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Docket Nos: 50-400
and 50-401

Mr. J. A. Jones
Vice Chairman
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Jones:

Subject: Request for Additional Information - Shearon Harris FSAR

As a result of the staff's review of the Shearon Harris FSAR, we have identified a number of areas where we require additional information. The areas include geology, seismology, corrosion engineering, reactor systems, radiological assessment and thermal-hydraulic performance of the core. The specific requests for information in these areas are listed in the Enclosure.

We request that you provide responses to the enclosed request and amend your FSAR accordingly. In order to maintain the schedule for the review of the FSAR, the responses should be provided by August 2, 1982. If you cannot meet this date, inform us of the date you plan to meet so that we may determine whether the review schedule needs to be revised.

Please advise us if you have any questions regarding this request.

Sincerely,

Original signed by
Frank J. Miraglia

Frank J. Miraglia, Chief
Licensing Branch No. 3
Division of Licensing

Enclosure:
Request for Additional
Information

cc w/enclosure:
See next page

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OFFICE	DL:LB#3	DL:LB#3					
SURNAME	ELicitra:ph	FMiraglia					
DATE	4/26/82	4/26/82					

Mr. J. A. Jones
Vice Chairman
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

George F. Trowbridge, Esq.
Shaw, Pittman, Potts &
Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

Richard E. Jones, Esq.
Associate General Counsel
Carolina Power & Light Company
411 Fayetteville Street Mall
Raleigh, North Carolina 27602

Thomas S. Erwin, Esq.
115 W. Morgan Street
Raleigh, North Carolina 27602

M. David Gordon
Attorney Associate General
State of North Carolina
P. O. Box 629
Raleigh, North Carolina 27602

George Maxwell
Resident Inspector/Harris NPS
c/o U. S. Nuclear Regulatory Commission
Route 1, Box 315B
New Hill, North Carolina 27562

Charles D. Barham, Jr.
Vice President and Senior Counsel
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

John Runkle, Executive Coordinator
Conservation Council of North Carolina
307 Granville Rd.
Chapel Hill, North Carolina 27514

Wells Eddleman
325 E. Trinity Avenue
Durham, North Carolina 27701

George Jackson, Secretary
Environmental Law Project
School of Law, 064-A
University of North Carolina
Chapel Hill, North Carolina 27514

Dr. Phyllis Lotchin
108 Bridle Run
Chapel Hill, North Carolina 27514

Mr. Travis Payne, Esq.
723 W. Johnson Street
P. O. Box 12643
Raleigh, North Carolina 27605

Daniel F. Read, President
CHANGE
P. O. Box 524
Chapel Hill, N.C. 27514

Daniel F. Read
100-B Stinson St.
Chapel Hill, N.C. 27514

Patricia T. Newman, Co-Coordinator
Slater E. Newman, Co-Coordinator
Citizens Against Nuclear Power
2309 Weymouth Ct.
Raleigh, N.C. 27612

Richard D. Wilson, M.D.
725 Hunter St.
Apex, N.C. 27502

ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION

RESULTING FROM REVIEW OF

SHEARON HARRIS FSAR

DOCKET NOS.: 50-400 & 50-401

230.0 GEOSCIENCES - SEISMOLOGY

230.1 (2.5.2.1) Update Table 2.5.2-1 to include earthquakes since 1977.

230.2 (2.5.2.1) Provide a map showing earthquake epicenters and seismic zones within 200 miles of the site. The map should be larger scale than Figure 2.5.2-2. (See requirements in Standard Review Plan, NUREG-0800).

230.3 (2.5.2.1.2) The staff's review of the SSE will consider the possibility of seismicity associated with the reservoirs at the Harris site. Assuming that the reservoirs could trigger earthquakes, what is the largest earthquake that could be triggered without exceeding the SSE? Consider the depth of such events, their distance to the plant and any possible associations with geologic structures. What is the closest distance magnitude 3, 4, and 5 earthquakes could occur and still not exceed the SSE at the plant site? For your information the NRC consultant, Lawrence Livermore National Laboratory, is currently performing studies aimed at estimating site specific spectra for shallow near surface reservoir-induced earthquakes at the Harris site.

230.4 (2.5.2.6) Current staff practice is to evaluate the SSE by comparison to site specific response spectra developed by performing statistical analyses on strong motion records for sites with similar foundation conditions at less than 25 km from events with magnitudes within one-half unit of the maximum magnitude. Estimate the magnitude of the maximum random earthquake near the Harris site.

(1) Compare the SSE to site specific spectra shown in NUREG/CR-1582, Vol. 4 or

(2) Compare the SSE to your own estimates of site specific spectra. Choose events that are within one-half unit of the magnitude chosen for the maximum earthquake. Select response spectra from accelerograms for recording sites less than 25 km from the source and with foundation conditions similar to the Harris site. Do not scale or normalize the response spectra. For the data set compute 50th and 84th percentile response spectra assuming the spectral ordinates are log normally distributed. On a plot similar to FSAR Figure 2.5.2-12 compare these spectra to the SSE.

231.0 GEOSCIENCES - GEOLOGY

231.2 (2.5.0) (2.5.1) (2.5E)

(1) The FSAR states on page 2.5.0-9 under the heading Site Fault, "On January 6, 1976, Carolina Power and Light Company was formally notified by the NRC that the fault discovered at the Shearon Harris Nuclear Power Plant is not a capable fault as defined in Appendix A to 10 CFR Part 100."

Also, page 2.5.1-16 in the FSAR states, under the heading Site Structural Geology, "A minor high angle fault was discovered in the foundation of the plant during excavation. This fault was subjected to an intensive investigation which led to the conclusion with which the NRC concurred, that the fault is not capable." Is the same fault being referred to in both quotes? If so, it is not clear how or where the formal notification was provided by the staff. The publication containing the notification should be cited for clarification.

(2) Again the FSAR states on page 2.5.0-9 under the heading Main Dam Faults, "Because the small amount of movement along these faults took place prior to deformation-mineralization which occurred more than 225 million years ago, the faults are not considered to be capable faults as defined in Appendix A to 10 CFR Part 100. The NRC concurred with this conclusion based on detailed reports which were submitted and on field inspections by their geological staff. Reports on these small faults are catalogued in the Foundation Report, Appendix 2.5E." The publication which contains the NRC concurrence should be cited for clarification.

231.3 (2.5-3) On page 2.5.3-20 in Section 2.5.3.3 of the FSAR reference 2.5.3-15 is given for a letter to CP&L from NRC which concurs with the Shearon Harris Fault Investigation Report. This reference number does not appear in the list of references for Section 2.5. Explain.

231.4 (2.5E) The following data should be added to Table 2.5E-1, page 2.5E-18.

Date	Personnel	Items Inspected
Sept. 18-21, 1979	J. R. Harris	Main Dam: Foundation and geology mapping between Sta 12+50 and Sta 15+90
Oct. 9, 1980	J. R. Harris	Main Dam: Spillway faults J1 and J2 between Sta 14+45 and Sta 14+85

231.5 (2.5E) The listing of features and faults with location, reporting date, and NRC inspection date presented in Table 2.5E-2, Main Dam Features and Faults Discovered to Date, corresponds with information reported to Region II and observations by Region II inspectors except for the following feature/fault change and addition:

(1) Page 2.5E-19, Change Fault A5 to Gneiss Layer 63 and Shistose Zone 82.

(2) Page 2.5E-20, add:

Feature	Location	Report to NRC Initial	Report to NRC Written	Report to NRC Inspection
Fault J1 & J2	Spillway Sta 14+47 to Sta 14+57	10/08/80	10/27/80	10/09/80

231.6 (2.5E) The following addition should be made to Table 2.5E-3, Special Reports Main Dam, page 2.5E-21.

(1) "Geologic Features in the Main Dam Spillway, Spillway Faults J1 and J2," October 27, 1980.

231.7 (2.5E) Precambrian and early Paleozoic granitic, gneissic and shistose units and features shown on Figure 3, Geologic Plan of Main Dam Diversion Conduit and Figure 4, Sheets 1 to 3, Geologic Plans and Main Dam Core Trench, correspond to observations made during site inspections. The following deficiencies were noted in the above listed figures.

(1) Fault F3, which intersects the centerline of the diversion conduit at Sta 5+50 and terminates at a point just north of the centerline, is not identified on Figure 3, Geologic Plan of the Main Dam Diversion Conduit.

(2) Dip symbols of faults A2, A3, E, F and F2 appearing on Figure 4, Geologic Plans for Main Dam Core Trench, Sheet 1 are incorrect. Symbols as shown indicate these features are joints.

