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AUTH. NAME AUTHOR AFFILIATION
 MCDUFFIE, M. Carolina Power & Light Co.
 RECIP. NAME RECIPIENT AFFILIATION
 DENTON, H. R. Office of Nuclear Reactor Regulation, Director

SUBJECT: Forwards response to request for addl info re draft SER open items & safety review questions.

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Carolina Power & Light Company

SERIAL: LAP-83-428

SEP 19 1983

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
UNIT NOS. 1 AND 2
DOCKET NOS. 50-400 AND 50-401
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION

Dear Mr. Denton:

Carolina Power & Light Company hereby transmits one original and forty copies of additional information requested by the NRC as part of the safety review of the Shearon Harris Nuclear Power Plant. Some of the enclosed responses relate to Draft Safety Evaluation Report open items, while others relate to Safety Review Questions. The cover sheet of the attachment summarizes the related Open Item or the Safety Review Question addressed in the attachment along with the corresponding review branch and reviewer for each response.

We will be providing responses to other requests for additional information shortly.

Yours very truly,

M. A. McDuffie
Senior Vice President
Nuclear Generation

FXT/tda (7919FXT)
Enclosure

cc: Mr. B. C. Buckley (NRC)
Mr. G. F. Maxwell (NRC-SHNPP)
Mr. J. P. O'Reilly (NRC-RII)
Mr. Travis Payne (KUDZU)
Mr. Daniel F. Read (CHANGE/ELP)
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Mr. J. L. Kelley (ASLB)

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PDR ADOCK 05000400
E PDR

Boo 1
1/1

411 Fayetteville Street • P. O. Box 1551 • Raleigh, N. C. 27602

1944

MEMORANDUM

TO: SAC, NEW YORK
FROM: SAC, PHOENIX
SUBJECT: [Illegible]

Reference is made to your memorandum dated [Illegible] and to the report of [Illegible] dated [Illegible].

It is noted that [Illegible] advised that [Illegible] had been observed at [Illegible] on [Illegible]. It is suggested that you continue to maintain contact with [Illegible] and advise this office of any further information received.

Very truly yours,
[Illegible Signature]

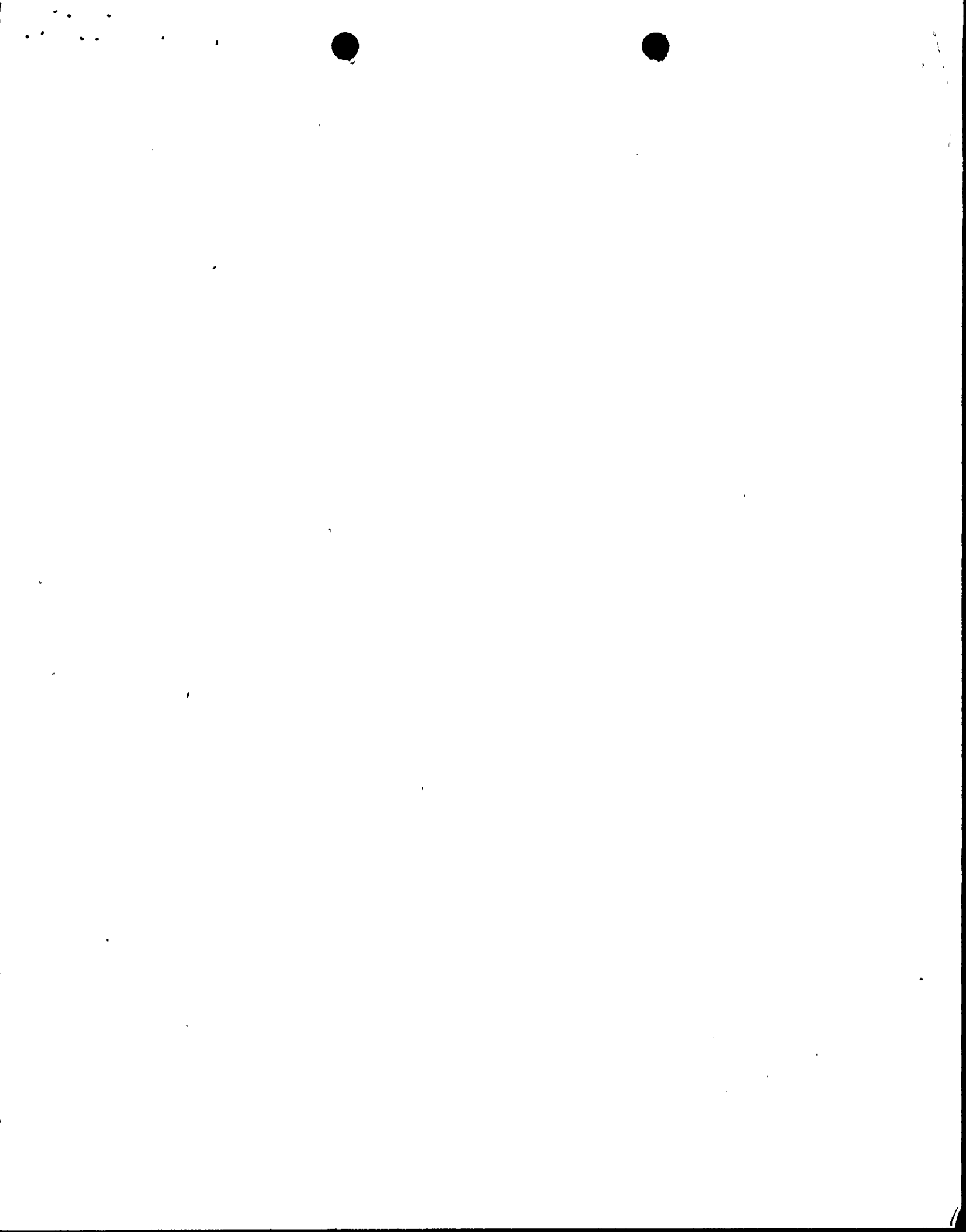
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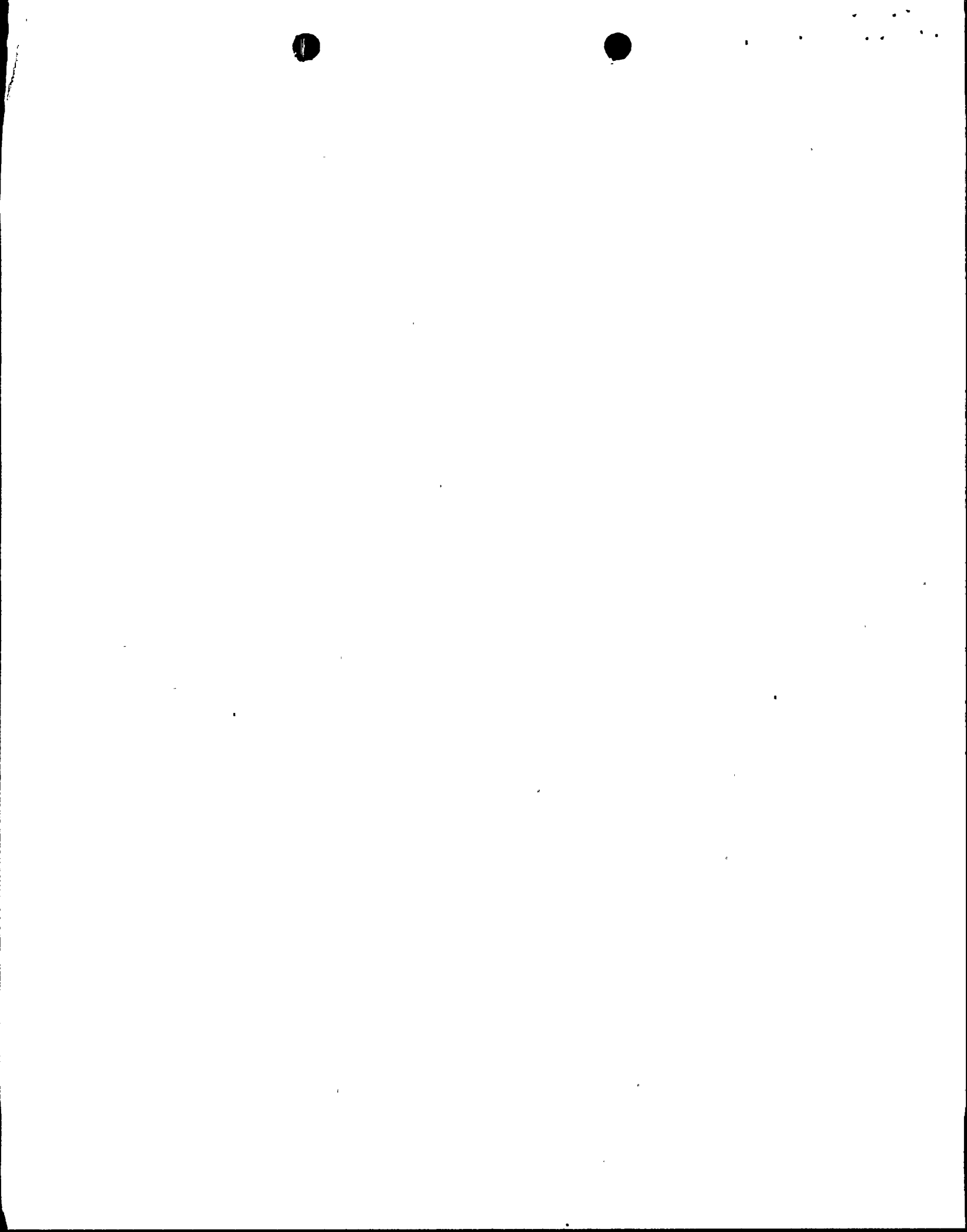
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LIST OF OPEN ITEMS/SAFETY REVIEW QUESTIONS, REVIEW BRANCH AND REVIEWER

