated the habitability of the control room with respect to toxic gases in accordance with SRP Section 6.4 and RGs 1.78 and 1.95. The major potential toxic gas problem at Shearon Harris is the 55 tons of chlorine to be stored on site about 1550 ft from the nearest control room normal air intake. The FSAR evaluation of the chlorine hazard assumes a maximum control room inleakage of 142 cfm (i.e., 0.06 volume/hour). On the basis of this evaluation, a higher inleakage would result in an unacceptably high chlorine concentration in the control room following an onsite chlorine tank car rupture

The staff will, therefore, require a Technical Specification to the Operating License that will require periodic testing that demonstrates that the control room pressurization flow is less than or equal to 142 cfm at pressures greater than or equal to 1/8-in. water gauge. Such a Technical Specification is consistent with the Standard Technical Specifications (NUREG-0452). Subject to the implementation of this Technical Specification requirement, the staff concludes that the control room habitability system provides adequate protection against toxic gases.

On the basis of the foregoing, the staff concludes that the applicants have demonstrated that the control room habitability systems will adequately protect the control room operators in accordance with the requirements of NUREG-0737, Item III.D.3.4, and GDC 19.

6.6 Inservice Inspection of Class 2 and 3 Components

This section was prepared with the technical assistance of DOE contractors from the Idaho National Engineering Laboratory. This evaluation supplements conclusions in the SER that addressed the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g). The SER describes Open Item 6 concerning the Preservice Inspection (PSI) Program. The applicants have provided additional information on this issue that sufficiently resolves Open Item 6 so that the subject can be considered a confirmatory issue.

6.6.3 Evaluation of Compliance With 10 CFR 50.55a(g)

Review has been completed on the information presented in the FSAR through Amendment 19 dated January 1985; Revision 1 of ISI-201 ASME Preservice Inspection Program Plan (except RPV) submitted March 5, 1985; the inspection isometric drawings submitted June 15, 1984; and a letter dated May 9, 1985.

In accordance with the requirements set forth by 10 CFR 50.55a, the PSI Program must meet the requirements of the 1974 Edition of ASME Code Section XI, including Addenda through Summer 1975. However, the applicants have elected to meet the requirements of a more recent Code and Addenda and to comply with the 1980 Edition of Section XI, including Addenda through Winter 1981, and ASME Code Case N-335. The extent of examination of Class 2 piping welds has been determined by the requirements of the 1974 Edition of Section XI including Addenda through Summer 1975 as permitted by 10 CFR 50.55a(b)(2). The applicants have responded to issues identified by the staff by providing additional information.

Revision 1 of ISI-201 states that the line numbers identified in the line lists with the corresponding required examination method are applicable for PSI examinations on all circumferential welds located on that portion of pipe. This approach is conservative for weld selection on Class 2 piping and consequently will provide enhanced baseline information for future inservice examinations. Category C-F components are being excluded from the volumetric and surface examination requirements of Paragraph IWC-2500 in accordance with Paragraph IWC-1220 (1974 Edition through and including Summer 1975 Addenda), except that the control of water chemistry to minimize stress corrosion described in Paragraph IWC-1220(c) is not used as a basis for exempting components.

10 CFR 50.55a(b)(2)(iv) requires that ASME Code Class 2 piping welds in the residual heat removal (RHR) systems, emergency core cooling (ECC) systems, and

containment heat removal (CHR) systems shall be examined. These systems should not be completely exempted from preservice volumetric examination based on Section XI exclusion criteria contained in Paragraph IWC-1220. To satisfy the inspection requirements of General Design Criteria (GDC) 36, 39, 42, and 45, the PSI Program must include volumetric examination of a representative sample of welds in the RHR, ECC, and CHR systems. In ISI-201, Revision 1, Appendix D (Augmented Examinations), the applicants included two 12-in.-diameter and two 14-in.-diameter lines in the RHR system for volumetric examination. However, the document indicates that the containment spray system will receive only surface examinations. In a letter dated May 9, 1985, the applicants stated that the PSI Program contains a typographical omission. Volumetric examinations are being performed on containment spray lines 2 CT 8-10 and 2 CT 8-15. The applicants will add these two lines and correct the examination method in the next revision of Procedure ISI-201.

The applicants also committed to submit all requests for relief from specific areas in which the Code requirements for PSI cannot be met and to include a supporting technical justification. This submittal will be made at least 6 months before fuel loading.

On the basis of the applicants' submittals, the staff has determined that the PSI Program for Class 2 and 3 components is acceptable and the review is a confirmatory issue contingent on the applicants revising the volumetric examination of the welds in the containment spray system based on the May 9, 1985, letter and submitting all requests for relief from impractical requirements with a supporting technical justification. The staff will report this evaluation in a future supplement to the SER.

The initial Inservice Inspection Program has not been submitted by the applicants. This program will be evaluated after the applicable ASME Code edition and addenda can be determined based on 10 CFR 50.55a(b), but before inservice inspection commences during the first refueling outage.

9 AUXILIARY SYSTEMS

9.2 Water Systems

9.2.2 Primary Component Cooling Water System

The applicants provided, on their own initiative, a revised response relating to the portion of the component cooling water (CCW) system used to cool the reactor coolant pump (RCP) motor bearings and seals. This issue had been resolved in the SER.

The CCW system provides cooling to several heat sources, including the RCP thermal barriers, which limit heat transfer from the hot reactor coolant to the RCP seals. In the event of a Phase B isolation signal, seal injection water is provided to the RCP seals to prevent overheating; seal injection water is provided by the chemical and volume control system. However, no water is provided to cool the RCP motor bearing oil cooler to prevent overheating of the bearings.

In the new response, dated April 9, 1984, the applicants stated that loss of cooling water to the bearing oil coolers would be detected by redundant, safety-grade flow instrumentation with alarms in the control room to alert the operator to loss of flow.

The applicants also committed to provide procedures to initiate manual protection of the RCPs in the event cooling water cannot be restored within 10 min. The applicants also noted the provision of other instrumentation to alert the operator to single failures, as follows:

(1) CCW containment isolation valve position indicators

(2) CCW flow alarms for each of the RCP motor oil coolers return lines (upper and low oil cooler common return)

(3) CCW temperature alarm for the common return lines from all RCP motor oil coolers

(4) RCP motor upper thrust shoe, lower thrust shoe, and upper guide bearing temperature alarms (these bearings are served by the upper oil cooler)

(5) RCP motor lower guide bearing temperature alarm (this bearing is served by the lower oil cooler)

(6) RCP motor winding temperature alarm

The applicants also propose to lock open the manually operated isolation and throttle valves on the CCW lines to the RCP motor bearing coolers to prevent inadvertent operator action. The applicants note that RCP motor vibration detectors and alarms are available to detect excess vibration.

The staff finds the applicants' new response acceptable and considers this matter closed.

9.3 Process Auxiliaries

9.3.1 Compressed Air System

In the SER, the staff stated that the applicants had not met the full requirements for periodic testing of instrument air quality as recommended in American National Standards Institute (ANSI) MC 11.1-1976 (Instrument Society of America (ISA) S7.3), "Quality Standard for Instrument Air."

The applicants, in a submittal dated November 4, 1983, stated that the instrument air system supplied air to operate safety-related valves. Some of these fail in a safe position on loss of air. The remaining safety-related valves require a motive force to function and are supplied with air accumulators to function on loss of the instrument air system. Thus, all air-operated safety-related valves are independent of the instrument air systems in the event of an accident. Valves in this latter class include the power-operated relief valves on the pressurizer, containment hydrogen purge valves, and containment vacuum relief valves. The applicants noted, further, that the instrument air system provides air for many plant components that do not have an effect on accident analyses or where non-random, multiple failures are bounded by the accident analyses; these components are part of the waste processing system, water treatment system, and portions of the main steam, condensate, and feedwater systems that are isolated by the main steam isolation valves or the main feedwater isolation valves. The applicants stated that a regulatory commitment to instrument air quality is not required for these components.