(3) The orientation of fault C2 is not shown on Figure 4, Sheet 1.

231.8 (2.5E) The following figures were omitted from the review package and were not reviewed.

(1) Figure 2, Geologic Plans and Section of Plant Excavation, Sheet 9

(2) Figure 5, Geologic Plan of Main Dam Spillway Foundation

(3) Figure 6, Geologic Plan of Main Dam Embankment Foundation.

282.0 CHEMICAL ENGINEERING - CORROSION

282.1 (9.1.2) The information you have provided is insufficient for us to evaluate the compatibility and chemical stability of materials wetted by the pool water. Provide the proposed water chemistry limits for the spent fuel storage pool.

282.2 (10.3.5) (10.4.6) The information you have provided is insufficient for us to complete our evaluation of the secondary water chemistry control program.

Branch Technical Position MTEB 5-3 describes an acceptable way of monitoring secondary side water chemistry in PWR steam generators. The following additional information is required to complete our evaluation of the Shearon Harris program in terms of this BTP. The numbers in parenthesis refer to the relevant paragraph in MTEB BTP 5-3.

(1) Describe the method used to confirm the "metal clean" state of surfaces in the steam generators after hot functional testing. (II-2).

(2) Define in terms of percent rated thermal power and approximate temperature range the terms covered by the chemistry specifications for each column of Table 10.3.5-1. (II-3a).

(3) Identify the sampling schedule for the critical parameters during each mode of operation. (II-3b).

(4) Identify the composition, quantities, and proposed addition rates of additives. (II-3d(1)).

(5) Identify the procedures used to measure the value of each of the critical parameters. (II-3f(2)).

(6) Identify the sampling points. (II-3f(3)).

(7) State the procedure for recording and management of data, defining corrective actions for various out of specification parameters. (II-3f(4)).

440-0 REACTOR SYSTEMS

440.5 (5.2.2) Section 5.2.2.1 of the FSAR states that the RCS average temperature and pressure are assumed to be at their maximum values. Provide the assumed values and verify that the maximum instrumentation and control errors have been assumed. Also discuss the preoperational tests which will verify the accuracy of instrumentation systems used to initiate overpressure protection.

440.6 (5.2.2) In Section 5.2.2 references are made to WCAP 7769. Provide a comparison of Shearon Harris parameters for all parameters listed in Table 2.2 of the topical report. Where differences exist, show that these differences will not affect the conservatism of the results given in WCAP 7769.

440.7 5.2.2) WCAP 7769, Section 3.4 assumes failure of one steam generator safety relief per loop. Provide assurance that your remaining safety valves can provide the required minimum capacity or justify why your analysis assumes only a single failure in one loop.

440.8 (5.2.2) Section 5.4.13.2.1 references a backpressure compensation feature on the pressurizer safety reliefs. Provide a discussion of this feature which explains how this function is performed.

440.9 (5.2.2) The Branch Technical Position RSB 5-2 in NUREG-0800 relates to overpressure protection of PWRs while operating at low temperatures. Demonstrate compliance to this position.

440.10 (Fig. 3.4-2) (16.2) Figure 3.4-2, which is the Appendix G curve, has a relatively large line which makes it difficult to determine the minimum pressure indicated. Provide a new curve and identify the minimum pressure value.

440.11 (5.2.2) Section 5.2.2.1 of FSAR states that the nominal flow from any steam generator is 0.97×10^6 lb/hr. Justify this value.

440.12 (5.2.2) In Section 16.2, Tech. Spec. 3.4.9.3, PORV relief settings and RCS cold leg temperatures are needed. Supply these values.

440.13 (5.2.2) Section 5.2.2.7 of FSAR references Sections 10.3.2.2 and 10.3.2.3 for material specifications of the steam system safety and power operated relief valves. This information is not provided in the referenced sections. Provide this information.

440.14 (Fig. 3.4-2) (16.2) Provide justification for allowing maximum heatup rates of 100°F/hr as authorized by Tech. Spec. 3/4.4.9 when the Appendix G curve is curve is based on a maximum of 60°F/hr.

440.15 (5.2.2) Provide verification that during your analysis of operational transients and faulted conditions concerning overpressure protection that you assume the reactor trip is initiated by the second safety grade signal, Ref. SRP 5.2.2 II.A.c.iii.

440.16 (5.4.7) Following an electrical system single failure, you state that limited action outside the control room is necessary to open the suction isolation valves for initiation of RHR cooling. What actions are necessary and where must they be performed? Provide the procedures which the operators will need to use following a postulated failure such as discussed above.

Provide details of the alarms and indications which would inform the operators that an RHR suction line isolation valve has closed while the plant is in shutdown cooling. Is there any common failure which would result in both valves being closed while in shutdown cooling?

440.17 (5.4.7) Discuss the potential for exceeding the allowable cooldown rate of the RHR and the reactor coolant system during the shutdown cooling mode of operation assuming loss of the nonsafety-grade instrument air system which controls the RHR heat exchanger outlet and bypass valves. Failure of the RHR heat exchanger bypass valves in the closed position would cause loss of bypass flow with the possibility of exceeding the

440.17 (5.4.7) (cont'd) allowable cooldown rate. The resulting stresses on the piping systems must be assessed.

440.18 (5.4.7) Assuming the most severe overpressure transient at low temperatures, will system relief capacity be adequate to prevent RHR system pressure from exceeding 110% of design? Justify your choice of the most severe overpressure incident at low temperature considering events which have occurred in operating reactors.

440.19 (5.4.7) Provide the following information related to pipe break or leaks in high or moderate energy lines outside containment associated with the RHR system when the plant is in a shutdown cooling mode.

(1) Determine the maximum discharge rate from a pipe break in the systems outside containment used to maintain core cooling.

(2) Determine the time available for recovery based on these discharge rates and their effect on core cooling.

(3) Describe the alarms available to alert the operator to the event, the recovery procedures to be utilized by the operator, and the time available for operator action.

440.20 (5.4.7) Provide or reference a discussion of your compliance with each item of RSB BTP 5-1 in NUREG-0800. Justify any deviations from this Branch Technical Position.

It is the staff's position that all operator actions necessary to take the plant from normal operation to cold shutdown should be performed from the control room. Indicate whether there are any systems or components needed for shutdown cooling which are de-energized or have power locked out during plant operation. If so, indicate what actions have to be taken to restore operability to the components or systems.

440.21 (5.4.7) Provide additional information regarding the power sources supplied to the RHR isolation valves. The staff's position is that a single failure of a power supply or interlock will not prevent isolation of the RHR when RCS pressure exceeds its design pressure. Additionally, loss of a single power supply cannot result in the inability to initiate at least one 100 percent RHR train.

440.22 (5.4.7) Section 5.4.7.2.4 of FSAR states that "Each discharge line from the RHRS to the RCS is equipped with a pressure relief valve designed to relieve the maximum possible back leakage through the valves." What is the basis for determining the maximum possible back leakage? Is this back leakage consistent with a relief flow capacity of 20 gpm at a set pressure of 600 psig? Show that there are design provisions to permit periodic testing for leak tightness of the check valves that isolate the discharge side of the RHRS from the RCS.

440.23 (5.4.7) Section 5.4.7.2.4 of FSAR states that "Each inlet line to the RHRS is equipped with a pressure relief valve sized to relieve the combined flow of all charging pumps at the relief valve set pressure. Each valve has a relief flow capacity of 900 gpm at a set pressure of 450 psig." This capacity appears to be less than the capacity indicated by the performance curve (Figure 6.3.2-9) for the charging pumps. What alerts the operator to the opening of RHRS relief valves? What procedures are available to the operator for responding to this event?

440.24 (5.4.7) Recent plant experience has identified a potential problem regarding the loss of shutdown cooling during certain reactor coolant system maintenance evolutions. On a number of occasions when the reactor coolant system has been partially drained, improper reactor coolant system level control, a partial loss of reactor coolant inventory, or operating the RHR system at an inadequate NPSH has resulted in air binding of the RHR pumps with a subsequent loss of shutdown cooling. Regarding this potential problem, provide the following additional information:

440.24 (5.4.7) (cont'd)

(1) Discuss the design or procedural provisions incorporated to maintain adequate reactor coolant system inventory, level control, and NPSH during partial drain evolutions.

(2) Discuss the provisions incorporated to ensure the rapid restoration of the RHR system to service in the event that the RHR pumps become air bound.

(3) Discuss the provisions incorporated to provide alternate methods of shutdown cooling in the event of loss of RHR cooling during shutdown maintenance evolutions. These provisions should consider maintenance evolutions during which more than one cooling system may be unavailable, such as loss of steam generators when the reactor coolant system has been partially drained for steam generator inspection or maintenance.