Auxiliary Systems Branch/N. Wagner
Open Item 133

Instrumentation and Control System Branch/H. Li
Open Item 96

Mechanical Engineering Branch/D. Terao
Open Item 274

Power Systems Branch/E. Tomlinson
Open Item 114, Safety Review Questions 430.52,
430.66, 430.73

Structural Engineering Branch/S. Kim
Open Item 5

(7919FXIttda)

Shearon Harris Nuclear Power Plant

Draft SER Open Item 133

(Supplemental Information)

Provide the reactor vessel internals drop analysis in accordance with the guidelines of NUREG-0612, Control of Heavy Loads.

RESPONSE: See attached report, Heavy Load Analysis.

(7917NECccc)

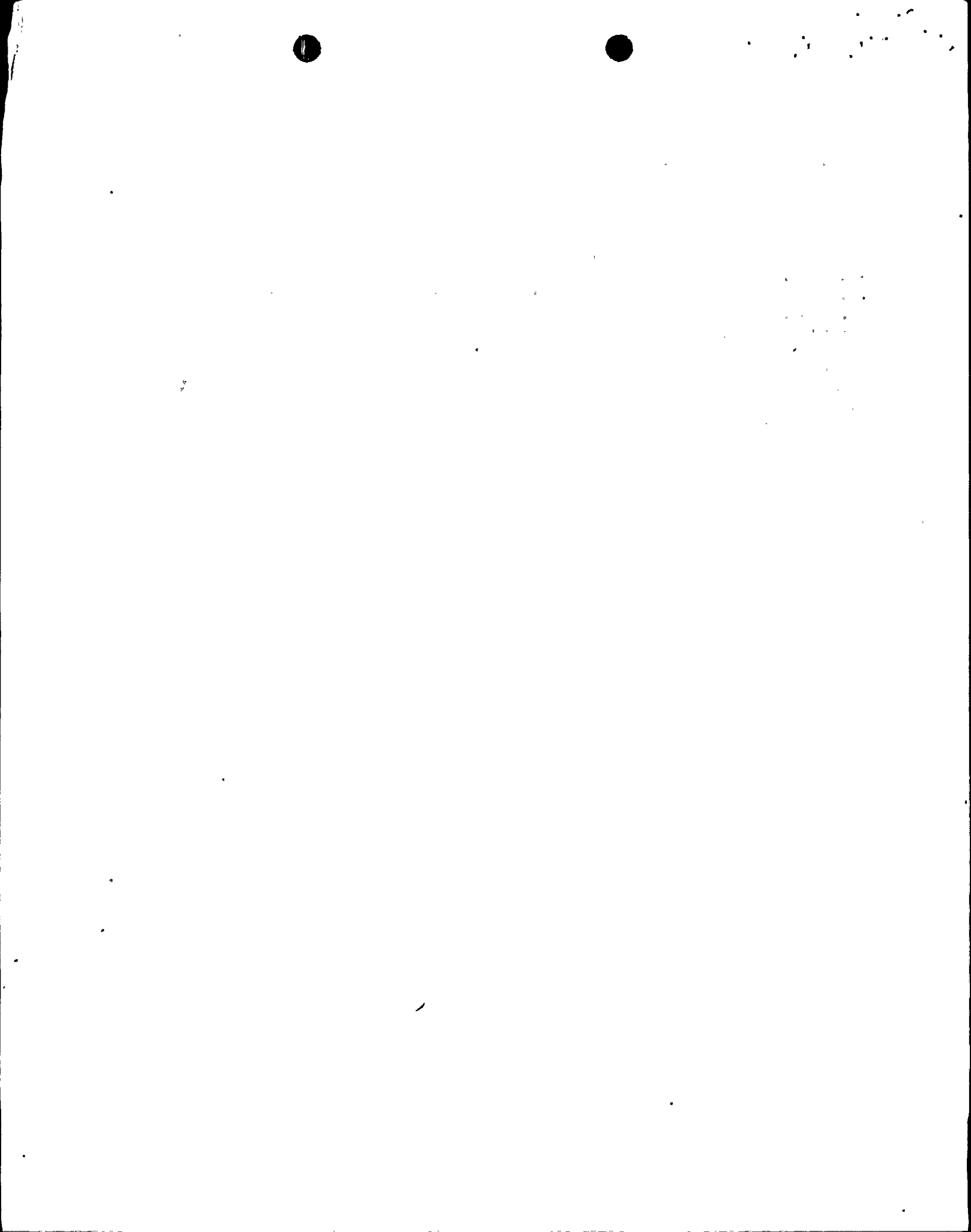
HEAVY LOAD ANALYSIS

FOR

SHEARON HARRIS NUCLEAR POWER PLANT

1.0 PURPOSE OF REPORT

The purpose of this report is to consider the consequences of various postulated accident cases which involve dropping the vessel upper internals for the Shearon Harris Nuclear Power Plant of Carolina Power and Light Company. The various accident cases described in this report are considered from the critical points along its travel path to or from the internals storage stand. The reactor vessel upper internals analysis is performed in accordance with the guidelines of NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants" for use in responding to Safety Review Question 410.11(3).



2.0 SUMMARY OF RESULTS

For the heavy loads analysis of the Shearon Harris Nuclear Power Plant, drop of the upper internals is postulated to occur during refueling. In all the possible drop scenarios considered, it is postulated that failure of the polar crane occurs, and the upper internals assembly falls onto the reactor vessel.

The results of the upper internals drop analysis indicated that the deformations and stresses at impact are within the acceptable limits; and the integrity of fuel cladding, reactor vessel nozzles, vessel supports and the core cooling capability is maintained. The maximum primary membrane plus primary bending stress intensity at the vessel nozzle and the maximum bearing stresses at the vessel nozzle support-pads are 47,810 and 23,420 psi, which are well below the code allowable limits. The code allowables for the primary membrane plus primary bending (i.e., $P_m + P_b$) and the bearing stresses during faulted conditions are limited to $3.6 S_m$ (99,360 psi) and $2\sigma_y$ (100,000 psi), respectively.

3.0 ASSUMPTIONS

The analysis performed herein is conservative since it maximizes the impact energies of the system for various postulated accident conditions. Key elements of the assumptions made are:

1. Skin frictional drag of an accelerating body through water is neglected.
2. For a free drop through air and water, the resistive impulse force due to impact with water is neglected. Further, the hydrodynamic drag force is calculated using a conservative estimate of drag coefficient (i.e., $C_d = 0.5$).
3. Only one-half of the bouyant force is taken into account when more than half of the assembly is submerged in water.
4. Impacting mass of the upper internals assembly is assumed to be rigid and no allowance is taken for its flexibility at impact.
5. Lift rig guide bushings do not bind with the vessel guide studs and consequently no credit is taken for frictional losses.
6. Buckling of vessel guide studs is assumed to have an insignificant effect on the upper internals assembly as it drops.
7. The drop of upper internals assembly onto the reactor vessel is assumed to be concentric. The concentric drop configuration is the limiting condition for loading the reactor vessel nozzles and supports, since in a non-concentric drop most of the impact would be taken by the refueling cavity floor and steel liner.