The applicants stated that component filters would be provided in front of all the safety-related valves that require air to operate. The filters would be designed in accordance with recommendations by the valve manufacturers. The applicants noted that relatively little air is required to operate the valves so that little filter degradation is expected.

The applicants, in a submittal dated June 21, 1984, reported that sampling for particles would be done at the filter to verify proper filter operation. Results of this sampling would be used to determine a schedule for cleaning or replacing the filter. The applicants proposed to use the 3-micron particle level as a goal and not as an acceptance criterion. The staff was concerned about the plan to establish an acceptance criterion and the periodicity of inspection of valve filters. The applicants' submittal of April 5, 1985, contained a response to these concerns; the response is discussed under Item 1 below.

In the submittal dated June 21, 1984, the applicants stated that a dew point acceptance criterion of -20°F would be established for the instrument air system. The staff asked the applicants to ensure that this would be at least 10C° (18F°) below the lowest temperature expected in the instrument air system. The applicants responded to this concern in the June 5, 1984, submittal; the response is discussed under Item 2 below.

In discussions with the applicants, the staff told them of its concern regarding possible contaminants in the instrument air system before instrument air testing. The applicants responded to this concern in the June 5, 1984, submittal; the response is discussed under Item 3 below.

(1) Instrument Air Testing - Particulates

The staff was concerned about the applicants' plan to determine particulate acceptance criteria for the instrument air system, since no details had been provided. The staff also asked the applicants to provide information regarding the periodicity selected for the inspection of filters placed in front of valves that are designed to filter out particulates larger than those that have been found acceptable by the valve manufacturer.

The applicants stated that filtration of particles larger than 3 microns in the air stream of the instrument air system would be used as a goal. Therefore, the operation of the 3-micron filter would be monitored. Filter operation would be investigated and corrective action taken when the efficiency of filtration for particles larger than 3 microns became less than 95%. If a significant increase in the size and quantity of particles in excess of 3 microns is found, the applicants will investigate the filtration and identify corrective actions. The applicants will modify these criteria as required, on the basis of operating experience and compatibility with the actual air requirements for the safety-related, air-operated valves.

The applicants stated that filters will be replaced once each refueling outage. This surveillance period will be modified as experience dictates.

The staff finds this response acceptable and considers this issue resolved.

(2) Instrument Air Testing - Dew Point

The staff was concerned that the dew point in the instrument air system would not be maintained low enough to comply with the standard specified in ANSI MC 11.1-1976 (ISA S7.3), which specifies a dew point 10C° (18F°) lower than the lowest instrument air system temperature. The applicants stated that testing for dew point in the instrument air system will be conducted at least annually. The acceptance criterion for dew point will be -25°F, which is 18F° lower than the lowest outside air temperature (-7°F) experienced at the site.

The staff finds this acceptable and considers this issue resolved.

(3) Contaminants

The applicants flushed the instrument air system with high quality air to remove loose material that may have accumulated in the system during construction. The applicants also committed to test for hydrocarbons at least once every refueling outage. The acceptance criterion for hydrocarbons will be less than 1 ppm on a weight or volume basis (w/w or v/v).

The staff finds this acceptable and considers this issue resolved.

In view of the discussions above, the staff considers the matter of the compressed air system closed.

9.3.5 Item III.D.1.1 Integrity of Systems Outside Containment Likely to Contain Radioactive Materials

In the SER, the staff identified as a subpart of Outstanding Item 14 that the applicants should provide additional information on their leak reduction program to satisfy NUREG-0737, Item III.D.1.1, regarding the minimization of postaccident leakage outside the containment.

In a letter dated September 21, 1983, supplemented by a letter dated July 11, 1984, the applicants provided a description of the program designed to reduce leakage from systems outside containment that would or could contain primary coolant or highly radioactive fluids during a serious transient or accident. The applicants' program will be initiated during the preoperational test phase and will be performed in accordance with the applicable provisions of ASME Code, Section XI. The program is designed to detect leakage to the reactor auxiliary building (RAB) atmosphere from systems that would be used to bring the plant to a safe shutdown following a serious transient or accident. The following systems are included in the leak reduction program:

(1) residual heat removal system

(2) safety injection system, except boron injection recirculation subsystem

(3) containment spray system, except spray additive subsystem and refueling water storage tank

(4) chemical and volume control system (letdown subsystem, demineralizers, boron recycle holdup tanks, charging pumps, and seal water return system)

(5) postaccident sampling system

(6) postaccident RAB ventilation system

(7) valve leakoff equipment drain system

(8) gaseous waste processing system

The applicants' leak reduction program will employ (1) visual inspections of the mechanical joints and seals in the various systems to detect leakage and (2) measurement of observed leakage. These inspections will be conducted with the system pressurized to normal operating pressure using the system fluid or demineralized water as a test medium. The applicants will document the observed leakage and compare it against the acceptance criteria developed by them. Corrective action will be taken as appropriate, but in all cases, the action will be to reduce leakage to as-low-as-practical levels. Testing of gaseous systems will include helium leak detection or equivalent testing methods.

The applicants will initiate a program of preventative maintenance to reduce leakage to as-low-as-practical levels after commercial operation. This program will include periodic leak tests at each refueling.

The applicants have evaluated potential leak paths as discussed in the NRC letter from D. G. Eisenhut dated October 17, 1979, regarding the North Anna incident and, if a need is identified, have committed to implement design changes before fuel loading to eliminate the open paths to the atmosphere as discussed in the letter.

The staff reviewed the applicants' proposed program to reduce leakage from systems containing primary coolant or highly radioactive fluids outside containment in accordance with Item III.D.1.1 of NUREG-0737. On the basis of this review and the applicants' supplement to this program dated July 11, 1984, the staff has concluded that the applicants' program will meet the requirements of NUREG-0737 and is acceptable. Therefore, the staff considers this item closed.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection

Safe Shutdown

The applicants' safe shutdown analysis is contained in FSAR Section 9.5-1 and the "Safe Shutdown Analysis in Case of Fire," dated June 20, 1983. The applicants' letters dated February 24 and June 12, 1984, provide additional information and clarification of the safe shutdown analysis. The applicants' report identifies 23 fire areas that comply with Section C.5.b of Branch Technical Position (BTP) CMEB 9.5-1. Nine fire areas are identified where deviations from staff guidelines have been requested.

The staff has reviewed the fire protection provided for safe shutdown cables and equipment to verify that one train of cables and equipment needed for safe shutdown will be maintained free of fire damage. Except as noted below, all plant areas containing cables and equipment needed for safe shutdown are provided with fire protection measures consistent with Section C.5.b of the staff's guidelines.

In some areas the fire protection for safe shutdown capability deviates from the staff's guidelines. The staff has reviewed the applicants' fire protection measures in these areas to determine if a level of safety equivalent to the technical requirements of Section C.5.b of the staff's guidelines has been provided:

(1) Partial Suppression and Detection

- Fire Area 1-A-BAL

  This fire area consists of various portions of the reactor auxiliary building on elevations 190 ft, 216 ft, 236 ft, 261 ft, and 286 ft. The applicants' letter dated February 24, 1984, identifies various zones within fire area 1-A-BAL that are not provided with complete automatic suppression and/or detection systems. The staff has evaluated the fire protection features of these zones and concludes that, because of the low combustible loading, lack of concentrations of cable trays, and existing protection provided for safe shutdown equipment, the addition or extension of automatic suppression and/or detection systems into these zones would not greatly enhance fire protection.

- Fire Area 1-C

This fire area consists of all levels within containment. All cables and equipment in this area needed for safe shutdown are provided with fire protection in accordance with Section C.5.b of the staff's guidelines. Sprinkler systems are provided for the electrical cable trays, electrical penetration areas, and the charcoal filter housings. The areas without sprinkler systems contain low combustible loadings. On the basis of its evaluation, the staff concludes that, because of the lack of in situ combustible material and limited access, the addition of complete suppression and/or detection systems inside containment would not greatly enhance fire protection.