440.25 (5.4.7) RHR suctions lines can have water trapped between the two isolation valves. Address the possibility of pressure increasing in this pipe due to a temperature rise with respect to protection needed to assure valve and piping integrity. Identify any other sections ECC or RHR system pipe without overpressure protection and justify exclusion of such protection.

440.26 (5.4.7) What indicates loss of component cooling water to RHR pumps? Do all of these instruments meet IEEE 279 requirements? How long could the pumps continue to run following a loss of component cooling water without damage?

440.27 (5.4.7) The RHR miniflow bypass lines allow bypass flow when RHR pump discharge flow is insufficient. At what frequency is the operability of these miniflow lines verified? What assurances are available to the operating staff that the miniflow isolation valves are not misaligned due to operator error?

440.28 (5.4.7) Figures 5.4.7-1, 5.4.7-2, 6.3.2-3 and 6.3.2-4 are not consistent as to the location of the RHR miniflow bypass lines. Resolve this discrepancy and correct the figure(s). Revise these figures to show the RHRS with the same main components (valves, relief valves, and check valves, etc.).

440.29 (5.4.7) Table 5.4.7-1 of FSAR states that design temperature of the component cooling water system is 105°F. The text in 5.4.7.3 of FSAR for performance evaluation states a maximum CCW temperature of 120°F as does the text for Mode A of Notes to Figure 5.4.7-1. Provide the temperature used in calculating the cooldown times given in Section 5.4.7.1 and justify this value.

440.30 (5.4.7) The text of 5.4.7.1 of FSAR states that RHR operation is started with RCS pressure of 425 psig. Sheet 4 of Notes to Figure 5.4.7-2 Mode A, Initiation of RHR Operation, states that the pressure on PT-403 must be 400 psig before the suction valves can be opened. Provide clarification to this discrepancy.

440.31 (5.4.7) Section 5.4.7.2.4 of FSAR states that the RHRS suction side reliefs have a set pressure of 450 psig. It also states that the RHRS is not isolated from the RCS until a pressurizer bubble is formed at 600 psig and that the isolation valves receive an automatic close signal at 750 psig. Provide an explanation as to how the RHRS can be kept in service above 450 psig.

440.32 (5.4.7) Several interlocks are utilized in the ECC and RHR systems. Potential interlock failures should be addressed in the failure modes and effects analysis.

440.33 (6.3) Provide a discussion on the construction of the RWST enclosure. Is the enclosure heated? Vented? Where is the RWST vent line located? Include in your response the methods that will be utilized to prevent freezing of the RWST and the RWST vent.

440.34 (6.3) Section 6.3.3.1 of FSAR discusses the failure of a single steam dump In your design, could a single failure in the steam dump control circuitry cause more than one steam dump to fail open or inadvertently come open?

440.35 (6.3) Table 6.3.2-6 of FSAR shows the sequence of changeover from injection to recirculation. Provide a time reference for each action given in this table. Indicate the time required to complete each action, and what other duties the operator would be responsible for at this point in the postulated accident. How much time does the operator have to realign the SI and charging pumps before RWST water is exhausted? Show the required NPSH is available for these pumps in your answer.

440.36 (6.3) P&ID Figure 6.2.2-1 of FSAR indicates one sump isolation valve. P&ID Figure 6.3.2-3 indicates 2 sump isolation valves. Provide a comprehensive P&ID which shows the relationship of the RWST, containment sumps, all ECC pumps, containment spray pumps, and all interconnections.

440.37 (6.3) Plant experience has identified a potential problem regarding the long-term reliability of some pumps used for long-term core cooling following a LOCA. For all pumps that are required to operate to provide long-term core cooling, provide justification that the pumps are capable of operating for the required period of time. This justification could be based on previous testing or on previous operational experience of identical pumps. Differences between expected post-LOCA conditions and the conditions during previous testing or operational experience cited should be justified (i.e., water temperature, debris, water chemistry).

440.38 (6.3) Provide a list of all active components which are required for operation and support of the ECCS. Provide safety and seismic classification for each component and indicate what services such as cooling, lube oil and air are necessary for the proper functioning of each component. Also show the associated train for this service.

440.39 (6.3) Provide a discussion of procedures and administrative controls for manually resetting SIS following a LOCA. Specifically address the minimum time after actuation that the SI signal can be reset, and procedures to be followed if a reset were to be followed by a loss of offsite power.

440.40 (6.3) Provide a discussion of NPSH requirements for all ECCS pumps. Include in this discussion NPSH as required by pump warranty, estimated variability between pumps, and testing inaccuracies. Also provide the assumptions and calculations used to establish available NPSH.

440.41 (6.3) Certain automatic safety injection signals and certain safety systems components, such as accumulators, charging pumps and/or SI pumps, are blocked to preclude unwanted actuation of these systems during normal shutdown and startup operations. Describe the alarms available to alert the operator to a failure in the primary or secondary system during this phase of operation and the time frame available to mitigate the consequences of such an accident. If applicable, provide or reference sensitivity studies to demonstrate that these cases are bound by existing analyses.

440.42 (6.3) The ECCS should retain its capability to cool the core in the event of a failure of any single active component. Section 6.3.1 of FSAR states "Spurious movement of a motor-operated valve due to the actuation of its positioning device coincident with a LOCA has been analyzed and found not to be credible for consideration." It is the staff's position that the spurious movement must be specifically addressed. Identify all single failures that could prevent the ECCS from performing its function for each mode of ECCS operation and discuss the direct effect of each failure.

440.43 (6.3) During our review of license applications we have identified concerns related to the containment sump design and its effect on long term cooling following a Loss of Coolant Accident (LOCA).

440.43 (6.3) (cont'd)

These concerns are related to (1) creation of debris which could potentially block the sump screens and flow passages in the ECCS and the core, (2) inadequate NPSH of the pumps taking suction from the containment sump, (3) air entrainment from streams of water or streams which can cause loss of adequate NPSH, (4) formation of vortices which can cause loss of adequate NPSH, air entrainment and suction of floating debris into the ECCS and (5) inadequate emergency procedures and operator training to enable a correct response to these problems. Preoperational recirculation tests performed by utilities have consistently identified the need for plant modifications.

We require the following actions to provide additional assurance that long-term cooling of the reactor core can be achieved and maintained following a postulated LOCA.

(1) Establish a procedure to perform an inspection of the containment, and the containment sump area in particular, to identify any materials which have the potential for becoming debris capable of blocking the containment sump when required for recirculation of coolant water. Typically, these materials consist of: plastic bags, step-off pads, health physics instrumentation, welding equipment, scaffolding, metal chips and screws, portable inspection lights, unsecured wood, construction materials and tools as well as other miscellaneous loose equipment.

"As Licensed" cleanliness should be assured prior to each startup.

This inspection shall be performed at the end of each shutdown as soon as practical before containment isolation.

(2) Institute an inspection program according to the requirements of Regulatory Guide 1.82, item 14. This item addresses inspection of the containment sump components including screens and intake structures.

440.43 (6.3) (cont'd)

(3) Discuss possible actions for the operator to address both a possible vortexing problem (with consequent pump cavitation) and sump blockage due to debris. These should address all likely scenarios and should list all instrumentation available to the operator (and its location) to aid in detecting problems which may arise, indications the operator should look for, and operator actions to mitigate these problems.

(4) Pipe breaks, drain flow and channeling of spray flow released below or impinging on the containment water surface in the area of the sump can cause a variety of problems; for example, air entrainment cavitation and vortex formation.

Describe any changes you plan to make to reduce vortical flow in the neighborhood of the sump. Ideally, flow should approach uniformly from all directions.

(5) Evaluate the extent to which the containment sumps in your plant meet the requirements for each of the items previously identified; namely debris, inadequate NPSH, air entrainment, vortex formation, and operator actions.

The following additional guidance is provided for performing this evaluation.

(6) Refer to the recommendations in Regulatory Guide 1.82 (Section C) which may be of assistance in performing this evaluation.

(7) Provide a drawing showing the location of the drain sump relative to the containment sumps.

(8) Provide the following information with your evaluation of debris:

(a) Provide the size of openings in the fine screens and compare this with the minimum dimensions in the pumps which take

440.43 (6.3) (cont'd) suction from the sump, the minimum dimension in any spray nozzles and in the fuel assemblies in the reactor core or any other line in the recirculation flow path whose size is comparable to or smaller than the sump screen mesh size in order to show that no flow blockage will occur at any point past the screen.