4.0 METHODS OF ANALYSIS

4.1 Impact and Sudden Loading

Assuming that the stresses due to impact are distributed throughout any elastic body exactly as in the static loading, it can be shown that the vertical deformation δ_i and stress σ_i , due to impact produced in any such body (bar, beam, truss, etc.) by vertical impact of a body falling from a height h ; are greater than the deformation δ_{st} and stress σ_{st} produced by the weight of the body applied as static load by the ratio [Reference 2]:

$$\frac{\delta_i}{\delta_{st}} = \frac{\sigma_i}{\sigma_{st}} = 1 + \sqrt{1 + 2 \frac{h}{\delta_{st}}} \quad (1)$$

where

δ_i and σ_i are deformation and stress due to impact and δ_{st} and σ_{st} are deformation and stress due to static loading.

Note that if $h = 0$, we have the case of sudden loading, and $\frac{\delta_i}{\delta_{st}} = \frac{\sigma_i}{\sigma_{st}} = 2$

4.2 Energy Losses

The above approximate relationship of Eqn. (1) is derived on the assumption that impact strains the elastic body in the same way (though not in the same degree) as static loading and that all the kinetic energy of the moving body is expended in producing this strain. Actually; in the impact, some kinetic energy is dissipated; and this loss, which can be found by equating the momentum of the entire system before and after impact; is most conveniently taken into account by multiplying the available energy by a factor k , the value of which is as:

A moving body of mass M strikes axially one end of a bar of mass M , the other end of which is fixed. Then the dissipation factor K is:

$$K = \frac{1 + 1/3 \frac{M_1}{M}}{\left(1 + 1/2 \frac{M_1}{M}\right)^2} \quad (2)$$

If there is a mass of M_2 attached to the struck end of the bar, then

$$K = \frac{1 + 1/3 \frac{M_1}{M} + \frac{M_2}{M}}{\left(1 + 1/2 \frac{M_1}{M} + \frac{M_2}{M}\right)^2} \quad (3)$$

using the dissipation factor K , an estimate of the impact energy can be made.

4.3 Equation of Motions in the Fluid Medium

The equation of motion that describes the travel of a body through fluid media is derived from the balance of forces, i.e.,

$$\Sigma F = \frac{W}{g_c} \frac{dV}{dt} \quad (4)$$

or

$$\left(\frac{MV}{g_c}\right) \frac{dV}{dy} = W - F_B - \left(\frac{C_D \rho_W A_P}{2 g_c}\right) v^2 \quad (5)$$

where

W = Weight of the body

F_B = Total bouyant force

C_D = Drag Coefficient

ρ_W = Density of Water

Solution of Eqn. (5) yields the velocity for impact.

5.0 POSTULATED DROP SCENARIOS AND THEIR CONSEQUENCES

In evaluating the consequences of the upper internals drop analysis for the Shearon Harris Nuclear Power Plant, a series of drop scenarios are postulated. For the purpose of clarity, Figure 1-1 through 1-3 show the refueling canal elevations, the upper internals assembly and the 17 x 17 fuel assembly for the Shearon Harris Power Plant. In the following we shall discuss briefly the drop scenarios and their consequences.

Scenario (1)

The upper internals lift rig is removed from the storage stand using the polar crane and lowered over the guide studs. While the lift rig is at its maximum height from the vessel flange surface, the polar crane is assumed to fail and the weight of the lift rig and the weight of polar crane lower block assembly drops onto the vessel.

If the bushings on the lift rig engage the vessel guide studs during the concentric drop, the drop weight of the polar crane lower block could impact the control rod drive shafts.

Consequence:

The assembly weight of 26,850 lb_f falls 10.2 feet through water to impact the control rod drive shafts. Some of the kinetic energy will be expended to buckle the control rods and the rest will be absorbed by the upper internals assembly at impact. The maximum load that can be experienced by the fuel assemblies would be the buckling load of the control rods which is significantly small (i.e., 5050 lb_f) and does not cause any damage to the fuel assemblies (Reference [3]).

Scenario (2)

If in Scenario (1), the guide bushings do not engage the vessel guide studs, the drop weight of the assembly could impact either the guide studs or the control rod drive shafts or both.

Consequence:

Same as in Scenario (1).

Scenario (3)

Upper internals assembly is lifted up from the reactor vessel for removal. When the guide bushings reach the top of the vessel guide studs, the polar crane fails and the weight of the assembly consisting of upper internals, lift rig and the crane lower block falls onto the reactor vessel flange.

Consequence:

Scenario (4) is more limiting.

Scenario (4)

This scenario is similar to the above Scenario (3), except that the upper internals assembly is lifted further up so that the upper support plate reaches the height of water level in the refueling cavity. It is assumed that the bushings engage the vessel guide studs and the assembly has a concentric drop of the vessel flange.

Consequence:

The assembly weight of 140,000 lb_f falls 23.75 feet through water and impacts on the top of the core hold-down spring and core barrel flange assembly which is supported at the vessel ledge. The idealized spring-mass system of the struck body (i.e., core hold-down spring, barrel flange, vessel and nozzle supports) is able to absorb all the kinetic energy of the dropped assembly without over stressing the system. The calculated impact load at each nozzle is 7.3×10^6 lb_f resulting in vessel nozzle pad bearing stress to be 23,400 psi which remains elastic and is well below the code allowable of $2\sigma_y = 100,000$ psi.

Scenario (5)

In Scenario (4), the control rod drive shafts will experience the same drop velocity as the upper internals assembly and could impact the fuel assembly nozzle spider hub.

Consequence:

The kinetic energy of control rod drive shaft at the time of impact is significantly less than the energy absorption capacity of the fuel assembly; and consequently no damage occurs to the fuel assembly.

Scenario (6)

In Scenario (4), it is assumed that the guide bushings on the lift rig do not engage the vessel guide studs. For the case when bushings do not engage the guide studs, the dropped weight of the assembly could impact the vessel guide studs and then impact the vessel flange.

Consequence:

In this scenario, some the kinetic energy of the system will be expended in buckling the guide studs and the dropped mass will impact the vessel flange with less impact velocity as compared to the above case and, therefore, Scenario (4) is more limiting.

Scenario (7)

In Scenario (4), it is assumed that after the bushings engage the vessel guide studs, the upper internals assembly rotates, losing the alignment with the head-vessel alignment pins, upper core plate alignment pins and the fuel assembly nozzle.

Consequence:

The assembly weight of 140,000 lb_f would impact the four head-vessel alignment pins and deform them plastically to absorb the total impact energy. The total deformations of the head-vessel alignment pins precludes the upper core plate impacting the fuel assembly spider hub. Consequently fuel assemblies do not see any load.

6.0 REFERENCES

- [1] Docket No. STN-50-516, 50-517 "Further Additional Supplemental Testimony on Contention I.D.2 (Spent Fuel Handling Accident)."
- [2] Roark, R. L., "Formulas for Stress and Strain," McGraw-Hill Book Company.
- [3] Simom, A. E., and Eng., G. H., "Fuel Rod and Fuel Assembly Mechanical Tests." WCAP-7687; March, 1971.