(2) One-Hour-Rated Fire Barriers

- Fire Zones 1-A-3-MP, 1-A-3-PB, and 1-A-4-CHLR

These fire zones are located within fire area 1-A-BAL. A deviation has been requested from providing 1-hour-rated fire barriers between redundant air handling units AH-11 A and B, AH-6 A and B, AH-7 A and B, AH-19 A and B, and AH-20 A and B.

The redundant pairs of air handling units are located throughout fire area 1-A-BAL. Each set of air handlers consists of redundant A and B units. There is adequate separation between pairs (A and B) of units to preclude the loss of more than one pair in a fire.

The redundant (A and B) air handling units are separated by approximately 10 ft. Automatic suppression and detection systems are provided over each set of redundant air handlers. The redundant air handling units function only as local coolers. On the basis of its evaluation, the staff concludes that, because of the automatic suppression systems and the fact that the air handling units provide only local cooling, a fire that would damage redundant air handling units would result only in the loss of local cooling. It, therefore, concludes that the addition of 1-hour-rated fire barriers between the redundant air handlers would not greatly enhance fire protection.

(3) Intervening Combustible Material

- Fire Zones 1-A-3-PB, 1-A-4-CHLR, and 1-A-3-CEMB

These zones are located in fire area 1-A-BAL. Deviations are requested from the requirement to provide 20 ft of separation, free of intervening combustible materials, for the following equipment:

- auxiliary feedwater pumps PIA-SA and PIX-SAB
- component cooling water pumps 1B-SB and 1C-SAB
- component cooling water heat exchangers 1A-SA and 1B-SB

- service water booster pumps 1A-SA and 1B-SB

- charging pumps 1A-SA, 1B-SB, and 1C-SB

- heating ventilation, and air conditioning (HVAC) chillers WC-2 1A-SA and 1B-SB

- chilled water pumps P4 1A-SA and 1B-SB

- condenser water circulation pumps 1A-SA and 1B-SB

Each of the above listed pieces of equipment is separated from the redundant train by more than 20 ft; however, there are intervening cable trays that contain Institute of Electrical and Electronics Engineers (IEEE) 383-rated cables. Fire stops consisting of noncombustible materials such as 1-hour-rated fire barriers or penetration seals are provided in the intervening cable trays. The fire stops serve to prevent the spread of flame along the cable trays between the redundant equipment. Full automatic sprinkler coverage is provided in all of the zones. On the basis of its review, the staff concludes that the full automatic sprinkler coverage in addition to the cable tray fire steps will provide a level of protection equivalent to that specified in Section C.5.b of the staff's guidelines.

(4) Component Cooling Water (CCW) Pumps

The CCW pumps are located in fire zone 1-A-3-PB within fire area 1-A-BAL. The A-train pump is separated from the swing pump by approximately 15 ft. The B-train pump is separated by more than 100 ft. Full automatic suppression and detection systems are provided in the area. If the B-train pump is out of service for more than 7 days, a fire watch will be established in the area. Separating the A pump and the B swing pump by an additional 5 ft will not greatly enhance fire protection. On the basis of its review, the staff concludes that this combination of protection provides a level of protection equivalent to that specified in Section C.5.b of the staff's guidelines.

(5) Charging Pumps

The charging pumps are also located in fire zone 1-A-3-PB. The pumps are separated by approximately 17 ft and are located within individual concrete cubicles that are 3-hour-rated fire barriers. The airtight door to each pump room is not a fire-rated door. Automatic suppression and detection systems are provided in each pump room and in adjacent areas. On the basis of its review, the staff concludes that this combination of protection provides a level of protection equivalent to that specified in Section C.5.b of the staff's guidelines.

(6) Diesel Oil Storage Areas (Unit 1 Fire Area 5-0-BAL)

The diesel oil storage area is a separate building enclosed by 3-hour-rated fire barriers containing non-fire-rated doors. A deviation is requested from providing 3-hour-rated fire doors at the exterior stairways to the building. There are no in situ fire hazards in the yard area adjacent to the diesel oil storage building. On the basis of its review, the staff concludes that, because of the lack of significant fire exposure to the building, the addition of 3-hour-rated fire doors would not greatly enhance fire protection.

(7) Emergency Service Water Intake Structure (Fire Areas 12-I-ESWPM and 12-I-ESWPB)

The emergency service water intake structure is a separate building enclosed by 3-hour-rated fire barriers containing non-fire-rated doors. A deviation is requested from providing 3-hour-rated fire doors in the exterior building walls. The A-train and B-train equipment is each provided with a separate cubicle. The exterior doors to the cubicle are installed in labyrinth shield walls. The minimum separation between the doors is more than 20 ft. On the basis of its review, the staff concludes that the separation provided between the doorways, the lack of significant fire exposure exterior to the doorways, and the non-rated-doors ensure a level of protection equivalent to that specified in Section C.5.b of the staff's guidelines.

(8) Main Steam Tunnel (Fire Zone 1-A-46-ST)

The main steam tunnel is open to the atmosphere. Redundant equipment consists of several air- and motor-operated valves. The redundant equipment is separated by less than 20 ft. The in situ fuel loading is negligible. Because of the low fuel loading and limited access to the steam tunnel, the staff concludes that the addition of 3-hour-rated fire barriers to separate the redundant valves would not greatly enhance fire protection.

(9) Emergency Service Water Screening Structure (Fire Area 5-S-BAL)

The emergency service water screening structure contains screenwash pumps and non-safety-related service water valves. A deviation is requested from providing fire detection in the area. In the event of a fire in this area, service water can be provided from the main reservoir through valves and piping located in another fire area. The combustible loading in the area is low. Because of the provision of redundant equipment in another fire area, the staff concludes that the addition of fire detectors to the emergency service water screening structure would not greatly enhance fire protection.

(10) Fire Area Boundaries

The applicants requested a deviation from providing fire dampers in ventilation ducts that penetrate certain 3-hour-fire-rated boundaries which provide separation of fire areas. The applicants' letter of February 24, 1984, lists 12 ventilation ducts that penetrate 3-hour-fire-rated concrete floors and are not provided with fire dampers.

The ventilation ducts are all installed in areas provided with partial automatic suppression and detection. Duct penetrations 25, 40, 41, and 70 are not directly covered by automatic suppression; however, the partial suppression system provided covers the fire risk in adjacent areas and will prevent a fire from spreading to the dampers.

There is no redundant safe shutdown equipment within 20 ft of the duct penetrations, either on the floor above or the floor below.

On the basis of its review, the staff concludes that this configuration provides a level of safety equivalent to providing 20 ft free of intervening combustible material with suppression and detection. The addition of 3-hour-rated fire dampers in the 12 identified ventilation ducts will, therefore, not greatly enhance fire protection.

On the basis of its review, the staff concludes that the fire protection for safe shutdown, with the approved deviations, meets the staff guidelines in Section C.5.b of BTP CMEB 9.5-1, and is, therefore, acceptable.

Control of Combustible Materials

Safety-related systems have been isolated or separated from combustible materials as much as possible. The storage of flammable liquids complies with Standard 30 of the National Fire Protection Association (NFPA 30). Compressed gases are stored outdoors.

By letter dated February 24, 1984, the applicants committed to provide hydrogen lines in safety-related areas that are either seismically designed, sleeved, or provided with excess flow valves. The staff finds this commitment acceptable.

On the basis of its review, the staff concludes that the control of combustible materials meets its guidelines in Section C.5.d of BTP CMEB 9.5-1 and is, therefore, acceptable.

Control Room

The control room complex is separated from all other areas of the plant by 3-hour-fire-rated assemblies. Peripheral rooms in the control room complex consist of offices. Each room is separated from the control room by 1-hour-fire-rated barriers. Smoke detectors that alarm and annunciate in the control room panel are provided in each room.

All cables entering the control room terminate there. No cables are routed through the control room from one area to another. There are no raised floors in the control room. There is a trench under the HVAC control, which is about 11 ft x 2 ft x 8 in. The fire loading is low, less than 200 Btu/ft^2. A suppression system is not provided. Redundant safety-related radiation monitoring cables are installed in conduits above the suspended ceiling.