(b) Estimate the extent to which debris could block the trash rack or screens (50 percent limit). If a blockage problem is identified, describe the corrective actions you plan to take (replace insulation, enlarge cages, etc.).

(c) For each type of thermal insulation used in the containment, provide the following information:

(i) type of material including composition and density,

(ii) manufacturer and brand name,

(iii) method of attachment,

(iv) location and quantity in containment of each type,

(v) an estimate of the tendency of each type to form particles small enough to pass through the fine screen in the suction lines.

(d) Estimate what the effect of these insulation particles would be on the operability and performance of all pumps used for recirculation cooling. Address effects on pump seals and bearings.

440.44 (6.3) The staff has reviewed the SHNPP proposed containment recirculation sump test, and has determined that the "Acceptance Criteria" for the test was not acceptable. The staff will require verification that

440.44 (6.3) (cont'd) no vortexing tendencies exist in the recirculation sump. Provide response to the following comments concerning the staff's review of the proposed test.

(1) The applicant should take measures to reduce the Froude Number, thus eliminating any vortex formation in the test. Also, if vortex suppression equipment is intended to be used, the tests should conclusively demonstrate the ability of the equipment to prevent vortexing over the full range of anticipated sump conditions.

(2) Verify that the test flow is based on the worst flow rate by assuming all ECCS pumps are operated at run-out conditions.

(3) The applicant should address the "uncertainties" of the test result due to level decrease during test which may cause a mass transient effect which would or would not be actually seen during LOCA events. Based on the staff's estimation, the decreasing rate of water level is 8" - 13"/minute.

(4) The applicant should verify the available NPSH to satisfy the required NPSH for ECCS pumps. In the determination of this available NPSH, the applicant should consider the temperature effect to the ΔP across the pump suction lines since the proposed test will be conducted at 170°F instead of the actual post-LOCA sump water temperature (above 200°F).

(5) Address the possibility of subsurface vortexing.

440.45 (6.3) Recently, a similar plant has indicated that a design error existed in the sizing of their RWST. This error was discovered during a design review of the net positive suction head requirements for the containment spray and residual heat removal pumps. The review showed that there did not appear to be sufficient water in the RWST to complete the transfer of pump suctions from the tank to the containment sump, before the tank was drained and ECCS pump damage occurred.

440.45 (6.3) (cont'd) It was reported that in addition to the water volume required for injection following a LOCA, an additional volume of water is required in the RWST to account for:

(1) Instrument error in the RWST level measurements

(2) Working allowance to assure that normal tank level is sufficiently above the minimum allowable level to assure satisfaction of technical specifications

(3) Transfer allowance so that sufficient water volume is available to supply pumps during the time needed to complete the transfer process from injection to recirculation.

(4) Single failure of the ECCS system which would result in a larger volume of water being needed for the transfer process. In this situation, the worst single failure appears to be failure of a single ECCS train to realign to the containment sump upon low RWST signal. This results in the continuation of large RWST outflow and reduces the time available for manual recirculation switchover, before the tank is drawn dry and the operating ECCS pumps are damaged.

(5) Unusable volume in the tank is present because once the tank suction pipes are reached, the pumps lose suction and any remaining water is unusable. Additionally, some amount of water above the suction pipes may also be unusable due to NPSH considerations and vortexing tendencies with the tank.

Preliminary indications are that approximately an additional 100,000 gallons of RWST capacity were needed to account for these considerations. It is our understanding that the design parameters for instrument error, transfer allowance and single failure have changed since the original sizing of the tank.

440.45 (6.3) (cont'd) In light of the above information, discuss the adequacy of your Refueling Water Storage Tank. Provide a discussion of the necessary water volumes to accommodate each of the five considerations indicated above. Justify your choice of volumes necessary to account for each consideration. Provide drawings of your RWST, showing placement and elevation of tank suction lines, and level sensors. Also, provide operator switchover procedures for aligning to the recirculation mode, with estimates of the time required for each action.

440.46 (6.3) Describe the means provided for ECCS pump protection including instrumentation and alarms available to indicate degradation of ECCS pump performance. Our position is that suitable means should be provided to alert the operator to possible degradation of ECCS pump performance. All instrumentation associated with monitoring the ECCS pump performance should be operable without offsite power, and should be able to detect conditions of low discharge flow.

440.47 (6.3) Table 6.3.2-2 lists the capacity of the accumulator relief valves as 1500 SCFM. Verify that this capacity is adequate to relieve all possible RCS backleakage and that it is adequate to prevent accumulator overpressurization during level adjustments, assuming equipment malfunction or operator error while adding water to the accumulators. Show the relief valve fluid flow rate and temperature assumed in this calculation.

440.48 (6.3) Provide a discussion of methods employed to prevent hot leg injection during the ECCS cold leg injection phase.

440.49 (6.3) Provide a discussion of methods used to insure that the ECC system is placed and maintained in a water filled condition to preclude the effects of water hammer.

440.50 (6.3) The piping system associated with the BIT is apparently susceptible to single failures (single surge tank heater). Address single failures in this system and provide justifications that these single failures will not impair the function of the BIT.

440.51 (6.3) Provide a discussion on excessive boron concentration in the reactor vessel and hot leg recirculation flushing related to long-term cooling following a LOCA. During hot leg injection, what will be the minimum expected flow rate in the hot leg, and what is the required flow rate to match boil-off?

440.52 (6.3) Discuss in detail how your preoperational test program for the ECCS will conform to the recommendations of Regulatory Guides 1.68 and 1.79. Specifically, include the procedures which will be used to verify nominal and runout ECCS flow, pump characteristics, piping losses and verification that each check valve in the system is capable of performing both its isolation and flow function.

440.53 (6.3) Identify all ECCS LOCA related instruments, valve and valve motors which are expected to be flooded following a postulated LOCA. For any ECCS or RHR valve motors which are submerged following a LOCA, evaluate the consequences of spurious activation or failure of the valves.

440.54 (6.3) Westinghouse has indicated a potential problem associated with the volume control tank level instrumentation and level control system. In some designs a potential single failure could cause loss of suction and subsequent damage to all safety injection pumps. Provide a discussion of this potential problem for the Shearon Harris design.

440.55 (6.3) The Standard Review Plan (NUREG-800) indicates that ECC testing should include delivery of coolant to the vessel during shutdowns for refueling. Provide or reference a discussion of proposed ECC testing during refueling.

440.56 (6.3) Provide or reference a discussion to confirm that there are provisions for maintenance of the long-term coolant recirculation and decay heat removal systems (pump or valve overhaul) in the post-LOCA environment, including consideration of radioactivity.

440.57 (6.3) Provide or reference a discussion to confirm that ECCS components located exterior to the reactor containment are housed in a structure which, in the event of leakage from the ECCS, permit venting of releases through iodine filters designed in accordance with Regulatory Guide 1.52.

440.58 (6.3) Describe the instrumentation available for monitoring ECCS performance during post-LOCA operation (injection mode and recirculation mode). Include a description of the instrument location and power supply as well as environment qualification and safety characterization.

440.59 (15.0) Section 15.0.2 of FSAR states that "A control system setpoint study is performed in order to simulate performance of the reactor control and protection systems: In this study, emphasis is placed on the development of a control system which will automatically maintain prescribed conditions in the plant even under a conservative set of reactivity parameters with respect to both system stability and transient performance." Show that the results of this study and the system setpoints are consistent with the accident analysis assumptions and that these assumptions are conservative taking into consideration instrumentation errors.

440.60 (15.0) Section 15.0.8 of FSAR states "The pressurizer heaters are not assumed to be energized during any of the Chapter 15 events." For each of these events show that this is a conservative assumption or quantify the effects of the heaters being energized.

440.61 (15.0) For each Chapter 15 event analyzed, list the components (such as relief, safety, isolation valves) that are assumed to function. Provide the sequence of events assumed and indicate when signals are generated, equipment is actuated, valves are closed, etc. For each component, supply the response time, discharge rates, pump flows, etc., that were used in the analysis and show how these values are verified to be conservative. Also, Tables 15.0.8-1 and 15.1.5-1 are inconsistent, e.g., Table 15.1.5-1 lists emergency service water which

440.61 (15.0) (cont'd) is not listed in Table 15.0.8-1. These Tables should be corrected or deleted. Your response to the above question should not reference these Tables unless they are corrected.

440.62 (15.0) Identify the version of codes used in the Chapter 15 analysis of the core, RCS, and secondary coolant system and reference NRC approval letters for use of these versions of the codes.