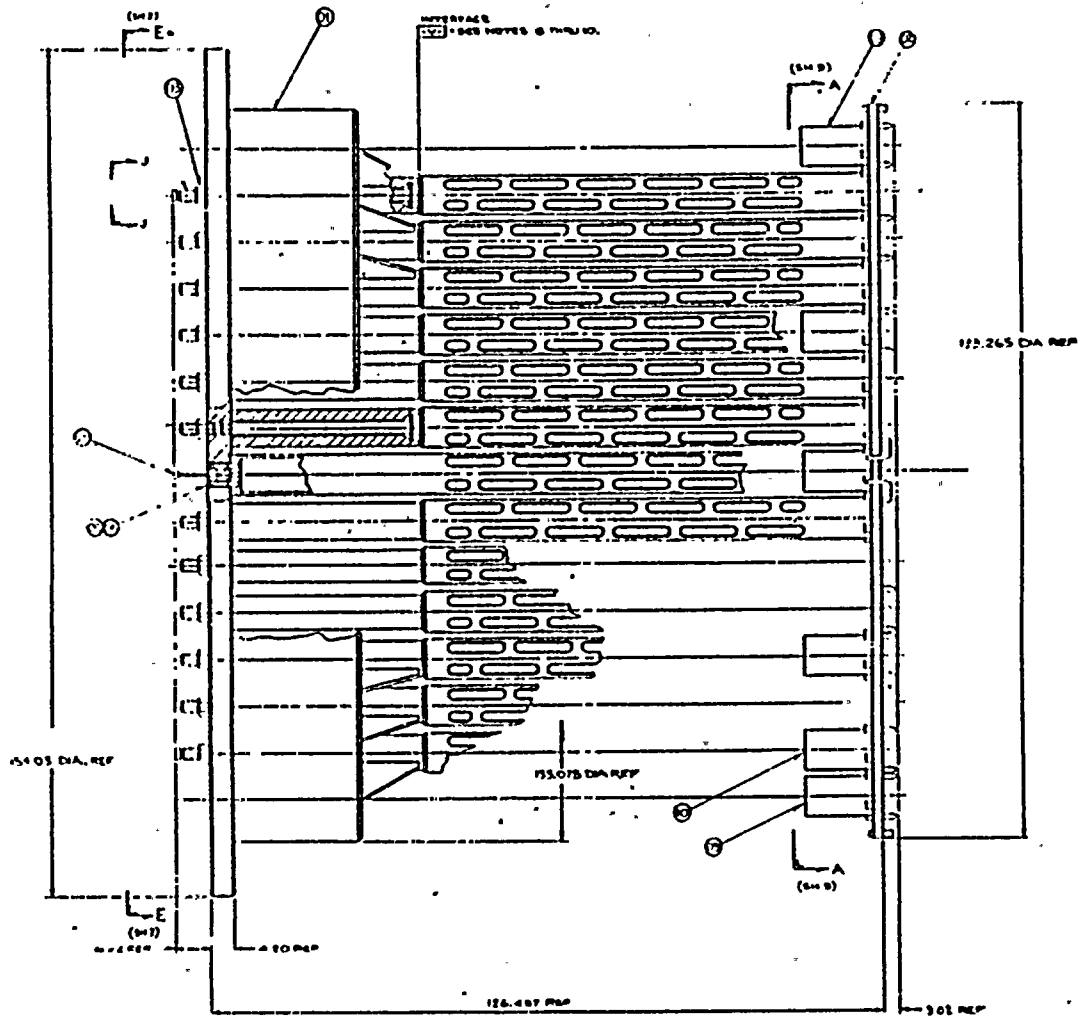


FIGURE 1-2 UPPER INTERNALS ASSEMBLY

Shearon Harris Nuclear Power Plant
Draft SER Open Item 96
Supplemental Information

Describe the modification to the Low Temperature Overpressurization System (LTOP) to correct a single failure potential.

RESPONSE: (This amends the previous CP&L response on this item dated 7-1-83)

To prevent a single failure of a temperature auctioneering device from preventing either Train A PORV or Train B PORV from performing its intended function, the control logic for the Low Temperature Overpressurization System (LTOP) will be modified.

The hot-leg temperature inputs that normally supply Train A auctioneering device, will also supply an additional auctioneering device that provides the Train B PORV permissive. The cold-leg temperature inputs that normally supply Train B auctioneering device, will also supply an additional auctioneering device that provides the Train A PORV permissive.

While the system remains automatic, with alarm annunciation, the potential single failure concern of the auctioneering device is eliminated.

(7909NECtda)

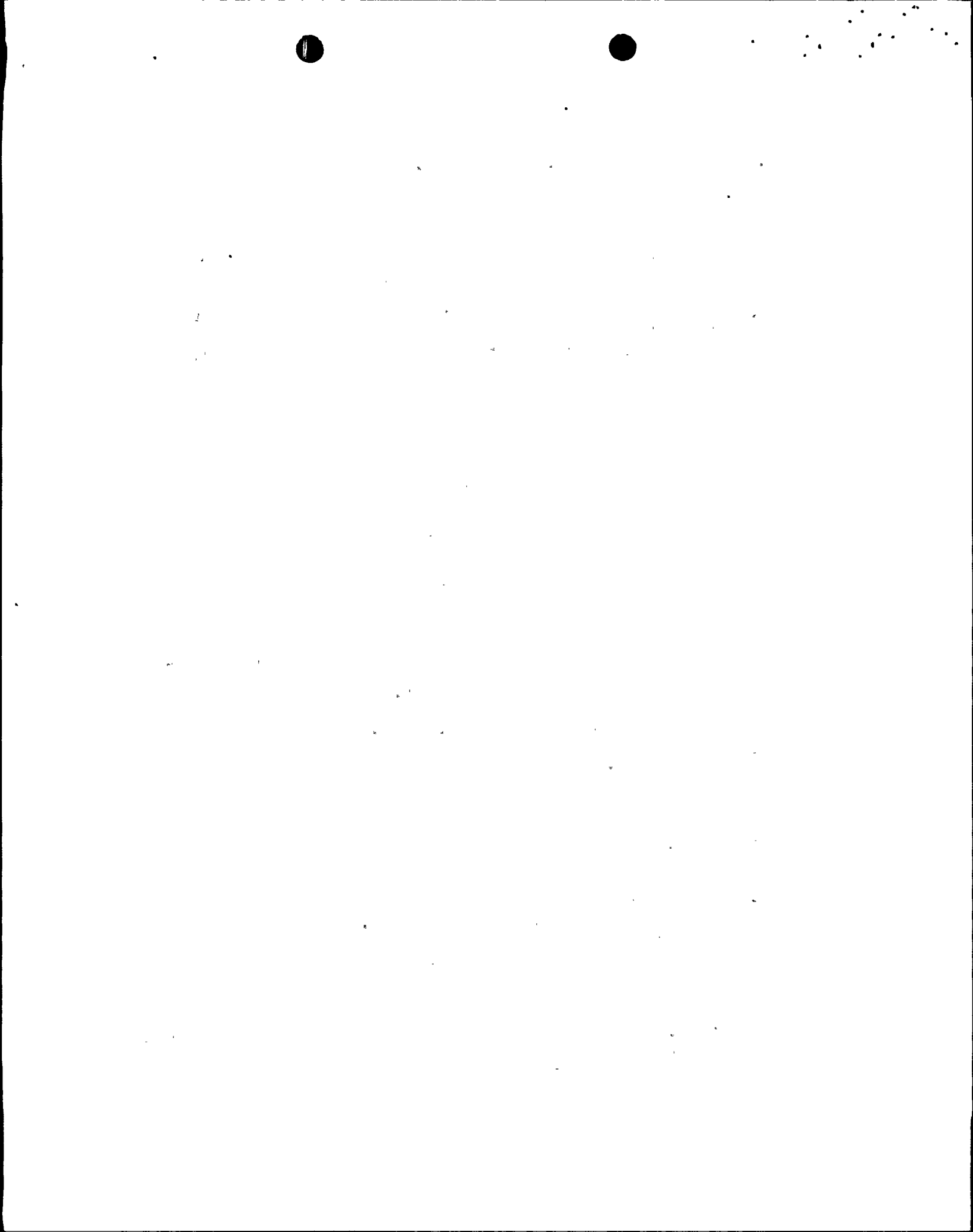
Shearon Harris Nuclear Power Plant
Draft Safety Evaluation Report Open Item 274
Supplemental Information

The acceptance criteria for piping vibration testing in FSAR Section 14.2.12.1.12 (d) (1) (b) is unacceptable. The NRC requires that the maximum alternating stress intensity, when measured by instrumentation, be .62 of the endurance limit from the ASME Code.

Response:

Carolina Power & Light Company will comply with NRC's position. The FSAR will be revised as illustrated on the attached FSAR page 14.2.12-15 in a future amendment.

(7914PSAtda)



- 4) Overall ESF actuation is within the times specified in Technical Specifications 3/4.3.2.

14.2.12.1.12 Piping Vibration Test Summary

a) Test Objectives

- 1) To verify that piping and piping restraints will withstand dynamic effects due to testing actions such as pump and valve trips, and that piping vibrations are within acceptable levels.
- 2) The program will include all ASME Class 1, 2, and 3 piping inside Containment as well as all high energy piping outside Containment and all Seismic Category I moderate energy piping outside Containment.

b) Prerequisites

- 1) A listing of the different flow modes of operation and transients to which the components will be subjected is developed.
- 2) Preservice inspection of piping hangers and restraints is completed.
- 3) The general prerequisites are met.

c) Test Method

- 1) Observe the system under operational and transient modes to identify any excessive vibration and take quantitative measurements at selected points chosen as the most likely high vibration locations.
- 2) If an observed displacement is judged to be excessive anywhere in the prescribed system, the displacement will be quantitatively measured and corrective action taken. If the displacement measured is three times the acceptance criteria, the Test shall be stopped or the excessively vibrating pipe isolated.

d) Acceptance Criteria

- 1) Displacements are acceptable if they produce stresses according to the following criteria:

(a) If the maximum amplitude is determined by visual observation or if the piping is ASME Section III, Class I, then the maximum alternating stress intensity will be limited to 0.5 of the endurance limit from Figures I-9.1 or I.9.2, ASME Code, Section III.