The control room suspended ceiling is the aluminum luminous louver-type, eggcrate construction. A perforated duct located above the hung ceiling introduces air into the control room. The space above the hung ceiling does not contain any cable tray, only conduits.

Smoke detectors will be provided on the south side of the control room reinforced concrete ceiling, as well as below the hung ceiling.

Because of the low fuel loading and the small size of the trench, combined with the installed early-warning smoke detection system and continuous manning of the control room, the staff finds the installation of conduits in the control room ceiling, the omission of a suppression system in the trench, and the omission of a sprinkler system in the peripheral rooms acceptable deviations from Section C.7.b of BTP CMEB 9.5-1.

Ionization smoke detectors have been installed in the control room. By letter dated February 24, 1984, the applicants committed to provide smoke detectors inside the main control board. The staff finds this commitment acceptable.

On the basis of its review, the staff concludes that the control room with approved deviations meets its guidelines in Section C.7.b of BTP CMEB 9.5-1 and is, therefore, acceptable.

9.5.1.4 General Plant Guidelines

Building Design

In the SER, the staff stated that door openings in fire-rated barriers were not provided with Underwriters Laboratory (UL)-listed fire door assemblies that have ratings commensurate with the fire ratings of the walls in which they are installed. The staff considered this issue unresolved.

By letter dated November 8, 1984, the applicant committed to provide doors with a 1½-hour-or 3-hour-fire rating as determined by the test method of American Society for Testing and Materials (ASTM) Standard E-119 when conducted by a nationally recognized independent testing laboratory. This commitment applies to all door openings in fire-rated barriers, as delineated by the applicants, except where plant design requirements specify the use of special-purpose doors. Where UL-listed doors are installed, the protection of doorways in fire-rated walls complies with Section C.5.a of BTP CMEB 9.5-1 (NUREG-0800) and is, therefore, acceptable.

The doorways protected by special-purpose doors are delineated in the applicants' letter of November 8, 1984. The special-purpose doors are doors such as missile-resisting or bulletproof ones that serve functions other than solely fire protection. The staff has reviewed the design and construction of the doors and their respective locations within the plant to determine their fire-resisting capability on the basis of the relative fire hazard in the area where they are installed. The doors are constructed of heavy-weight, reinforced steel plates thicker than those used in approved fire doors. The latching mechanisms utilize multiple-point steel locking pins. If a fire occurred in the vicinity of one of the special-purpose doors, the mass of the door would provide a thermal heat sink that would require a significant fire exposure to raise the temperature of the steel to its yield point. Unequal thermal expansion of the door and its frame which could cause warping of the door will be prevented by the multiple-point steel locking pins. The locations within the plant where the special-purpose doors are installed do not contain significant amounts of combustible material that, in the staff's opinion, would provide a fire exposure capable of causing failure of the doors.

On the basis of its evaluation, the staff concludes that the applicants' special-purpose fire doors are an acceptable deivation from Section C.5.a of the fire protection guidelines contained in BTP CMEB 9.5-1.

9.5.1.7 Summary of Deviations From CMEB 9.5-1

The technical requirements of Appendix R to 10 CFR 50 and Appendix A to BTP ASB 9.5-1 have been included in BTP CMEB 9.5-1. The following deviations from the guidelines of BTP CMEB 9.5-1 were approved in the SER:

(1)	non-IEEE 383-rated cables for lighting and communications systems	BTP CMEB 9.5-1 Section C.5.e
(2)	fixed emergency lighting	BTP CMEB 9.5-1 Section C.5.g
(3)	3000-gal capacity diesel generator day tanks	BTP CMEB 9.5-1 Section C.7.i
(4)	conduits in the control room ceiling	BTP CMEB 9.5-1 Section C.7.b
(5)	lack of automatic sprinklers in the control room peripheral rooms	BTP CMEB 9.5-1 Section C.7.b
(6)	lack of automatic suppression in control room trench	BTP CMEB 9.5-1 Section C.7.b
(7)	separation of redundant cables in the switchgear room by 1-hour barriers, sprinklers, and detectors	BTP CMEB 9.5-1 Section C.7.c

On the basis of the above evaluation, the staff also concludes that the following deviations are acceptable:

(8) fire protection for safe shutdown — BTP CMEB 9.5-1 Section C.5.b

- (a) partial suppression and detection
- (b) 1-hour-rated fire barriers
- (c) intervening combustible material
- (d) component cooling water pumps
- (e) charging pumps
- (f) diesel oil storage area
- (g) emergency service water intake structure
- (h) main steam tunnel
- (i) emergency service water screening structure
- (j) fire area boundaries

(9) special-purpose doors in fire walls listed in the applicants' letter of November 8, 1984 — BTP CMEB 9.5-1 Section C.5.a

The following is the fire protection open item:

(1) alternate shutdown — BTP CMEB 9.5-1 Section C.5.c

10 STEAM AND POWER CONVERSION SYSTEM

10.4 Other Features

10.4.2 Main Condenser Evacuation System

10.4.2.1 Summary Description

In the SER, the staff expressed concerns (listed as Outstanding Item 9) about unmonitored releases of condenser discharge during startup "hogging" operations. In a letter dated December 13, 1984, the applicants stated that if the reactor has been shut down for more than 30 days, at plant startup and before tubine operation, part of the noncondensible gases can be discharged directly to the turbine building atmosphere and the balance discharged past the noble gas, iodine, and particulate monitor at the turbine building vent stack. In addition, if the reactor has been shut down for less than 30 days, at plant startup before turbine operation, the vacuum pump discharge will be routed directly to the turbine building vent stack where a noble gas, iodine, and particulate monitor surveys the releases. With turbine operation, the discharge from the mechanical vacuum pumps is directed to the turbine building vent stack without filtration. Any radioactivity exceeding the noble gas monitor setpoint will initiate an alarm in the control room, and the operators will monitor and redirect the offgas through high efficiency particulate air (HEPA)/charcoal filters, as necessary, to comply with the annual release limits of 10 CFR 20.

10.4.2.2 Evaluation Findings

During hogging operation as discussed above, the vacuum pump discharge will be monitored along the normal vacuum pump exhaust path, line 7A12-9-1 through valve 7AE-B3-1, past noble gas monitor REM-1TV-3534 and also at the turbine building vent stack monitor. With these monitors, action can be taken to close valves 7AE-B3-1 and 7AE-B9-1 on a high radiation signal, and the offgas can be rerouted through the condenser vacuum pump effluent treatment system (CVPETS). The CVPETS consists of a demister, an electrical heating coil, a HEPA filter, a 4-in. charcoal adsorber, another HEPA filter, and two 100% capacity fans in parallel.

It is the staff's position that as long as flow is directed past radiation monitor RE-1TV-3534, the applicants would have a measurement of the concentration of noble gases released during hogging operations. On the basis of these considerations, the staff finds the condenser vacuum pump offgas adequately monitored for radioactivity thus meeting the intent of Table 1 of SRP Section 11.5. To determine the release of noble gases through the hogging valve directly to the atmosphere, the applicants must have some means of correlating the radiation monitor concentration reading with the discharge flow through the hogging line. The applicants will be required to include this methodology in the Offsite Dose Calculation Manual. The staff considers this issue closed.

10.4.3 Turbine Gland Sealing System

10.4.3.2 Evaluation and Findings

In the SER, the staff identified as Confirmatory Item 26 a commitment by the applicants to monitor the turbine gland sealing steam condenser exhaust. In a letter dated July 27, 1984, the applicants confirmed a design change to redirect the condenser exhaust to the turbine building exhaust vent upstream of the vent noble gas, particulate, and iodine radiation monitor. This is acceptable to the staff.