440.63 (15.0) A change in the Westinghouse fuel rod internal pressure design criteria will permit the internal fuel rod pressure to exceed system pressure. For some events, this will result in an increase in the number of rods normally expected to fail. If the fuel design is based on this higher fuel rod internal pressure design criteria, show that the effects of the higher fuel rod internal pressure have been properly factored into predictions of the effects of fuel rod ballooning and number of rod failures.

440.64 (15.0) Discuss the loss of instrument air showing that it meets the appropriate acceptance criteria for a moderate frequency event. Causes of a loss of instrument air and consequences should be addressed. The loss of instrument air should be considered during all phases of reactor operation. Also, present your plans and capability for preoperational or startup tests to substantiate the analyses.

440.65 (15.0) Provide a discussion of long-term effects and events for each accident analyzed in Chapter 15.0, assuming no operator action prior to times justified by ANSI N660. When operator action is needed, provide a complete assessment of the operator's role and show that sufficient time is allowed for operator action to be accomplished.

440.66 (15.0) For each event that is analyzed in Chapter 15, provide an additional analysis that shows the incident, in combination with the MOST LIMITING single failure of any safety grade component or operator error, will not result in loss of any barrier other than a limited number of fuel rod cladding perforations. The most limiting single failure as defined

440.66 (15.0) (cont'd) in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and should satisfy the positions of Regulatory Guide 1.53. Provide justifications for your single failure or operator error selections as being the most limiting. Specify to what extent fuel failure occurs, if any, and the basis for it. Recognize that single failures or operator errors that have "No Effect" are not acceptable as "Most Limiting" responses, unless they are thoroughly justified.

440.67 (6.3 & (15.0) Provide a time sequence for the operation of the ECCS components that were used in the Chapter 15 analysis with and without offsite site power. What are the bases for these times and will they all be verified during preoperational testing?

440.68 (15.1) Table 15.1.4-1 & Figure 15.1.4-3 of FSAR indicate that the pressurizer is emptied during the inadvertent opening of steam relief or safety valve and steam line break transients. Discuss the potential effects of this condition, including the potential for and recovery from void formation in the RCS.

440.69 (15.1.2) In Section 15.1.2.1 of FSAR regarding increased feedwater flow incidents, you state "The overpower overtemperature protection (overtemperature and overpower ΔT trips) prevents any power increase which could lead to a DNBR less than 1.30." Later in 15.1.2.2-b it reads "Following turbine trip, the reactor will be automatically tripped." In the time sequence of events for this transient, Table 15.1.2-1, you do not identify the signal utilized to initiate a reactor trip. Provide clarification of the reactor trip sequence used in this analysis and the justification for the assumed sequence.

440.70 (15.1.2) In Section 15.1.2.2 of FSAR regarding an increase in feedwater flow transient, you state that the case assuming maximum reactivity feedback coefficients with rod control results in the greatest power increase. Other similar plants reviewed have stated that the greatest power increase occurs without rod control. Justify that your assumption is correct and does in fact result in the greatest power increase.

440.71 (15.1.3) In Section 15.1.3.1 of FSAR regarding an excessive increase in secondary steam flow, you state "This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control." What are the maximum possible load increases from these initiating events and how does this relate to the assumed 10 percent step load increase from rated load?

440.72 (15.1.5) Provide more detailed information concerning the auxiliary feed system and operator action assumed for the main steam line rupture analysis. Specifically address:

(1) Assumed auxiliary feed flow

(2) Time to deliver auxiliary feed

(3) Auxiliary feed temperature

(4) Operator actions assumed

(5) Time frame for operator action

(6) Alarms and indications provided to assist the operator in determining the correct course of action.

440.73 (15.1.5, 15.3.3) In regard to Section 15.1.5, main steamline break accident, of the Shearon Harris FSAR, we have noted during our review that the values listed in Table 15.1.5-4 for dose to the thyroid for an accident-generated iodine spike is 48 rem while the SRP Appendix A acceptance criteria (II.2) is 30 rem.

440.74 (15.2.3) In Section 15.0.6 of the Shearon Harris FSAR it states "There are various instrumentation delays associated with each trip function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to

440.74 (15.2.3) (cont'd) trip is the difference between the time that trip conditions are reached and the time the rods are free and begin to fall. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.0.6-1." With respect to accident analyses, the critical parameter is the time of insertion of the rod cluster up to the dashpot entry. In some of the analyses there is some confusion as to what time delays are being considered. A case in point is the analysis for turbine trip incidents. In Table 15.2.3-1 Accident 1, the overtemperature ΔT reactor trip setpoint is reached in 7.8 seconds and rods begin to drop in 9.8 seconds. From Table 15.0.6-1 we see that the time delay of the overtemperature ΔT is 6 seconds (which includes the total time delay from the time the temperature difference in the coolant loops exceeds the trip set). No explanation is given for the two seconds from reaching the overtemperature ΔT reactor trip setpoint and the time that the rods begin to drop. Provide clarification of these various time delays assumed for the analyses and show that the total delay to insertion up to the dashpot is conservative in all cases.

440.75 (15.2.6) Provide the basis for the steam generator heat transfer coefficient and flow during natural circulation flow in the RCS. Describe the available data, or data that you will obtain, which will verify the conservatism of the analysis of the loss of nonemergency AC power accident.

440.76 (15.2.6) In Table 15.2.6-1 of FSAR you indicate that the reactor coolant pumps begin to coastdown at 61.8 seconds following loss of nonemergency AC power. Provide justification for the time used for initiation of coastdown. Provide or reference sensitivity studies which indicate that assumptions used in this transient are conservative.

440.77 (15.2.6 & 15.2.7/ 15.2.8) Provide figures showing the DNBR versus time for the following transients; loss of nonemergency AC power, loss of normal feedwater flow, and feedwater system pipe break.

440.78 (15.2.8) Section 15.2.8.2 of FSAR states "Figures 15.2.8-1 and 15.2.8-8 show that following reactor trip, the plant remains critical." Both figures, however, show that total core reactivity ($\Delta K/K$) remains negative following the reactor trip. Explain this discrepancy.

440.79 (15.2.8) What operator actions, if any, are assumed in your analysis of the feedwater system pipe break? If operator actions are assumed, show that sufficient time as defined by ANSI N660 is available for completion of these actions.

440.80 (15.2.8) In Section 15.2.8.2 of FSAR you state "A 60 second delay was assumed following the low-low level signal to allow time for startup of the standby diesel generators and the auxiliary feedwater pumps. An additional 317 seconds was assumed before the feedwater lines were purged and the relatively cold (120°F maximum) auxiliary feedwater entered the unaffected steam generators." Table 15.2.8-1 indicates that "auxiliary feedwater is delivered to two intact steam generators" at 94.5 seconds. Provide clarification as to the assumed conditions for the transient analysis and the justification for these conditions.

440.81 (15.3.2) Section 15.3.2.1 of FSAR states "This event is classified as an ANS Condition III incident (an infrequent incident)." Section 15.3b states "Complete loss of forced reactor coolant flow (ANS Condition II event)." Section 15.3.2.3 states all acceptance criteria are met. Provide classification and the acceptance criteria utilized in evaluating this transient.

440.82 (15.3.3 & 15.3.4) Demonstrate that a rotor seizure and shaft break in a reactor coolant pump will not by itself generate a more serious condition or result in a loss of function of the reactor coolant system or containment barriers.

440.83 (15.3.3) The analyses of a locked reactor coolant pump rotor and a sheared reactor coolant pump shaft in Section 15.3 of the FSAR assumes the availability of offsite power throughout the event. In accordance

440.83 (15.3.3) (cont'd) with Standard Review Plan 15.3.3, 15.3.4 and GDC 17, we require that this event be analyzed assuming turbine trip and consequential loss of offsite power to the plant auxiliaries and resulting coastdown of all undamaged pumps. Appropriate delay times may be assumed for loss of offsite power if suitably justified.

The event should also be analyzed assuming the worst single failure of a safety system active component. Maximum technical specification primary system activity and steam generator tube leakage at the rate specified in the Technical Specifications should be assumed. The results of the analyses should demonstrate that offsite doses following the accident are less than the 10 CFR 100 guideline values.

440.84 (15.4.4) In the analysis of the startup of an inactive reactor coolant pump at an incorrect temperature the reactor trip is assumed to occur when the power range neutron flux exceeds the P-8 setpoint. Your analysis mentions a P-8 setpoint of 79 percent of rated power, which you state is the nominal setpoint plus 9 percent for nuclear instrumentation errors. Table 15.0.6-1 lists the P-8 setpoint as 85% and your proposed Technical Specifications list P-8 at $\leq$49% of rated power. What setpoint was used in your analysis and how did you arrive at that value? Show that the assumed setpoint is conservative for this transient analysis.