(b) If the maximum amplitude is determined by instrumentation, (i.e. accelerometer) and the piping is not ASME Code, Section III, Class I, then the maximum alternating stress intensity will be limited to 0.62 of the endurance limit from Figures I-9.1 or I-9.2, ASME Code, Section III.

0.62

Shearon Harris Nuclear Power Plant
Draft SER Open Item 114
Supplemental Information

NRC is concerned with fuel oil storage tank piping that interfaces with the tank liner. NRC considers the liner nozzle as part of the piping, therefore, it should be ASME Section III Class 3, Seismic Category 1. Since the pipe and tank are already fabricated, provide information indicating that piping nozzle is functionally equivalent to ASME III, will have adequate inspection, and that stress calculations show that it is within allowable limits. Confirm piping to liner is at least B31.1, QA applied is 10 CFR 50 Appendix B.

Response:

Figure 114-1 provides a typical detail for piping interfacing with the fuel oil storage tank liner. The piping interfacing with the liner is B31.1, A 106 GrB up to the mating flange which is ASTM A-105. The piping passing through the storage tank wall is provided with a backing collar which is welded to the pipe. The collar is welded to the tank liner. In our analysis of the piping System the non-safety portion of the system was modeled and the results were found to be within the allowables of the ASME Code Section III.

The piping interfacing with the liner is Ebasco Category 7 piping and as such is subject to the QA requirements and documentation considerations as specified in Ebasco specification M-30 General Power Piping Specification. As noted in the specification, the material shall be certified by submitting the Fabricator's (Southwest Fabricating & Welding Co) certificate(s) of conformance with the material specification. Nondestructive examination reports are traceable to each pipe, fitting and weld, as applicable, when such testing is required. The Certified Material Test Reports for materials and weld filler metal include the following information:

- a - Applicable code and identification, eg. ASME/ASTM/SFA specification, grade and classification, pipe or fitting size and schedule, electrode size.
- b - Mill heat or lot number.
- c - Chemical analysis.
- d - Mechanical properties.
- e - Test results when required by the ASME/ASTM/SFA material specification or Ebasco's material specification (s), eg, hydro, notch ductility test data, NDE results, delta ferrite content.

- f - Heat treatment performed by the material manufacturer to satisfy requirements of the materials specification may be reported on the CMTR, otherwise heat treat charts or a summary description of heat treat time and temperature data certified by Manufacturer or Installer shall be provided.
- g - Filler metal material test reports, with the above information, as applicable, except that Certificates of Conformance only are required for backing rings used in Piping System Categories 4, 5, 6, 7 and 8.

The records shall include, as applicable, the following information for weld joints and weld repairs:

- a - Identification of the welding procedures used
- b - Identification of the welder or welders performing the welding operation
- c - Postweld heat treatment procedure identification
- d - Indication fit-up inspection was conducted and data observed
- e - Postweld heat treatment data consisting of the automatic temperature recording charts
- f - Identification of the filler metals involved

(7900NEctda)

EBASCO SERVICES INCORPORATED

BY Y. PATEK DATE 8/22/83

SHEET 1 OF 1

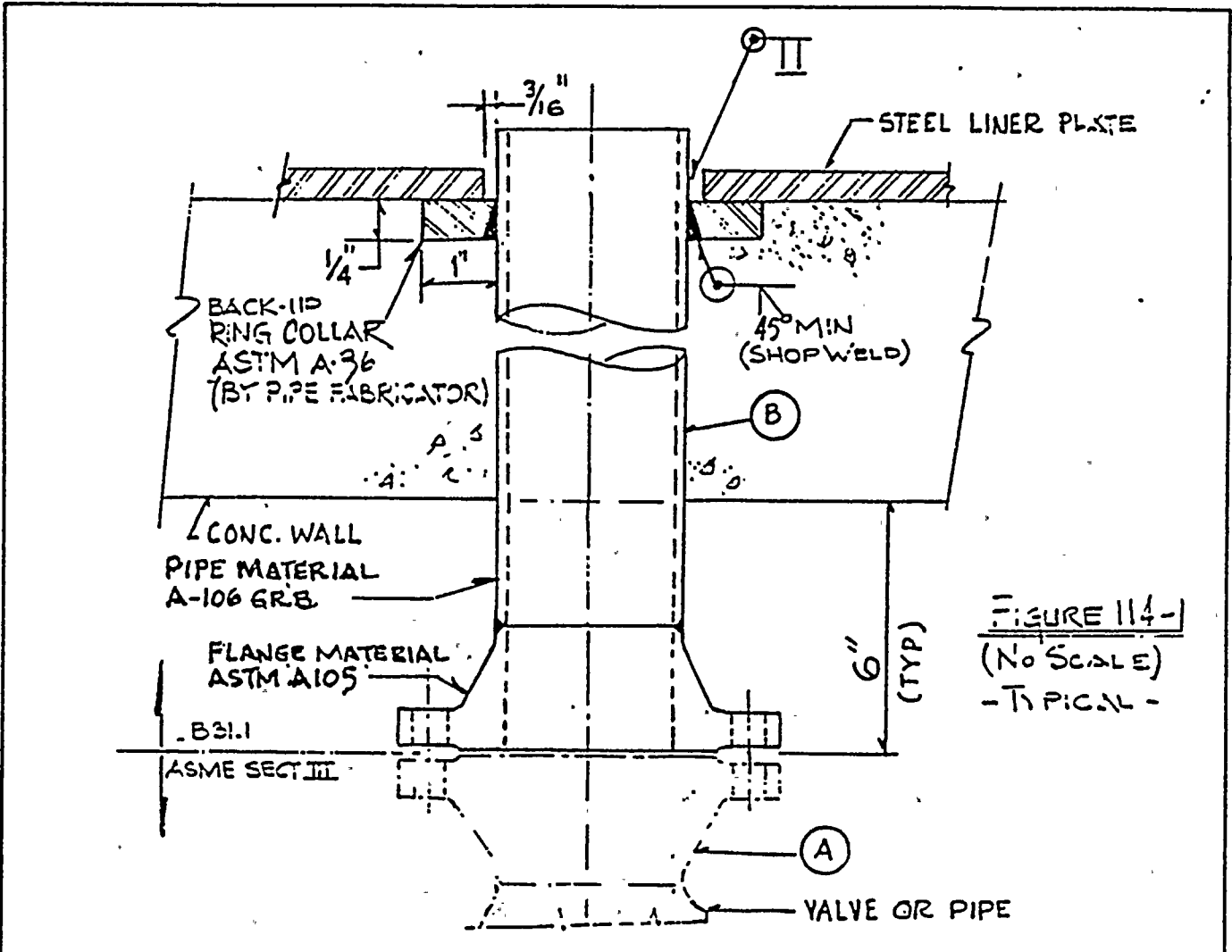
CHKD. BY Y. PATEK DATE 8/22/83

DEPT. NO. _____
OFS NO. FIG. 114-1

CLIENT CAROLINA POWER & LIGHT CO

PROJECT SHEARON HARRIS NUCLEAR POWER PLANT

SUBJECT DIESEL FUEL OIL STORAGE TANK PIPE WALL PENETRATIONS



A	B
LINE NO	PENETR. LINE NO.
3FC4-9SA-1&2	7F04-152-1&2
3F03-40SA-1	7F04-174-1
3F01-109 SA-1	7F01-176-1
3F01-168SA-1&2	7F01-180-1&2
3FC1-109 SA-2	7F01-176-2
3F03-40SA-2	7F04-174-2

A	B
LINE NO	PENETR. LINE NO.
3F04-115B-1&2	7F04-183-1&2
3F03-415B-1	7F04-175-1
3F01-1115B-1	7F01-177-1
3F01-1695B-1&2	7F01-181-1&2
3F01-1115B-2	7F01-177-2
3F03-415B-2	7F04-175-2

Shearon Harris Nuclear Power Plant
FSAR Question 430.52
Supplemental Information

Reviewer requested expansion of discussion on auxiliary lube oil pump controls/alarms.