11 RADIOACTIVE WASTE MANAGEMENT

11.4 Solid Waste Management System

11.4.2 Evaluation and Findings

In the SER, the staff requested the following additional information to resolve the confirmatory items listed below:

(1) drawings to show how filter sludges are processed (Confirmatory Item 27)

(2) a description of the solid waste processing polymer binding system (Confirmatory Item 29)

In a letter dated June 15, 1984, the applicants provided the requested information. In brief, filter sludges will not be processed through the volume reduction (VR) system. Instead, they will be sent to the waste processing decanting tanks where the excess carrying water will be decanted and drained to the floor drain treatment system. The concentrated filter sludges wil be transferred to the waste processing cement drumming stations where they will be solidified with cement in accordance with solidification procedures.

The polymer/dry salt solidification system (P/DSSS) is designed to solidify the dry salt product generated in the VR system by the application of a vinyl ester polymer product as a solidification agent. The polymer exists in a liquid state and, in combination with the promoter and catalyst, constitutes the bulk of the three-element polymer solidification agent. The P/DSSS consists of the following subsystems: (1) dry product storage and transfer system, (2) polymer fill station and storage tank, and (3) polymer/dry salt drumming station.

The staff has reviewed this information and finds it acceptable; it therefore, concludes that Confirmatory Items 27 and 29 are resolved.

In the SER, the staff also placed License Condition 4 on the applicants so that filter sludges will not be processed by the solid waste volume reduction system. This condition was imposed because the subject system was never qualified to process filter sludges. In a letter dated June 15, 1984, the applicants agreed not to process filter sludges by the volume reduction system but to process it as described above. The staff finds this position acceptable and removes the need to condition the license on this issue.

11.5 Process and Effluent Radiological Monitoring and Sampling Systems

11.5.2 Evaluation and Findings

In the SER, the staff identified the following open items:

(1) Unmonitored condenser vacuum pumps exhaust during hogging operations (Outstanding Item 9). This item is resolved in Section 10.4.2 of this supplement.

(2) Service water effluent not properly monitored (Outstanding Item 11)

Outstanding Item 11 has been resolved as a result of information supplied by the applicants in a letter dated March 22, 1984. Liquid radiation monitor REM-3500A will be located in line 7SW24-544-1 in the waste processing building before valve 7SW-B44-1 shown on FSAR Figure 9.2.1-2. Liquid radiation monitor REM-3500B will be located in line 7SW42-1-1 in the turbine building after valve 7SW-B53-1 shown on FSAR Figure 9.2.1-2. Liquid radiation monitors REM-3500A and REM-3500B are described in FSAR Sections 11.5.2.5.7 and 11.5.2.6.2, which include information on detector capability, sensitivity, and range. As stated in FSAR Section 11.5.1.3, liquid radiation monitors are provided with grab sample capability. The service water system, downstream of all inputs, will be monitored continuously before discharge to the circulating water system. The staff finds this acceptable.

13 CONDUCT OF OPERATIONS

13.2 Training

13.2.1 Licensed Operator Training Program

13.2.1.2 Licensed Operator Requalification and Replacement Training Program

13.2.1.2.1 Requalification Training

By letter dated June 1, 1984, the applicants submitted a revised requalification training program for licensed reactor operators and senior reactor operators for staff review. The staff has evaluated the revised program, and its findings are discussed below. This section supersedes Section 13.2.1.2.1, Items (1) through (3), of the SER. The requalification training program consists of the following areas:

(1) Lecture Series

The requalification training program will include preplanned lectures on a regular and continuing basis. Annual written examination results will indicate the scope and depth needed in the following areas:

(a) theory and principles of operations
(b) general and specific plant operating characteristics
(c) plant instrumentation and control systems
(d) normal and abnormal procedures and emergency instructions
(e) radiation control and safety
(f) Technical Specifications
(g) chemistry
(h) quality assurance responsibilities
(i) heat transfer, fluid flow, and thermodynamics
(j) mitigating core damage
(k) pressurized thermal shock
(l) significant system modifications or installation
(m) significant plant operating experience, events, procedures, and issuance of new procedures

(2) Reactivity Control Manipulations and Plant Evolutions

The applicants have indicated that, during the 2-year term of license, each individual will perform or participate in those reactivity manipulations and plant evolutions listed in the requalification program. The staff has reviewed the list of control manipulations and plant evolutions and finds that the applicants' commitment of control manipulations and plant evolutions required for licensed operators does comply with the requirement as specified in Appendix A to 10 CFR 55 and in Enclosure 4 of H. R. Denton's letter of March 28, 1980, to all power reactor applicants and licensees.

(3) Simulator Training

The applicants have indicated that licensed operators and NRC-certified individuals are required to receive approximately 64 hours of training on a simulator each year. The simulator used will have an arrangement of instrumentation and controls similar to that at Shearon Harris and will reproduce the general operating characteristics of Shearon Harris. The staff finds that the applicants have committed to the requirement, as specified in Enclosure 1 of H. R. Denton's letter of March 28, 1980, which requires all licensed operators to participate in a simulator training program as part of the requalification program.

(4) On-the-Job Training

This program, which is designed to maintain operator skills for coping with normal, abnormal, and emergency situations, consists of individual reviews, instructional sessions, and, where applicable, a walkthrough of controls and instrumentation. The major areas to be covered in this program are vital information or alarm settings, safety limits, abnormal condition symptoms for operation, operating sequences, and emergency immediate-action steps. In addition, the Technical Specifications, along with flow, logic, and functional diagrams, will be reviewed.

(5) Evaluation

As required in Appendix A to 10 CFR 55, the evaluation program for licensed personnel will include the following:

(a) Annual Written Examination

An annual written examination, comparable in scope and degree of difficulty to an NRC examination, will be administered to all licensed personnel. An individual who receives an overall grade of less than 80% on the annual examination, or receives a grade of less than 70% in any category of the annual examination, will be removed from licensed duties and placed in an accelerated requalification program.

(b) Annual Oral Examination

The simulator training staff will observe and evaluate the performance of licensed personnel under simulated abnormal and emergency conditions during simulator retraining. In addition, walkthrough oral examinations will be conducted either at the simulator or at the operating plant on an annual basis for selected individuals. Any individual given an unsatisfactory evaluation will require accelerated requalification.

On the basis of its review, the staff finds that the applicants' requalification training program conforms to the requirements of 10 CFR 50 and Appendix A of 10 CFR 55 and follows the guidance in RG 1.8. In addition, the program conforms to the requirements outlined in a letter from H. R. Denton to all power reactor applicants and licensees, dated March 28, 1980. Therefore, the staff concludes that the applicants' requalification training program for licensed reactor operators and senior reactor operators is acceptable.

13.3 Emergency Preparedness

13.3.1 Introduction

The applicants' emergency plan was reviewed in the SER, and revisions to the plan and procedures were reviewed in Supplement No. 1 to the SER. The applicants further revised the emergency plan and procedures on the basis of comments in Supplement No. 1 and in later submittals by the applicants and the emergency preparedness appraisal conducted at Shearon Harris during March 1985.

This supplement to the SER contains the review of those portions of the emergency plan and procedures that previously were found to need correction and/or clarification. Section 13.3.3 contains the conclusions of the staff based on the previous reviews and this review.

13.3.2 Evaluation of the Emergency Plan

13.3.2.4 Emergency Classification System

The applicants submitted a draft of the symptomatic based emergency action level (EAL) scheme on September 12, 1984, and a revised set of EALs that was evaluated during the emergency preparedness appraisal in March 1985.

The applicants have generated three sets of EAL criteria and related emergency classifications. These are as follow:

(1) Internally Generated EALs

These EALs can be entered from corresponding symptomatic emergency operating procedures (EOPs). These EALs can be related directly to the status of critical safety functions and the failure or challenge to the fission product barriers.

(2) Externally Generated EALs

These EALs address the initiating conditions in NUREG-0654, Appendix 1, that indirectly affect fission product barriers (e.g., earthquake, sabotage, fire).

(3) Unusual Event

These EALs are in tabular form and address the initiating conditions in NUREG-0654 under the unusual event category.