440.85 (15.4.4 & (15.0) Section 3.4.1.1 of FSAR of the proposed Shearon Harris Technical Specifications indicates that plant operation is allowed at 10% of of rated power and below with fewer than all three of the reactor coolant loops in operation. If this is in fact your intent, then the Chapter 15 analyses should show that the acceptance criteria for each event are met for all allowed loop operating modes.

440.86 (15.4.6) The Standard Review Plan, Section 15.4.6, has specific time criteria for acceptable operator action during a boron dilution event, namely:

(1) 30 minutes during refueling, and

440.86 (2) 15 minutes at all other times.
(15.4.6)
(cont'd) The reference point for "starting the clock" is when there is an identifiable alarm in the control room alerting the operator to the situation.

During cold shutdown your analysis indicates that 4.14 minutes elapse before the shutdown margin is lost.

For each of the cases evaluated in the FSAR, identify the redundant alarms that alert the operator, provide the time interval from this alarm to when the core would go critical, and identify Limiting Conditions of Operation for the Technical Specifications related to the sensors, alarms and equipment necessary to mitigate all of these events.

440.87 Section 15.4.6.2 of FSAR states "An uncontrolled boron dilution
(15.4.6) accident cannot occur during refueling as a result of a reactor coolant makeup system malfunction. This accident is prevented by administrative controls which isolate the reactor coolant system from the potential source of unborated water. Valves FCV-110B, FCV-111B, 8441, 8453, and 8439 in the CVCS will be locked closed during refueling operations. These valves will block the flow paths which could allow unborated makeup water to reach the RCS."

Describe the function of each valve, the type of valve, method of lock out, the administrative controls, and the effects and consequences of single failures and operator errors. Justify the assumption of having no source of dilution flow during refueling.

440.88 A PWR recently experienced a boron dilution incident due to inadvertent
(15.4.6) injection of NaOH into the reactor coolant system while the reactor was in a cold shutdown condition. Discuss the potential for a boron dilution event caused by the chemical addition portion of the CVCS and by dilution sources other than the CVCS (for example, via the engineered safety systems).

440.89 (15.4.6) For each dilution event analyzed, confirm that a conservatively high reactivity addition rate is assumed taking into account the effect of increasing boron worth with dilution and that all fuel assemblies are installed in the core.

440.90 (15.4.6) As required by Standard Review Plan, Section II.C.5.(vi), the analysis of boron dilution during refueling should assume all control rods are withdrawn from the core. Describe your assumptions regarding control rod location and justify any credit taken for the rods not being withdrawn.

440.91 (15.4.6) Provide details of the basis for the minimum water volume assumed for boron dilution during the various phases of plant operation. Specifically, discuss the RCS volume when the coolant is drained to the elevation of the hot leg piping and any credit taken for RHRS volume per RHRS loop.

440.92 (15.5) In Section 15.5.1.1 of FSAR you state that the operator would stop the safety injection after ensuring satisfactory plant conditions per operating procedures. Provide the operating procedure basis that the operator uses to determine if the safety injection signal is spurious.

440.93 (15.5) In Section 15.5.1.2 of FSAR the assumption of zero injection line purge volume (IE initial injection is borated to 2000 ppm) is used. Provide a discussion as to why this is more limiting than the possible transient that could be caused by the injection of unborated cold water into the cold legs.

440.94 (15.6.1) In Section 15.6.1.2 of FSAR reference is made to the digital computer code LOFTRAN. Verify that the values used in the computer code are valid for Shearon Harris.

440.95 (15.6.3) Figure 15.6.3-2 of FSAR indicates that the break flow is zero when reactor coolant system pressure is 1250 psia. It would appear that the break flow should be zero only for RCS pressures below the the

440.95 (15.6.3) (cont'd) lowest S/G safety valve setting. This also would then increase the amount of water/steam being dumped to the atmosphere. Provide the the assumptions used in deriving the break flow curve and justify their use.

440.96 (15.6.3) In Section 15.6.3.3.1 of FSAR, item j makes several statements that require clarification. Describe specifically how the affected S/G is isolated and how the operator terminates the break flow. It would appear that if the centrifugal HPSI pumps are still operating the S/G safety and/or relief valves may continue to cycle thereby releasing steam and/or fission product activity to the atmosphere. Provide the basis for these assumptions and justify why they are conservative.

440.97 (15.6.5) Provide assumptions in determining the RCP operation in all LOCA analyses. Additionally, show that you meet the requirement of NUREG-0737, Section II.K.3.5 that RCS pumps should be tripped automatically in case of a small break LOCA.

440.98 (15.6.5) Identify single failures and operator errors that would divert ECCS flow. For both large and small breaks discuss the effect of these failures on flow to the core, the containment water level and conformance with the 10 CFR 50.46 acceptance criteria.

440.99 (15.6.5) In the LOCA analysis, an upper head temperature equal to the cold leg temperature is assumed. Justify this assumption.

440.100 (15.6.5) Provide an analysis of the transient resulting from a break in the ECCS injection line. Describe the flow splitting which will occur in the event of the most limiting single failure and verify that the amount of flow actually reaching the core is consistent with the assumptions used in the analysis. Show that 10 CFR 50.46 acceptance criteria are satisfied.

440.101 (15.6.5) Does the model used to perform the LOCA analyses include an acceptable model for fuel cladding swelling and rupture? If so, identify

440.101 (15.6.5) (cont'd) justifying documentation. If not, correct the Shearon Harris LOCA analyses to include adequate treatment of fuel cladding swelling and rupture.

440.102 (15.6.1) Describe the recovery from the inadvertent opening of a pressurizer safety or relief valve accident. Include information on operator action, the pressurizer water level, the potential for void formation in the RCS, DNBR, and actuation of the engineered safety features.

440.103 (15.6.5) Westinghouse has recently indicated that for some plant designs full flow ECC assumed in the LOCA analysis results in a higher peak clad temperature. Provide an assessment for the Shearon Harris LOCA analysis, assuming NO failures in the ECC system.

440.104 (6.3 & 15.6.5) Sections 6.3.3.2 and 6.3.3.3 of the FSAR defines a small-break LOCA as a break up to 0.5 ft^2 in area and a large-break LOCA as larger than 0.5 ft^2. Section 15.6.5.1 defines a large break as greater than 1.0 ft^2 and a small break as less than 1 ft^2. Resolve the discrepancy.

440.105 (7.6.1.11 & 5.2.2.5) FSAR Section 7.6.1.11 references Section 5.2.2.5 for a discussion of pressure control during low temperature operation. This information is not provided in the referenced section. Provide this information.

440.106 (5.2.2) The staff is concerned that your proposed low temperature overpressure protection (LTOP) system does not adequately protect the reactor vessel during transient events where the vessel wall temperature lags behind the temperature used in the variable setpoint calculator. For example, starting an RCP in a loop with a hot steam generator when the RCS is water solid causes the RCS pressure and temperature to rise. Your LTOP system would automatically raise the PORV setpoint as a function of auctioneered cold or hot leg temperature, but the vessel wall will not be heated in this transient at the same rate. Thus, due to the LTOP system auctioneering scheme, the part of the RCS most vulnerable to brittle fracture will be protected to a higher pressure than its temperature allows.

440.106 (5.2.2) (cont'd) If, during a cooldown, the cold leg temperature detector downstream of the generator(s) being used failed, and a mass input event occurred, your proposed LTOP system may not protect the coldest location in the vessel since the setpoint would not be based on the coldest fluid temperature.

Address the above concerns by addressing the following questions:

(1) Show that for all normal events and events in which the RCS fluid temperature is changing, your proposed system suitably protects the reactor vessel at its coldest location.

(2) Show data to justify the RCS temperature transients assumed in (1) above.

(3) Include in your analyses the most limiting single failure, and justify the choice.

(4) Include in your analyses the effects of system and component response times, including

(a) temperature detectors
(b) pressure detectors
(c) logic circuitry
(d) PORV and its associated air system

Show the response times that were assumed and the techniques, including surveillance requirements, for ensuring their conservatism.

440.107 (6.3) Figure 6.3.2.5 indicates that the common header of the miniflow lines for the 3 safety injection pumps is designed to non-safety grade standards. Discuss the consequences of a postulated failure for non-seismic design miniflow common header with respect to safety injection pump operation.