RESPONSE: The ASME Section III auxiliary lube oil pump is a Class 1E motor-driven backup to the main engine-driven lube oil pump. The pump is required to operate only when the engine is running and the main engine driven lube oil pump has failed.

The auxiliary pump is controlled via pressure switches located in the lube oil piping. Manual and automatic control of the auxiliary lube oil pump is provided by a three-position control switch (off-auto-start) located locally on the Diesel Engine Control Panel. In the automatic mode, the auxiliary lube oil pump will start whenever the diesel engine is running and lube oil pressure becomes low (e.g. main engine driven lube oil pump failure). The pump will stop automatically when the lube oil pressure at the auxiliary lube oil pump reaches a preset high level (excessive pressure). The status indicating lights for the auxiliary lube oil pump and annunciation is provided at the DG local control panel. All the annunciators on the local Diesel Generator Control Panel are reflashed to the Main Control Boards inside the Control Room.

Power for the auxiliary lube oil pump motor is provided from the 480V safety-related MCC located in the diesel generator building, so that power to the auxiliary lube oil pump will be available during a loss of offsite power.

(7915NECtda)

Shearon Harris Nuclear Power Plant
FSAR Question 430.66
Supplemental Information

At an August 16, 1983 meeting in Bethesda Md, the NRC staff (PSB) requested information regarding the closure time of the extraction steam reverse current valves.

RESPONSE:

The Shearon Harris Extraction Steam reverse current valves are provided by Atwood and Morrill. The operation of reverse current valves when used in extraction steam application is dependent upon flow conditions for disc position. The valve disc will close under gravity at approximately the same time duration as the flow medium takes to decay. Experience has shown that full disc stroke (closure) can occur in approximately 0.4 to 1.0 seconds (see attached Atwood and Morrill letter).

(7916NECtda)

AUGUST 24, 1953

S.R. Question #33, 20

TO: EBASCO SERVICES, INC.

ATTN: EROL SURUN

SUBJ: YOUR ORDER NO. NY-435020

A&M S.O. 12999

THE A&M CHECK VALVES SUPPLIED FOR THE SUBJECT ORDER ARE DEPENDENT UPON FLOW CONDITIONS FOR DISC POSITION.

THE SIDE CLOSING AIR CYLINDER ACTS ONLY TO AID VALVE CLOSURE. THE SPRING FORCE OF THE CLOSING CYLINDER WILL NOT COMPLETELY CLOSE THE VALVE AGAINST INLET PRESSURE OR FLOW.

THE VALVE DISC WILL CLOSE UNDER GRAVITY AT APPROXIMATELY THE SAME TIME DURATION AS THE FLOW MEDIUM TAKES TO DECAY. THE VALVE DISC WILL OPEN AT APPROXIMATELY THE SAME TIME AS THE FLOW MEDIUM AND/OR PRESSURE BUILDUP INCREASES, ASSUMING THAT THE AIR CYLINDER HAS BEEN PRESSURIZED TO THE OPEN POSITION PRIOR TO PIPELINE FLOW OR PRESSURIZATION. THE AIR CYLINDER PRESSURED IN THE OPEN POSITION ALLOWS THE DISC TO ACT INDEPENDENT OF THE CYLINDER.

A&M EXPERIENCE HAS BEEN THAT FULL DISC STROKE CAN OCCUR IN APPROXIMATELY 0.4 TO MORE THAN 1.0 SECOND WITH INSTANCES OF 0.2 SECONDS UNDER CERTAIN FLOW CHARACTERISTICS.

REGARDS,

3) SYRAXOS/MICHAEL MACDONALD

ATWOOD & MORRILL CO., INC.

SHEARON HARRIS NUCLEAR POWER PLANT
FSAR QUESTION 430.73
REVISED RESPONSE

430.73 (10.3) As explained in issue No. 1 of NUREG-0138, credit is taken for all valves downstream of the Main Steam Isolation Valve (MSIV) to limit blowdown of a second steam generator in the event of a steam line break upstream of the MSIV. In order to confirm satisfactory performance following such a steam line break, provide a tabulation and description text (as appropriate) in the FSAR of all flow paths that branch off the main steam lines between the MSIV's and the turbine stop valves. For each flow path originating at the main steam lines, provide the following information:

- (1) System Identification
- (2) Maximum Steam Flow in Pounds Per Hour
- (3) Type of Shut-off Valve(s)
- (4) Size of Valve(s)
- (5) Quality of Valve(s)
- (6) Design Code of the Valve(s)
- (7) Close Time of the Valve(s)
- (8) Actuation Mechanism of the Valve(s) (i.e., Solenoid operated, motor operated, air operated diaphragm valve, etc.)
- (9) Motive or Power Source for the Valve Actuating Mechanism

In the event of the postulated accident, termination of steam flow from all systems identified above, except those that can be used for mitigation of the accident, is required to bring the reactor to a safe cold shutdown. For these systems describe what design features have been incorporated to assure closure of the steam shut-off. Describe what operator actions (if any) are required.

If the systems that can be used for mitigation of the accident are not available or decision is made to use other means to shutdown the reactor, describe how these systems are secured to assure positive steam shut-off. Describe what operator actions (if any) are required.

If any of the requested information is presently included in the FSAR text, provide only the references where the information may be found.

RESPONSE (Revised 8/26/83 per discussions with the NRC on August 16, 1983)

FSAR Figure 10.1.0-1 describes the flow paths of all branches off the main steam lines between the MSIVs and the turbine stop valves. Table 430.73-1 provides the following design information:

- (1) System Identification - branch off flow path
- (2) Maximum Steam Flow
- (3) Type of Shut-off Valve(s)
- (4) Size of Valve(s)
- (5) Quality of Valve(s)

RESPONSE (Cont'd)

- (6) Design Code of Valve(s)
- (7) Closure Time of Valve(s)
- (8) Actuation Mechanism of the Valve(s)
- (9) Motive or Power Source.

As described in Issue No. 1 of NUREG-0138, credit is taken for all valves downstream of the mainsteam isolation valves (MSIVs) to limit blowdown of a second steam generator in the event of a steam line break upstream of the MSIV. Specifically, the following accident scenario is described by Issue No. 1:

- (1) A rupture occurs upstream of the MSIV in one of the main steam supply lines.
- (2) A safety grade MSIV associated with one of the intact steam generators fails to close on demand.
- (3) In addition, the non-safety grade valves, such as turbine stop valves and control valves upstream of the turbine, or the turbine bypass valves fail to close on demand, providing a path for blowdown of a second steam generator.

This scenario exceeds the occurrences assumed under the single failure criteria and consequently has not been considered a design basis for the plant. NUREG-0138 acknowledges the fact that the probability of blowing down more than one steam generator as a result of the accident scenario described above is quite low. The Staff concluded in a survey of the operating experience of the turbine stop, control and intercept valves in operating PWR's, the reliability of these valves is of the same order of magnitude as that accepted for nuclear safety-grade components.

In addition, each branch line between the MSIVs and turbine stop valves is equipped with valving such that either remote operation (closure) of the valve from the main control room is possible or manual shut-off of the valve is available. Each motor and diaphragm operated valve can also be closed manually at the valve in the unlikely event that power or air is lost.