This approach has the benefit of being comprehensive with respect to existing event-oriented guidance, allowing full integration of EOPs and EALs where loss or challenge to critical safety functions and fission product barriers exists, and avoiding the mixing of minor and indirect initiating conditions with those directly affecting critical safety functions and fission product barriers. The staff considers this approach satisfactory, the EAL sets are adequate, and this matter is resolved.

13.3.2.7 Public Information

On August 14, 1984, the applicants presented a draft of the Public Information Brochure and subsequent revisions thereto. By letter dated April 30, 1985, the applicants committed to submit a final draft of the brochure in May 1985 and a completed brochure in June 1985 for final Federal Emergency Management Agency (FEMA) and NRC review. The results of the FEMA/NRC review will be published in a future supplement to the SER.

13.3.2.10 Protective Response

An evacuation time study for the Shearon Harris emergency planning zone was submitted on December 29, 1983. A review of this study revealed that the evacuation plan and the time estimate appear adequate and satisfy the requirement of 10 CFR 50, Appendix E, and the criteria of NUREG-0654/FEMA-REP-1, Part II, Section J, Paragraph 10, and Appendix 4 thereof.

13.3.3 Conclusions

On the basis of its review of the Shearon Harris Emergency Plan and revisions previously reported, the staff concludes that, upon satisfactory review of the Public Information Brochure as identified in 13.3.2.7 of this report, the Shearon Harris Emergency Plan will provide an adequate basis for an acceptable state of emergency preparedness.

After reviewing the finding and determinations made by FEMA on the adequacy of state and local emergency response plans, a future supplement to the SER will include the staff's 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.

15 ACCIDENT ANALYSES

15.1 Increase in Heat Removal by the Secondary System

15.1.6 Boron Injection Tank Concentration Reduction

The applicants in their submittals dated September 21, 1984, and January 29, 1985, have requested staff approval for reducing the boron concentration in the boron injection tank (BIT) to zero ppm. Reducing the boron concentration to zero ppm will eliminate the need for heat tracing of the system resulting in considerable reduction in maintenance. Additionally, Technical Specifications concerning BIT boron concentration, temperatures, and associated surveillance/maintenance will be eliminated.

The BIT is a component of the safety injection system whose function is to provide concentrated boric acid to the reactor coolant system (RCS) to mitigate the consequences of postulated steamline break accidents. The accidents that serve as the steamline break licensing basis and that define the existing requirements on the minimum BIT boron concentration are (1) the double-ended rupture of a main steamline for which the radiation releases must remain within the requirements of 10 CFR 100 and (2) the failed-open single steam generator relief, safety, or turbine bypass valve for which the radiation releases must remain within the requirements of 10 CFR 20.

The applicants' analysis assumes the BIT remains installed without heat tracing and with boric acid concentration reduced to zero ppm in order to provide the most limiting case for the analysis.

In the analysis the system parameters (RCS pressure, temperature, steam flow, core boron concentration, and core power) were calculated by using the LOFTRAN system transient analysis code. This code includes models of the reactor core, protection system, and engineered safety features system. The changes in the safety injection system's initial concentration and temperature were introduced into the analysis.

For the double-ended rupture, which is the most limiting event, the plant is assumed to be at hot zero power and at the minimum required shutdown margin, in the presence of a large end-of-life moderator coefficient of reactivity, the most reactive rod cluster control assembly in the fully withdrawn position, and offsite power available. The results indicate an increase in power until boron from the safety injection system reaches the core to begin the offset of the positive reactivity insertion caused by the cooldown. A comparison of the plant parameters' response with the current FSAR results indicates only small changes, with the exception of the initial core power, which is higher with BIT boron concentration at zero ppm, because, with the BIT assumed at zero ppm, the BIT acts as a dilution volume for the borated water being injected from the refueling water storage tank (RWST), thereby further delaying the effectiveness of the RWST. However, the departure from nucleate boiling (DNB) analyses performed using the same analysis methods and procedures as those used for the FSAR cases indicate that the DNB design basis is met and no fuel failures are anticipated. The reduction of boron in the BIT will result in a slightly higher containment pressure (39.1 psig) and temperature (379°F) during a main steamline break. These values are within the containment design values of 45 psig and 380°F and within the containment temperatures and pressures to which safety-related equipment in containment must be environmentally qualified.

For the failed-open single steam generator relief, safety, or turbine bypass valve event, the results indicate that the DNB design basis is also met and no fuel failure is predicted. These results are consistent with the results of the most limiting event of the double-ended rupture where no violation of the DNB design basis was calculated.

On the basis of its evaluation of the applicants' submittals, the staff has concluded that reduction of the boron concentration in the BIT to zero ppm will not violate the Commission's acceptance criteria during postulated steamline rupture events. The applicants have demonstrated that, for the limiting double-ended pipe rupture event, the DNB calculations predict no fuel failure, thus meeting the requirements of 10 CFR 100. The calculated peak containment pressure and temperatures are within the containment design criteria and environmental qualification values for safety-related equipment inside containment. For the steam generator failed-open dump valve event, the DNB calculations predict no fuel failure, thus meeting the requirements of GDC 10. The staff, therefore, concludes that the reduction of the boron concentration in the BIT is acceptable. Moreover, this section supersedes SER Sections 15.1.4, "Inadvertent Opening of a Steam Generator Relief Valve or Safety Valve," and 15.1.5, "Steamline Rupture."

15.6 Decreases in Reactor Coolant Inventory

15.6.3 Steam Generator Tube Rupture

As a result of negotiations between all parties to the Shearon Harris Unit 1 hearing proceedings, it was agreed by all the parties and approved by the Atomic Safety and Licensing Board that the license be conditioned as follows:

> Prior to startup, following the first refueling outage, applicants shall submit for NRC review and receive approval of a steam generator tube rupture analysis, including the assumed operator actions, which demonstrates that the consequences of the design basis steam generator tube rupture event for the Shearon Harris Plant are less than the acceptance criteria specified in the Standard Review Plan, NUREG-0800, at § 15.6.3 Subparts II(1) and (2) for calculated doses from radiological releases. In preparing their analysis applicants will not assume that operators will complete corrective actions within the first thirty minutes after a steam generator tube rupture.

By virtue of the applicants being members of the Westinghouse Owners Subgroup associated with the resolution of steam generator tube rupture licensing issues, the staff considers this issue to be reclassified from open to confirmatory. Analyses will be conducted to satisfy the above cited license conditions. These analyses will evaluate the probability and consequences of steam generator overfill, the worst active single failure, the radiological consequences, and the necessary operator action times. These evaluations will include plant-specific analyses of the effects of static and dynamic loads on steamline components and piping when subjected to steam generator overfill.

19 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Section 19 of Supplement No. 1 to the SER inadvertently failed to address one recommendation contained in the letter from the Advisory Committee on Reactor Safeguards (ACRS) dated January 16, 1984, pertaining to the Shearon Harris plant. The recommendation dealt with turbine missiles and is discussed below.

Turbine Missiles

The ACRS noted that because of the nonoptimum orientation of the turbine relative to vital components in the Shearon Harris plant, it recommended that a structured test program for evaluating the overspeed protection of the turbine be prepared and submitted for the NRC staff to review and approve before full-power operation. The staff requires the applicants to submit the above-cited test program for the staff's review and approval before full-power operation.

APPENDIX A

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

September 21, 1983	Letter from applicants concerning leakage outside containment.
November 4, 1983	Letter from applicants concerning instrument air quality.
February 24, 1984	Letter from applicants concerning fire protection.
April 18, 1984	Letter from applicants forwarding information on inadequate core cooling instrumentation.
April 19, 1984	Letter from Texas A&M University System advising that time estimate study is adequate.
April 19, 1984	Letter from applicants transmitting information on design of the fuel handling building retaining wall.
April 20, 1984	Letter from applicants transmitting updated construction status report.
April 26, 1984	Generic Letter 84-10 - Administration of Operating Tests Prior to Initial Criticality (10 CFR 50.25).
April 27, 1984	Meeting with applicants to discuss status of review of fire protection and internally generated missile issues.
April 27, 1984	Letter from applicants transmitting list of vendors who will be conducting equipment qualification testing.
April 30, 1984	Generic Letter 84-12 - Compliance With 10 CFR 61 and Implementation of Radiological Effluent Technical Specifications (RETS) and Attendant Process Control Program (PCP).
May 3, 1984	Generic Letter 84-13 - Technical Specifications for Snubbers.
May 7, 1984	Board Notification 84-098 - Additional Reports Submitted to NRC by TDI Owners Group.
May 8, 1984	Generic Letter 84-09 - Recombiner Capability Requirements of 10 CFR 50.44(c)(3)(ii).
May 11, 1984	Generic Letter 84-14 - Requalification Training Program.