471.0 RADIOLOGICAL ASSESSMENT

471.1 (13.1.2) The plant organization chart shown in Figure 13.1.2-1 shows the Radiation Protection Supervisor (RPS) reporting directly to the Manager, Technical Support. In accordance with Regulatory Guide 8.8 Section C.1.b.(3), it is our position that the Radiation Protection Supervisor should have access to the Harris Plant General Manager in radiation protection matters. In matters relating to radiological health and safety, the RPS has direct responsibility to both employees and management that can best be fulfilled if he is independent of station divisions such as operations, maintenance or technical support whose prime responsibility is continuity or improvement of station operability. The FSAR and proposed Technical Specifications should be revised to show how your planned radiation protection program reflects this position.

471.2 (12.2.1) (12.3.1) In Section 12.2.1, neutron and gamma streaming from the annulus between the RPV and the biological shield should be analyzed with respect to dose rate levels in containment where occupancy may be required. Section 12.3.1.11(c) states that a 3" neutron shield will be used to reduce this neutron streaming. Please specify the neutron and gamma dose equivalent rates that will exist at specific locations within the various levels of containment prior to shield installation and after the shield is installed (i.e. what is the

471.2 (12.2.1) (12.3.1) (cont'd)

effective factor of reduction for gamma and neutrons of the installed shield). A figure or table showing respective dose rates would be a suitable format. Please use relevent experience, as necessary, to demonstrate that your proposed shielding would accomplish the objective of Regulatory Guide 8.8 Section C.2.b, namely that it will provide sufficient shielding to achieve ALARA exposure to occupants in containment while the reactor is at power. Please specify the frequency at which entries are made into containment, the number of people making these entries, and their stay time.

471.3 (12.2.2) (12.3.3)

Table 12.2.2-1 shows that 0.12% failed fuel fraction was used to calculate airborne concentrations from equipment leakage and that the leak rate is based on operating plants experience. Section 12.3.3.1 states that the plant ventilation systems are designed to ensure that the maximum airborne radioactivity concentrations during normal operations, including anticipated operational occurrences, with design basis fuel defects of 1%, are maintained within the limits of Part 20. It is stated on page 12.2.2-3 that if 1% failed fuel was used in Table 12.2.2-1 calculations (to account for anticipated operational occurrances) "the occupancy in a few areas is expected to be very limited." This implies that based on the source terms in the aforementioned table, Part 20 limits would be exceeded, in spite of the ventilation design, during anticipated operational occurrances. Please explain the contradiction.

471.4 (12.2.2) Table 12.2.2-4 should reference the "Reactor Auxiliary Bldg" in the title and not the "Waste Processing Bldg." If not, please explain.

471.5 (12.3.2) Please specify those personnel who "reviewed, updated and modified" the shield design during all phases of the plants design and construction as specified in Section 12.3.2.2 (pg 12.3.2-5). Describe those areas of review and any design corrections that may have been made to ensure as low as is reasonably achievable exposures.

471.6 (12.3.2) Radiation levels in excess of 100 R/hr can occur in the vicinity of spent fuel transfer tubes; therefore, all accessible portions of the transfer tubes must be shielded during fuel transfer. In Section 12.3.2.8 you should address the manner in which access control and radiation monitoring will be incorporated into the radiation protection program to assure that during transfer of spent fuel from the reactor to the spent fuel pool through the fuel transfer tube, occupancy in the transfer tube area (e.g. during inspection of seismic bellows) will not occur. Use of removeable shielding for this purpose is acceptable. Provide appropriate figures (e.g. plan and elevation) that shows the shielding arrays for all direct gamma radiation and streaming pathways from the spent fuel during the transfer. On the same figure show the location of any

471.6 (12.3.2) (cont'd) administrative controls by barriers, signs, audible and visual alarms, locked doors, etc. All accessible portions of the transfer tubes that cannot be adequately shielded shall be clearly marked with a sign stating that potentially lethal fields are possible during fuel transfer. These markings should be clearly stated on portable shielding.

471.7 (12.3) Describe the radiation protection aspects of decommissioning which you have included in your plant design to ensure that exposures to workers, during decommissioning, will be ALARA in accordance with Regulatory Guide 8.8 Section C.1.d.(2).

471.8 (12.3) In accordance with the Standard Review Plan (NUREG-0800 Section 4 Facility Design Features) describe how airborne radioactivity and area radiation monitors will be used in the spent fuel pool area during normal and refueling operations. Specify in detailed drawings the location of all fixed monitors located in the spent fuel pool area that would be used to warn workers of inadvertent radioactivity releases to the spent fuel pool water and air in order to provide ALARA exposure to any occupants in the area.

471.9 (12.3) Section 12.3.4.1.8 states that area radiation monitors are shown in Figures 12.3.2-1 through 17. Location of these monitors in these figures has not been identified by the staff. Please provide revised figures that clearly show the location of these monitors.

471.10 (12.3.4) In accordance with the Standard Review Plan Section 12.4.(3) of Facility Design Features, specify which figures show that the ventilation duct airborne radioactivity monitors, described in Section 12.3.4.2.2, are upstream of HEPA filters so that they can survey the particulate airborne radioactivity levels where personnel normally have access. Describe how one would identify a specific enclosure, having the airborne radioactivity that may cause a ventilation monitor to alarm.

471.11 (12.5.1) Your Radiation Protection Supervisor back-up coverage, as described in Section 12.5.1.3, does not appear to meet our criteria. In the event of absence of the Radiation Protection Supervisor (e.g. vacation) the individual who will act as back-up should be qualified in accordance with the December 1979 revision of ANSI 3.1. That is the temporary replacement should have a B.S. degree in science or engineering, 2 years experience in radiation protection, 1 year of which should be nuclear power plant experience, 6 months of which should be on-site. Please describe your plan to meet these qualifications with your Radiation Protection Supervisor back-up.

471.12 (13.1.2) Page 13.1.2-10 indicates that the shift crew composition includes "an individual qualified in radiation protection procedures." NUREG-0654 "Criteria for Preparation and Evaluation of Radiological

471.12 (13.1.2) (cont'd) Emergency Response Plans and Preparation in Support of Nuclear Power Plants" requires that a radiation protection technician, whose qualifications are described in ANSI 18.1, shall be onsite at all times. The FSAR as written, would allow a designatd member of the shift crew (e.g. reactor operator) to act as health physics technician if he is qualified to implement radiation protection procedures. Only an assigned health physics (RC&T) technician should be assigned off-shift radiation protection responsibilities. Therefore, the FSAR should be revised accordingly.

471.13 (13.0) There is no figure in Chapter 13 with the Table of Organization for the Heath Physics staff including the RC&T placement in the chart and their respective numbers. Specify the number of H.P. technicians that will be used to accommodate the operation of the station. State your plan for staffing of H.P. technicians during major maintenance and refueling outages. Include the number of additional technicians that may be needed and their requirement to be qualified in accordance with ANSI 18.1.

471.14 (13.1.3) Section 13.1.2 of Regulatory Guide 1.70 "Standard Format and Content" specifies that the qualifications of appointees to plant positions should be presented in resumé format. Section 13.1.3 of the FSAR does not include the resumé of the on-site Radiation Protection supervisor to show that his qualifications are in accordance with Regulatory Guide 1.8. Please provide this resumé so that the staff can review the candidates qualifications against the Regulatory Guide.

471.15 (12.5.2) Section 12.5.2.1.1 states that routine counting of smears and air samples is performed in the health physics office "to minimize counting room background variations." Since the area directly below this room, at elevation 236', is characterized as Zone IV, please justify the rationale for the quote in the preceding statement and explain why these samples are not counted in the shielded counting room.

471.16 (12.5.2) i Instrument "primary" calibrations are usually performed annually. However, detector response characteristics are checked more frequently with a check-source to assure proper instrument operation and maintenance of calibration integrity. Explain your position of Section 12.5.2.1.7.3.1 that states that "CAM detector resonse to an appropriate check source will be performed on an annual basis," in lieu of more frequent detector response checks to assure constant instrument reliability. Please describe the calibration methods that will be used to calibrate the cams, the area monitor and fixed airborne radioactivity monitors. Include in the discussion the sources used, frequency of calibration and the training of personnel performing the calibration. Specify any national standards that are used in the procedure.

471.17 (12.5.2) Section 12.5.2.3 indicates facilities that allow for personnel entry into and exit from the Controlled Zone as being located on Elevation 261 feet of the Reactor Auxiliary Bldg. However,

471.17 (12.5.2) (Cont'd) these facilities are not shown in Figures 1.2.3.-27 thru 30 which are drawings of elevation 261 of the Auxiliary Bldg, but some are shown in Figure 1.2.2-45 at elevation 261 of the Waste-Processing Bldg. Please explain this discrepancy.