TABLE 430.73-1

BRANCH-OFF FLOW PATH DESCRIPTION	MAXIMUM STEAM FLOW lbs./hr x10 ³	TYPE OF VALVE(S) (NORMAL POSITION)	SIZE OF VALVE(S)	QUALITY OF VALVE(S)	DESIGN CODE OF VALVES	CLOSURE TIME OF THE VALVES	ACTUATION MECHANISM OF VALVE(S)	POWER, SOURCE OF VALVE(S)
<u>Steam Dump to Atmosphere</u>								
1. Eight (8) Control Valves	705 (Each)	Globe (Closed)	8"	NNS	B31.1	5 Sec.	Diaphragm	Air
2. Eight (8) Block Valves	705 (Each)	Gate (Locked Open)	8"	NNS	B31.1	NA	Hand	NA
<u>Steam Dump to Condenser</u>								
Six (6) Control Valves	705 (Each)	Globe (Closed)	8"	NNS	B31.1	5 Sec.	Diaphragm	Air
<u>Gland Seal Supply</u>	16	Globe (Locked Open)	4"	NNS	B31.1	NA	Hand	NA
<u>MSR Supply</u>								
1. Control Valves	966	Globe (Open)	12"	NNS	B31.1	65 Sec.	Diaphragm	Air
2. & Bypass Control Valves	966	Globe (Open)	4"	NNS	B31.1	11 Sec.	Diaphragm	Air
3. Block Valves (Motor)	966	Globe (Open)	12"	NNS	B31.1	10 Sec.	Motor	AC
		Globe (Open)	6"	NNS	B31.1	10 Sec.	Motor	AC
Block Valves (Manual)	966	Globe (Open)	12"/6"	NNS	B31.1	NA	Hand	NA

TABLE 430.73-1 (Cont'd)

BRANCH-OFF FLOW PATH DESCRIPTION	MAXIMUM STEAM FLOW lbs./hr x10 ³	TYPE OF VALVE(S) (NORMAL POSITION)	SIZE OF VALVE(S)	QUALITY OF VALVE(S)	DESIGN CODE OF VALVES	CLOSURE TIME OF THE VALVES	ACTUATION MECHANISM OF VALVE(S)	POWER, SOURCE OF VALVE(S)
<u>Back-up Auxiliary Steam Supply</u>	117.	Gate (Closed)	6"	NNS	B31.1	10 Sec.	Motor	AC
<u>Miscellaneous Vents & Drains</u>	—	Globe (Closed)	1"	NNS	B31.1	NA	Hand	NA

Shearon Harris Nuclear Power Plant
DSER Open Item No. 5
(Supplemental Information)

Question:

As a result of the cancellation of Units 3 & 4, the reviewer requested additional details relative to the design of the Retaining Wall West of the Fuel Handling Building.

Response:

The additional information requested by the reviewer is given below.

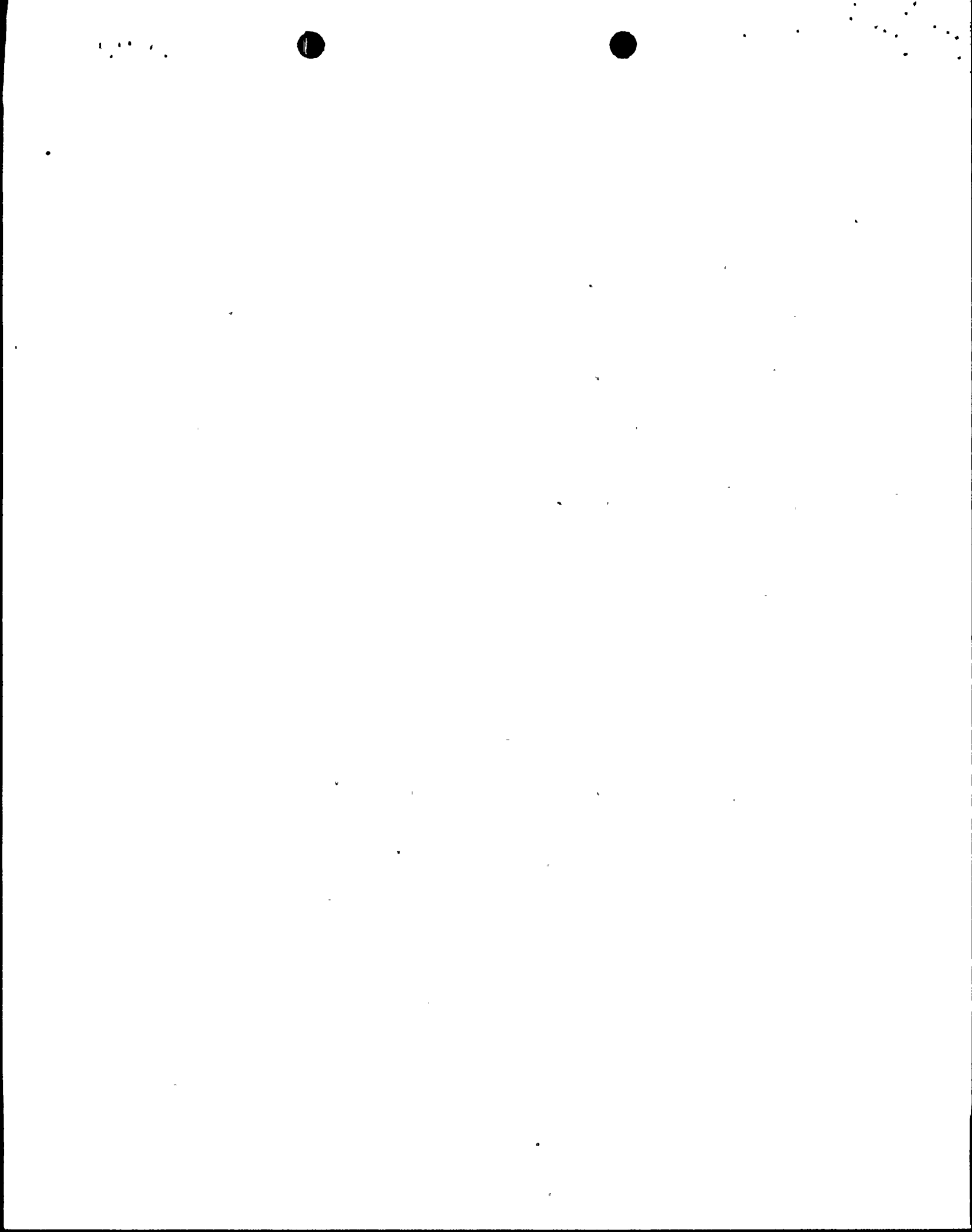
1. GENERAL

Due to the cancellation of Units 3 and 4, the reactor auxiliary buildings, containment buildings and tank buildings for Units 3 and 4 have been deleted, and a retaining wall west of the Fuel Handling has been provided to separate the Fuel Handling from the plant grade fill. The plant grade fill has; however, been extended to the north-west corner of the Waste Processing Building. The stability of the Waste Processing Building and structural integrity of Waste Processing Walls have been reviewed to satisfy the design criteria for the additional lateral soil and hydrostatic pressures.

The retaining wall has been physically separated from the Fuel Handling Building by a gap of three (3) feet along the length of the wall and a gap of approximately three (3) inches at the north end to allow for movement of the wall during an SSE. Therefore, the design of the Fuel Handling Building is not affected by the retaining wall. The layout of the retaining wall is shown in Ebasco drawing CAR-2167-G-2194 (see Attachment No. 4).

2. CLASSIFICATION OF RETAINING WALL

In addition to full hydrostatic pressure at the level of the ground water table specified for the site, the retaining wall has been seismically designed in accordance with Positions C-2 and C-4 of Regulatory Guide 1.29 because the plant grade fill west of the retaining wall does not support any safety related structure, system or subsystem, and as such, the retaining wall is not required for safe shutdown of the plant. Moreover, there is no safety related equipment west of the "N" line wall of the Fuel Handling Building, which is more than 75 feet east of the 45 foot high retaining wall. Since the design precludes gross failure of the retaining wall, the seismic design of the Fuel Handling Building and of any safety systems within the building would not be affected.



3. DESCRIPTION OF RETAINING WALL

The retaining wall consists of two rows of reinforced concrete pipes erected one over the other and a reinforced concrete wall on top of the pipes. The pipes are filled with concrete and are held back by tie rods and deadmen. The details of the retaining wall are shown in Ebasco drawings CAR-2167-G-2194, G-2195, and G-2196 (see Attachment Nos. 4, 5, 6).

4. DESIGN OF RETAINING WALL

The retaining wall has been designed for static and dynamic SSE seismic load conditions for the lateral soil pressures in combination with hydrostatic pressure, as shown in the pressure diagram (see Attachment No. 1). The wall has been designed for full hydrostatic pressure in case the filters become clogged for any reason even though it is highly improbable since the drainage material, is crushed rock transition filter material in accordance with Ebasco Specification CAR-SH-CH-4 (see FSAR Appendix 2.51). The transition filter material is well graded from minus three (3) inch to sieve #200 (see Attachment No. 2).

5. PROPERTIES OF SOIL BACK FILL

The soil backfill west of the retaining wall is random fill in accordance with Ebasco Specification CAR-SH-CH-8 (see FSAR Appendix 2.51). The following properties for the backfill have been assumed for long term stability of the retaining wall against lateral soil pressures.

Saturated Unit weight of soil	= 135 pcf
Coefficient of internal friction	= 30°
Cohension in the soil	= 0 ksf

The short term or construction condition properties of the backfill are:

Optimum moisture content fill	
Coefficient of internal friction	= 26°
Cohesion	= 1.2 ksf
Wet fill	
Coefficient of internal friction	= 9°
Cohesion	= 2.0 ksf

The above properties are based upon the tests performed on as built site conditions for the backfills.