May 15, 1984	Letter from applicants forwarding information on essential services chilled water system.
May 29, 1984	Letter from applicants transmitting additional information on equipment qualification program.
June 1, 1984	Letter from applicants transmitting "Licensed Operator Requalification Program."
June 7, 1984	Letter from applicants transmitting information on functional capability of Class 1 piping.
June 11, 1984	Issuance of Amendment 3 to Construction Permit to reflect name change of co-owner to North Carolina Eastern Municipal Power Agency.
June 12, 1984	Letter from applicants transmitting safe shutdown analysis summary.
June 12, 1984	Letter from applicants transmitting Amendment 13 to Final Safety Analysis Report (FSAR).
June 13, 1984	Letter from applicants transmitting Amendment 14 to FSAR.
June 15, 1984	Letter from applicants forwarding information for Confirmatory Items 27 and 29 and Licensing Condition 4.
June 15, 1984	Letter from applicants transmitting information on Preservice/Inservice Inspection Program.
June 20, 1984	Board Notification 84-104 - Westinghouse ECCS Actuation Logic Review.
June 20, 1984	Letter from applicants forwarding revision to general information to reflect cancellation of Unit 2.
June 21, 1984	Letter from applicants transmitting information on periodic testing of instrument air quality.
June 22, 1984	Meeting between NRC and its consultants and TDI Owners Group and its consultants.
June 27, 1984	Generic Letter 84-16 - Adequacy of On-Shift Operating Experience for Near-Term OL Applicants.
June 28, 1984	Letter from applicants forwarding "Report on Operability of Pressuizer Safety and Relief Valves, PORVs and Electrical Motor Operated Block Valves."
June 28, 1984	Letter from applicants forwarding additional information on load sequencer reliability.
June 29, 1984	Letter from applicants transmitting Amendment 15 to FSAR (reflects cancellation of Unit 2).

July 2, 1984	Generic Letter 84-15 - Proposed Staff Actions To Improve and Maintain Diesel Generator Reliability.
July 3, 1984	Meeting with applicants to discuss environmental equipment qualification issues.
July 3, 1984	Generic Letter 84-17 - Annual Meeting To Discuss Recent Developments Concerning Operator Training, Qualifications and Examinations.
July 3, 1984	Letter from applicants transmitting additional information concerning postaccident sampling system (PASS).
July 5, 1984	Letter from applicants regarding insulation on electrical containment penetration pigtails.
July 6, 1984	Letter from applicants transmitting correction to July 3, 1984, letter on PASS.
July 6, 1984	Generic Letter 84-18 - Filing of Applications for Licenses and Amendments.
July 9, 1984	Issuance of Supplement No. 1 to SER.
July 9, 1984	Letter from applicants transmitting additional information concerning condenser discharge during startup (hogging) operations.
July 11, 1984	Letter from applicants forwarding additional information regarding leak reduction program.
July 13, 1984	Letter to applicants transmitting draft design verification audit plan for safety parameter display system.
July 16, 1984	Letter from applicants transmitting emergency planning public information brochure.
July 16, 1984	Letter from applicants transmitting information on fire protection program.
July 17, 1984	Letter from applicants transmitting seismic report for second quarter 1984.
July 24, 1984	Letter from Duke Power Company transmitting "TDI Diesel Generator Owners Group EDG Rocket Arm Capscrew Stress Analysis."
July 24, 1984	Meeting with applicants to discuss resolution and schedule for resolution of steam generator tube rupture issue.
July 27, 1984	Letter from applicants transmitting revision to emergency planning public information brochure.

July 27, 1984	Letter from applicants transmitting "Class 1E Equipment Environmental Qualification Master List."
July 27, 1984	Letter from applicants transmitting additional information concerning turbine building vent radiation monitors.
July 27, 1984	Letter from applicants concerning training of cold license applicants.
July 27, 1984	Letter from applicants transmitting Amendment 16 to FSAR.
July 27, 1984	Letter from Duke Power Company transmitting "TDI Owners Group Report on the Emergency Diesel Generator Auxiliary Module Control Wiring and Termination Review."
July 30, 1984	Letter from applicants transmitting "pen and ink" version of Technical Specifications.
August 1, 1984	Letter to applicants transmitting comments on Design Evaluation Report Program Plan elements.
August 1, 1984	Letter from Duke Power Company transmitting "TDI Diesel Generator Report on the Elliot Model 65G Turbocharger Used on TDI DSRV-12-4 and DSRV-20-4 Emergency Diesel Generators."
August 10, 1984	Letter from Duke Power Company transmitting "TDI Diesel Generator Report on the Design Review of Engine Base and Bearing Caps for Transamerica Delaval DSRV-16 Diesel Engines."
August 14, 1984	Letter from applicants concerning emergency preparedness.
August 14, 1984	Letter from Duke Power Company transmitting "TDI Diesel Generator Report on the Design Review of Elliott Model 90G Turbocharger Used on Transamerica Delaval DSR-48 and DSRV-16 Emergency Diesel Generator Sets."
August 17, 1984	Meeting with applicants to discuss functional capability of Class 1 auxiliary piping systems.
August 20, 1984	Generic Letter 84-20 - Scheduling Guidance for Licensee Submittals of Reloads That Involve Unreviewed Safety Questions.
August 24, 1984	Letter from applicants transmitting responses to questions raised by Equipment Qualification Branch.
September 6, 1984	Letter from applicants forwarding "Functional Capability of ASME Class 1 Auxiliary Piping Systems," Revision 1.
September 10, 1984	Letter from applicants clarifying discrepancy in cable specification sheets.

September 11, 1984	Meeting with applicants to review seismic and dynamic qualifications of safety-related equipment.
September 12, 1984	Letter from applicants transmitting Revision 3 to Emergency Plan.
September 18, 1984	Letter from applicants transmitting emergency operating procedures generation package.
September 21, 1984	Letter from applicants transmitting results of Westinghouse report on boron injection tank concentration/elimination study.
September 26, 1984	Meeting with applicants to discuss safety parameter display system audit plan.
September 26, 1984	Letter from applicants transmitting additional information regarding interaction of non-Category I piping with Category I piping.
September 26, 1984	Letter from applicants transmitting additional information regarding reactor coolant system overpressure protection.
October 1, 1984	Letter to applicants transmitting request for additional information.
October 3, 1984	Letter from applicants providing clarification of measuring and test equipment versus process instrument calibration.
October 10, 1984	Letter from applicants transmitting information concerning fire protection.
October 16, 1984	Generic Letter 84-21 - Long-Term Low Power Operation in PWRs.
October 23, 1984	Letter from applicants transmitting July-September Quarterly Data Report.
October 25, 1984	Letter from applicants forwarding information on flood protection.
October 25, 1984	Letter from applicants transmitting information on internally generated missiles.
November 1, 1984	Letter from applicants transmitting Amendment 17 to FSAR.
November 2, 1984	Meeting with applicants concerning preservice inspection testing.
November 7, 1984	Letter from applicants forwarding information concerning minimum wall thickness.
November 7, 1984	Letter from applicants transmitting information on excore axial power distribution monitor.