471.18 (12.5.2) In accordance with Regulatory Guide 1.97, (Table 2, Type C Variable), radiation exposure ratemeters should have a range extending to 10^4R/hr in order to have the capability to monitor for accidents. Section 12.5.2.11.2 shows that the maximum range of the entire instrument inventory extends to 10^3R/hr. You should, therefore, include an inventory of instruments with a range to 10^4R/hr or provide justification why accident levels may not exceed 10^3R/hr.

471.19 (12.5.3) Section 12.5.3.6.1.1., Personnel Dosimetry Evaluation, specifies that film badges may be used to detemine neutron dose equivalent when neutron dosimetry problems of these device are resolved. The applicant should note that film badges, used for neutron dosimetry at LWR's, are no longer acceptable to the staff. Therefore, this specification should be removed from the FSAR.

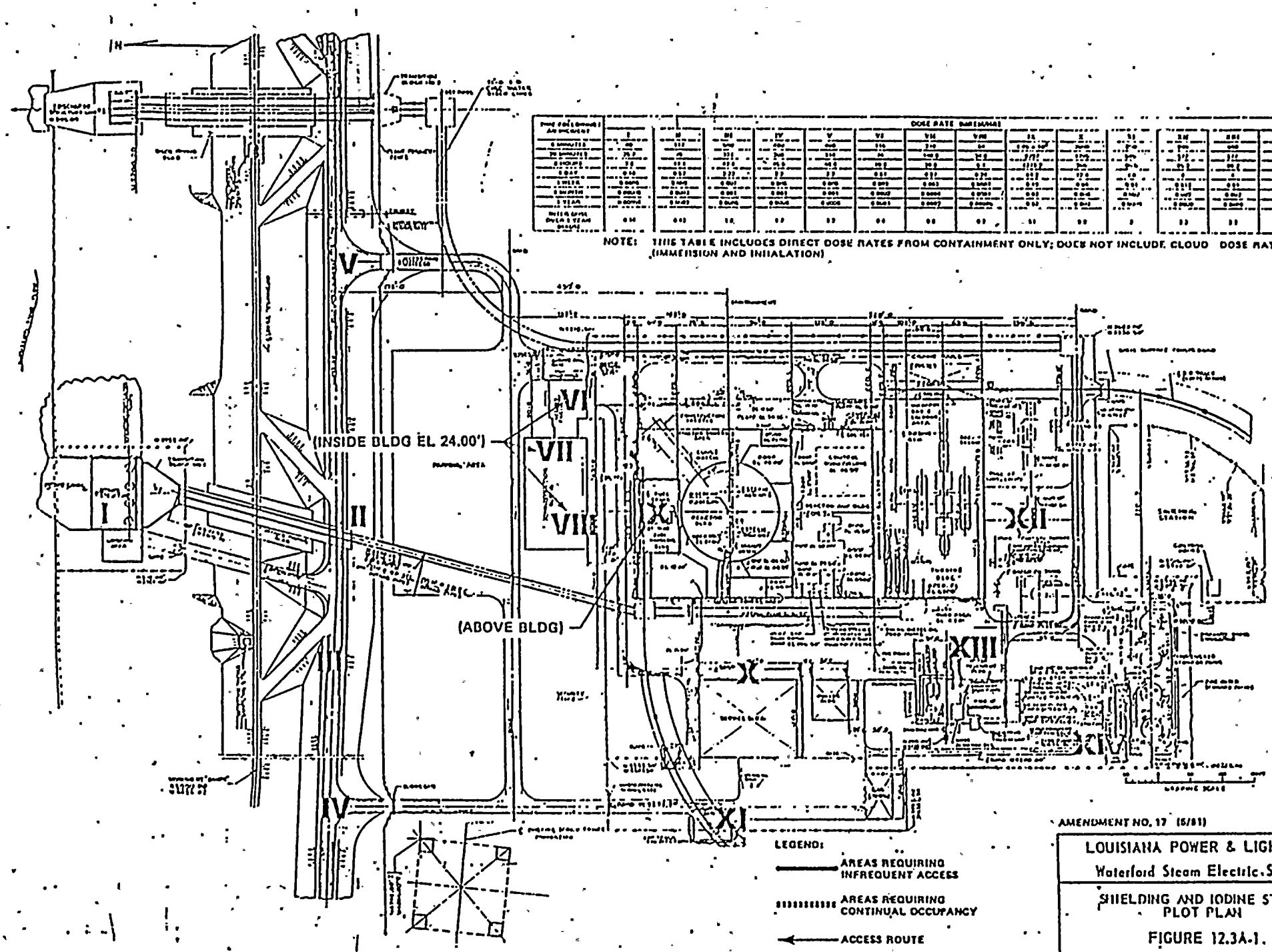
471.20 (12.5.3) You should commit to implement a bioassay whole body counting program in accordance with Regulatory Guide 8.26 or provide a description of an alternative to that Guide.

You should also commit to implement a training program in accordance with Regulatory Guide 8.13 (training on radiation risks to fetuses),

471.20 (12.5.3) (Cont'd) 8.27 (radiation protection training), and 8.29 (training on radiation risks).

471.21 (TMI) Although Item 2.1.6b of the TMI Appendix on Design Review of Plant Shielding for personnel access (II.B.2 of NUREG-0737) provides information to demonsrates compliance with Lessons Learned Short Term Requirements, the information contained is incomplete. You should provide dose rate zone maps that identifies, as a function of time following an accident, location of these vital areas, maximum dose rates in the area and occupancy requirements. An illustrative example of an acceptable zone map is attached to these Q-ls'.

471.22 (TMI) Provide a description and specify the location of the two high range monitoring systems that will be installed in containment per TMI Item 2.1.8.b.III. Varify that the monitors will meet the specifications of Table II.F.1-3 of NUREG-0737. A plant layout drawing should be used to show the location of the monitors.



NOTE: THIS TABLE INCLUDES DIRECT DOSE RATES FROM CONTAINMENT ONLY; DOES NOT INCLUDE CLOUD DOSE RATES (IMMERSION AND INHALATION)

AMENDMENT NO. 17 (5/81)

LOUISIANA POWER & LIGHT CO.

Waterford Steam Electric Station

SHIELDING AND IODINE STUDY PLOT PLAN

FIGURE 12.3A-1.

492.0 CORE PERFORMANCE - THERMAL/HYDRAULICS

492.1 (4.4.6) Operating experience on two pressurized water reactors, not of Westinghouse design, indicate that a significant reduction in the core flow rate can occur over a relatively short period of time as a result of crud deposition on the fuel rods. In establishing the Technical Specifications for the Harris, we will require provisions to assure that the minimum design flow rates are achieved. Therefore, provide a description of the flow measurement capability for the Harris units as well as a description of the procedure to measure flow.

492.2 (4.4.6) Regulatory Guides 1.133, Revision 1 and 1.70, Revision 3 require that FSAR Section 4.4.6 contain a description of the Loose Parts Monitoring System (LPMS) which will be installed at Harris. The information that should be supplied is:

(1) a description of the monitoring equipment including sensor locations;

(2) a description of how alert levels will be determined, including sources of internal and external noise, diagnostic procedures used to confirm the presence of a loose part, and precautions to ensure acquisition of quality data;

(3) a description of the operation program, including signature analysis during startup, normal containment environment operation, the seismic design, and system sensitivity;

(4) a detailed discussion of the operator training program for operation of the LPMS, planned operating procedures, and recordkeeping procedures;

(5) a report from the applicant which contains an evaluation of the system for conformance to Regulatory Guide 1.133; and,

(6) a commitment from the applicant to supply a report describing operation of the system hardware and implementation of the loose part detection program.

492.3 (4.4) State your intentions with regard to N-1 loop operation.

492.4 (4.4) The staff has developed interim criteria for evaluating the effects of rod bow on DNB for application to the Westinghouse standard 17x17 fuel assembly. The resultant reduction in DNBR due to rod bow is given by:

Burnup (MWD/MTU)	DNBR Reduction (%)
0	0
3500	0
5000	0
10000	2.15
15000	4.64
20000	6.74
25000	8.59
30000	10.27
35000	13.07
40000	19.09

Prior to issuance of the Technical Specifications, the applicant should present to the staff an acceptable method of accommodating the thermal margin reduction given above. Also, insert into the bases of the Technical Specifications any generic or plant specific margins that will be used to offset the DNBR reduction due to rod bowing.

492.5 (4.4.6) Provide the documentation required by NUREG-0737 Item II.F.2. The responses to the documentation should be given item-by-item.