The short term backfill properties have been taken into account and do not govern the design of the retaining wall even if the coefficient of internal friction is taken as 0° and cohesion as 2.0 ksf.

6. STABILITY OF WALL

For the design of deadmen the backfill is assumed to be submerged to EL 251 feet (Design basis groundwater table for the plant island).

The ultimate capacities of the deadmen and tie rods for a unit length of retaining wall equal to the outside diameter of the reinforced concrete pipe (11.4 feet) and the factors of safety for the static and dynamic conditions of loading are given in the attached table (see Attachment No. 3). If the saturated weight of the soil is assumed to be 140 pcf the increase in unit weight is only 4% and the factors of safety listed in Attachment No. 3 will still be greater than the design requirements.

The tie rods are protected against corrosion by coating the tie rods with Epoxy and also by electrically grounding all reinforcing steel bars and tie rods as indicated by the notes on Ebasco drawing CAR-2167-G-2194 at coordinate F-18 (see Attachment Nos. 4 and 7).

To avoid erosion of soil near deadmen the plant grade will be protected by turf and no storm drain or other pressure pipes will be provided parallel to and within 50 feet in front of the deadmen.



BY A. BAIG DATE 8-19-83

SHEET 29 OF _____

CHKD. BY W HSU DATE 8-23-83

CAROLINA POWER & LIGHT COMPANY

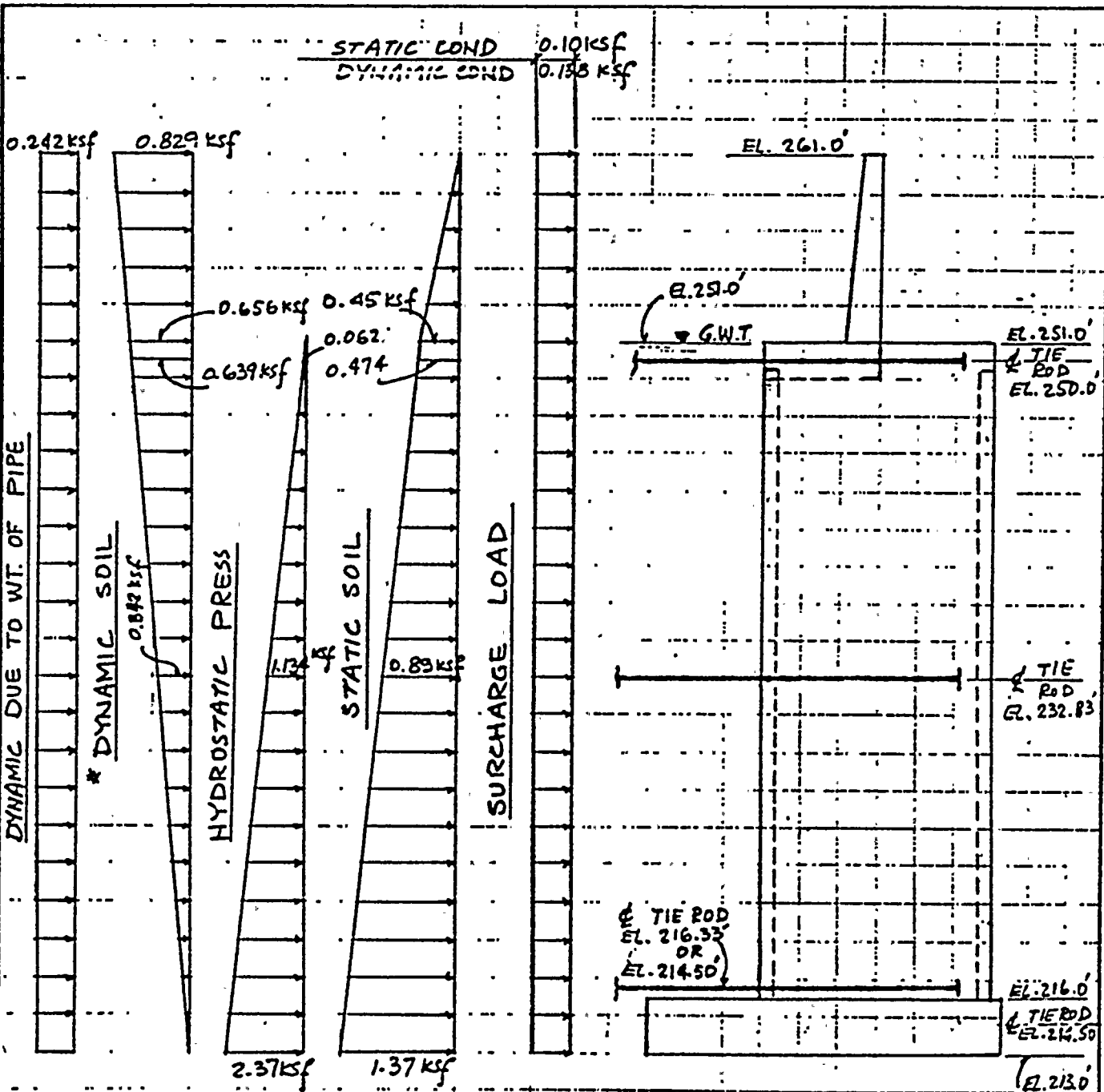
OFFS NO. 6418

DEPT. NO. 650

CLIENT SHEARON HARRIS NUCLEAR POWER PLANT

PROJECT 1984 86 900,000 KW UNITS 1, 2

SUBJECT FHB - RETAINING WALL - INCLUDING HYDROSTATIC PRESS



* DYNAMIC SOIL PRESSURE INCLUDES HYDRO DYNAMIC PRESSURE
(BY USING SATURATED WEIGHT OF SOIL)

ATTACHMENT 2

II. EMBANKMENT TRANSITION FILTERS

Materials

The filters shall consist of processed, very well graded coarse mixtures of cohesionless materials obtained from hard, dense durable rock such as granite, sandstone or conglomerate and not from elaystone or siltstone, and shall be graded as given below.

a - Transition filter zones for Main Dam shall conform to the following gradation:

<u>Sieve U.S. Stand Sq Mesh</u>	<u>Percentage by Weight Passing</u>		
	<u>Fine Filter</u>	<u>Coarse Filter</u>	<u>Trans. Filter</u>
6 in.	-	100	-
3 in.	-	86-100	100
1-1/2 in.	-	66-88	82-100
1/2 in.	100	45-68	60-90
3/8 in.	96-100	40-62	-
No. 4	83-100	30-48	45-75
No. 8	70-93	18-38	35-65
No. 40	30-34	0-21	10-40
No. 100	10-32	-	0-25
No. 200	0-13	-	0-15

b - Transition filter zone for Auxiliary Dam shall conform to the following gradation:

<u>Sieve U.S. Standard Sq Mesh</u>	<u>Percentage by Weight Passing</u>
3 in.	100
1-1/2 in.	82-100
1/2 in.	60-90
No. 4	45-75
No. 8	35-65
No. 40	10-40
No. 100	0-25
No. 200	0-15

c - Lateral Filter Blankets:

The elevation 214 filter blanket will be located in the downstream random rockfill shell between stations 17+00 and 29+00. The elevation 230 filter blanket will be located between stations 17+00 and 31+00 within the downstream random rockfill shell. Both filters will conform to the gradation requirements specified in Ebasco Specification CAR-SH-CH-8 for "crushed rock backfill". The filter blankets will be compacted to at least 85 percent relative density. These filters will be compacted in twelve inch lifts to a total thickness of three feet.

ATTACHMENT 3

RETAINING WALL STABILITIES

<u>ITEM</u>	<u>ELEVATION</u>	<u>ULTIMATE CAPACITY</u> <u>(kips)</u>	<u>FACTOR OF SAFETY</u>	
			<u>STATIC CONDITION</u>	<u>DYNAMIC CONDITION</u>
Deadman	250.00	428	2.73	1.17
	232.83	852	1.50	1.27
	214.50	674	1.83	1.63
Tie Rod	250.00	648	4.12	1.78
	232.83	864	1.50	1.28
	214.50	616	1.67	1.49
Middle Wall				
Sliding at Base **225.00(+)			7.54	6.23

NOTES:

Minimum factors of safety required are 1.5 for static, and 1.1 for dynamic conditions, respectively.

** Elevation of top of rock beneath base varies slightly.