November 8, 1984	Letter from applicants transmitting information on fire protection.
November 8, 1984	Letter from applicants regarding reactor trip setpoint methodology.
November 21, 1984	Letter from applicants forwarding response to SER Confirmatory Item 11.
November 27, 1984	Letter to applicants transmitting draft Technical Evaluation Report on Control of Heavy Loads.
November 30, 1984	Letter from applicants regarding elimination of postulated pipe breaks in primary main loop and arbitrary intermediate pipe breaks in Class 1, 2, and 3 pipes, and advising of plan to request exemption from General Design Criterion 4.
November 30, 1984	Letter from applicants transmitting Amendment 18 to FSAR.
December 3, 1984	Letter from applicants transmitting remaining master list of Class 1E electrical equipment.
December 4, 1984	Letter from Duke Power Company regarding November 6, 1984, letter on crank shaft forging process and tensile specimens.
December 5, 1984	Letter to applicants transmitting request for additional information regarding environmental qualification program audit schedule.
December 10, 1984	Letter to applicants transmitting request for additional information regarding consideration of superheat in main steamline break analysis.
December 13, 1984	Letter from applicants transmitting additional information concerning condenser discharge during startup operations.
December 20, 1984	Letter from applicants concerning emergency diesel generators.
December 27, 1984	Generic Letter 84-24 - Certification of Compliance to 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety of Nuclear Power Plants."
January 9, 1985	Generic Letter 85-01 - Fire Protection Policy Steering Committee Report.
January 11, 1985	Letter from applicants transmitting additional information regarding detailed control room design review.
January 14, 1985	Letter from applicants transmitting "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as a Structural Design Basis for Shearon Harris Unit 1," WCAP-10699.

January 24, 1985	Letter to applicants transmitting request for additional information.
January 28, 1985	Letter from applicants transmitting fourth quarter seismic report.
Jaunuary 29, 1985	Generic Letter 85-04 - Operator Licensing Examinations.
January 29, 1985	Letter from applicants concerning boron injection tank concentration reduction.
January 31, 1985	Letter from applicants transmitting Amendment 19 to FSAR.
January 31, 1985	Generic Letter 85-05 - Inadvertent Boron Dilution Events.
February 1, 1985	Letter from applicants requesting meeting to discuss need for early approval of exemption request which will allow removal of primary loop whip restraints.
February 8, 1985	Letter from applicants transmitting Revision 1 to "North Carolina Emergency Response Plan in Support of the Shearon Harris Nuclear Power Plant (SHNPP)."
February 11, 1985	Letter from applicants transmitting "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment, Master List."
February 11, 1985	Letter to applicants transmitting request for additional information on containment purge and vent valve operability.
February 11, 1985	Board Notification 85-011 - Shearon Harris - Construction Appraisal Team Inspection 50-400/84-41.
February 13, 1985	Letter from applicants forwarding additional information regarding elimination of postulated arbitrary intermediate breaks in the main feedwater system.
February 18, 1985	Letter from applicants regarding emergency operating procedures vendor review.
February 25, 1985	Letter from applicants transmitting information on concrete curing requirements.
March 5, 1985	Letter from applicants transmitting information on Preservice Inspection Program.
March 8, 1985	Letter from applicants advising of revised dates: Operating License needed by March 1, 1986, to meet commercial operation date of September 1986.
March 21, 1985	Letter to applicants transmitting request for additional information.
March 21-22, 1985	Meeting with applicants on various issues.

March 25, 1985	Letter to applicants transmitting request for additional information following preliminary staff review of response to Generic Letter 83-28.
March 28, 1985	Letter from applicants transmitting additional information concerning Phase I of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."
April 4, 1985	Letter to applicants transmitting request for additional information regarding the Offsite Dose Calculation Manual.
April 5, 1985	Letter from applicants transmitting additional information for SER open items on instrument air system, internally generated missiles, and fire protection.
April 9, 1985	Letter from applicants transmitting additional information concerning control room design review.
April 15, 1985	Letter to applicants forwarding results and conclusions of the integrated design inspection conducted by Office of Inspection and Enforcement.
April 16, 1985	Generic Letter 85-06 - Quality Assurance Guidance for ATWS Equipment not Safety-Related.
April 17, 1985	Generic Letter 85-02 - Recommended Actions Stemming From NRC Integrated Program for Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity.
April 19, 1985	Letter from applicants transmitting information on fire protection.
April 19, 1985	Letter from applicants forwarding revised page for January 14, 1985, submittal of elimination of postulated pipe breaks in the reactor coolant system primary loop.
April 22, 1985	Letter from applicants transmitting additional information concerning functional capability of Class 1 piping.
April 23, 1985	Letter from applicants transmitting updated copy of "pen and ink" version of Technical Specifications.
April 30, 1985	Letter from applicants transmitting quarterly report on seismic monitoring program.
April 30, 1985	Letter from applicants transmitting additional information on emergency preparedness.
May 2, 1985	Generic Letter 85-07 - Implementation of Integrated Schedules for Plant Modifications.
May 8, 1985	Letter to applicants transmitting for comment the notice of the Caseload Forecast Panel visit to be held June 18-19, 1985.

May 9, 1985 Letter from applicants concerning preservice inspection of Class 2 and 3 components.

APPENDIX B

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Carolina Power and Light Company, "Shearon Harris Nuclear Power Plant Final Safety Analysis Report," Docket No. 50-400.

Code of Federal Regulations, Title 10, "Energy," U.S. Government Printing Office, Washington, D.C. (contains general design criteria).

Denton, H. R., NRC, letter to all power reactor applicants and licensees, "Qualifications of Reactor Operators," Mar. 28, 1980.

Department of the Navy, Naval Facilities Engineering Command, NAVFAC DM-7.2, "Foundations and Earth Structures," May 1980.

Eisenhut, D. G., NRC, letter to all operating nuclear power plants, "Radioactive Release at North Anna Unit 1 and Lessons Learned," Oct. 17, 1979.

U.S. Nuclear Regulatory Commission, NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," various revisions.

---, NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36," July 1980.

---, NUREG-0654/FEMA-REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Rev. 1, Nov. 1980.

---, NUREG-0737, "Clarification of TMI Action Plan Requirements," Nov. 1980.

---, NUREG-0800, "Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," July 1981 (contains branch technical positions).

---, NUREG-1014, "Safety Evaluation Report Related to the D4/D5/E Steam Generator Design Modification," Oct. 1983.

APPENDIX D

ACRONYMS AND INITIALISMS

ACRS	Advisory Committee on Reactor Safeguards
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient(s) without scram
BIT	boron injection tank
BOP	balance of plant
BTP	branch technical position
CCW	component cooling water
CFR	Code of Federal Regulations
CHR	containment heat removal
CVPETS	condenser vacuum pump effluent treatment system
DBE	design-basis event
DNB	departure from nucleate boiling
DOE	Department of Energy
EAL	emergency action level
ECC	emergency core cooling
ECCS	emergency core cooling system
EOP	emergency operating procedure
ESCWS	essential services chilled water system
FEMA	Federal Emergency Management Agency
FSAR	Final Safety Analysis Report
GDC	general design criterion(a)
HEPA	high efficiency particulate air
HVAC	heating, ventilation, and air conditioning
ICC	inadequate core cooling
IEEE	Institute of Electrical and Electronics Engineers
ISA	Instrument Society of America
LOCA	loss-of-coolant accident
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
NSSS	nuclear steam supply system
PASS	postaccident sampling system
PCP	process control program
P/DSSS	polymer/dry salt solidification system
PORV	power-operated relief valve
PSI	preservice inspection
PWR	pressurized water reactor
RAB	reactor auxiliary building
RCP	reactor coolant pump
RCS	reactor coolant system
RETS	Radiological Effluent Technical Specifications
RG	regulatory guide
RHR	residual heat removal
RPV	reactor pressure vessel

RWST	refueling water storage tank
SER	safety evaluation report
SRP	Standard Review Plan
SSC	systems, structures, and components
TDI	Transamerica Delaval, Inc.
TMI	Three Mile Island
UL	Underwriters Laboratory
VR	volume reduction

APPENDIX E

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