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DESCRIPTION:
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PLANT NAME: Oconee, Units 1-2-3

ENCLOSURES:
SUPPL # 1 to Report OS-73.2 "Analysis of Effects Resulting from Postulated Piping Breaks Outside Containment"

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DUKE POWER COMPANY

POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28201

A. C. THIES
SENIOR VICE PRESIDENT
PRODUCTION AND TRANSMISSION

P. O. BOX 2178

June 22, 1973

Mr. Angelo Giambusso
Deputy Director for Reactor Projects
Directorate of Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

Re: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Mr. Giambusso:

On April 25, 1973, Duke Power Company transmitted for your review MDS Report No. OS-73.2, "Analysis of Effects Resulting from Postulated Piping Breaks Outside Containment for Oconee Nuclear Station, Units 1, 2, and 3." A meeting was held with AEC/DOL on June 12, 1973 to discuss this report. It was agreed to supplement this report with additional information needed to complete the review of the Oconee high energy line break analysis. Duke Power Company transmits herewith Supplement 1 to Report No. OS-73.2. This supplement will be incorporated by reference into the Oconee license application.

Very truly yours,

A. C. Thies P.H.B.

A. C. Thies

ACT:vr

Attachment

5232

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

DOCKETS 50-269, -270, AND -287

SUPPLEMENT 1

TO

MDS REPORT NO. OS-73.2

ANALYSIS OF EFFECTS RESULTING FROM

POSTULATED PIPING BREAKS OUTSIDE CONTAINMENT

FOR

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

The following information is voluntarily submitted in response to informal questions asked by the Division of Reactor Licensing during the June 12, 1973 meeting held in Bethesda, Maryland.

Docket 50-269, -270, -287
Supplement 1 to Report No. OS-73.2
June 22, 1973

Question 1

Describe the qualification data for electrical components required to function during and after a high energy line break.

Answer

Electrical equipment required to function during and after a high energy line rupture has the following qualifications:

1. HP Injection Valve Motor Operator Cables

   Samples of power and control cables which are of armored construction were subjected to temperatures in excess of 380°F for a period of 60 minutes in an autoclave test. Functional capability of the cables was demonstrated by energization during and after the test. No loss of function occurred. The postulated event environment is a 332°F peak temperature transient of less than one second duration. The required duration of operation time for the cables is less than 30 seconds after the event. These cables are, therefore, conservatively qualified for their application.

2. Low Reactor Pressure Sensor Cables

   Samples of these cables which are of armored construction were subjected to temperatures in excess of 380°F for a period of 60 minutes in an autoclave test. Functional capability of the cables was demonstrated by energization during and after the test; no loss of function occurred. The postulated event environment is a 332°F peak temperature transient of less than one second duration. The required duration of operation time for the cables is less than six seconds after the event. These cables are also very conservatively qualified for their application.

3. HP Injection Valve Operator (HP-26 & 27)

   These valve operators have Class "H" insulation and are the same type as are utilized on safety systems in the containment. They have been qualified to withstand a temperature of 340°F without loss of function. The Class "H" insulation as rated per NEMA standard MG-1-12.42 is qualified for 180°C (356°F) hot spot temperature at continuous operation. The allowable temperature rise of the operator over an ambient of 80°F would be 267°F. The actual temperature rise as determined by analytical calculations, based on a 380°F temperature transient of eight seconds, would be 111°F. This would then yield a 191°F temperature of the valve operator for the worst case condition as compared to the permissible temperature function in less than 30 seconds after an event. Due to their containment qualification testing, their Class "H" insulation and their short operating cycle, they are conservatively qualified for their application.

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June 22, 1973

Question 2

What pressures can the various penetration room critical walls, floors and ceilings withstand?

Answer

Area	Max. Allowable Pressure (psi)
Elev. 809'-3" Floor Slab	
Col. 64-71	6.61
Elev. 838'-0" Ceiling Slab	
Col. 64-65	2.81
Col. 65-66	4.67
Col. 66-71	8.54
Battery Room Wall (With proposed reinforcing)	5.69
Control Room Wall	17.36

Question 3

Describe how the battery room walls will be reinforced to prevent overpressurization, and state stress levels used for reinforcing members.

Answer

Battery room walls will be reinforced with a 1/2" thick steel plate skin stiffened by wide flange beams anchored to the building structural concrete. Maximum stress in 1/2" plate skin = 25,795 psi. Maximum stress in wide flange beam = 36,000 psi. All materials are ASTM A36 carbon steel selected for a design pressure of 5.7 psi.

(Revised plate thickness from 5/8" to 1/2")

Docket 50-269, -270, -287
Supplement 1 to Report No. OS-73.2
June 22, 1973

Question 3

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Question 4

Describe the existing restraint system for the Main Steam lines and any proposed modifications.

Answer

Present restraints for the Main Steam System consist of the following:

1. Variable and constant support spring hangers designed for normal piping dead loads.

2. Rigid hangers designed for normal dead loads, thermal loads, seismic loads, and safety valve thrust loads where applicable.

3. Horizontal rigid and/or hydraulic restraints for thermal, seismic and wind loads as applicable.

4. Reactor Building wall rigid anchor designed for normal dead loads, thermal loads, seismic loads, and pipe rupture loads.

Proposed modifications are to add a line interaction restraint between the two Main Steam lines immediately upstream of the Main Steam safety valves for the purpose of preventing a line break in Main Steam line B as a result of Main Steam line A terminal end break. The restraint will be designed to the following criteria:

1. Prevent Main Steam line A from interacting with Main Steam line B safety valves.

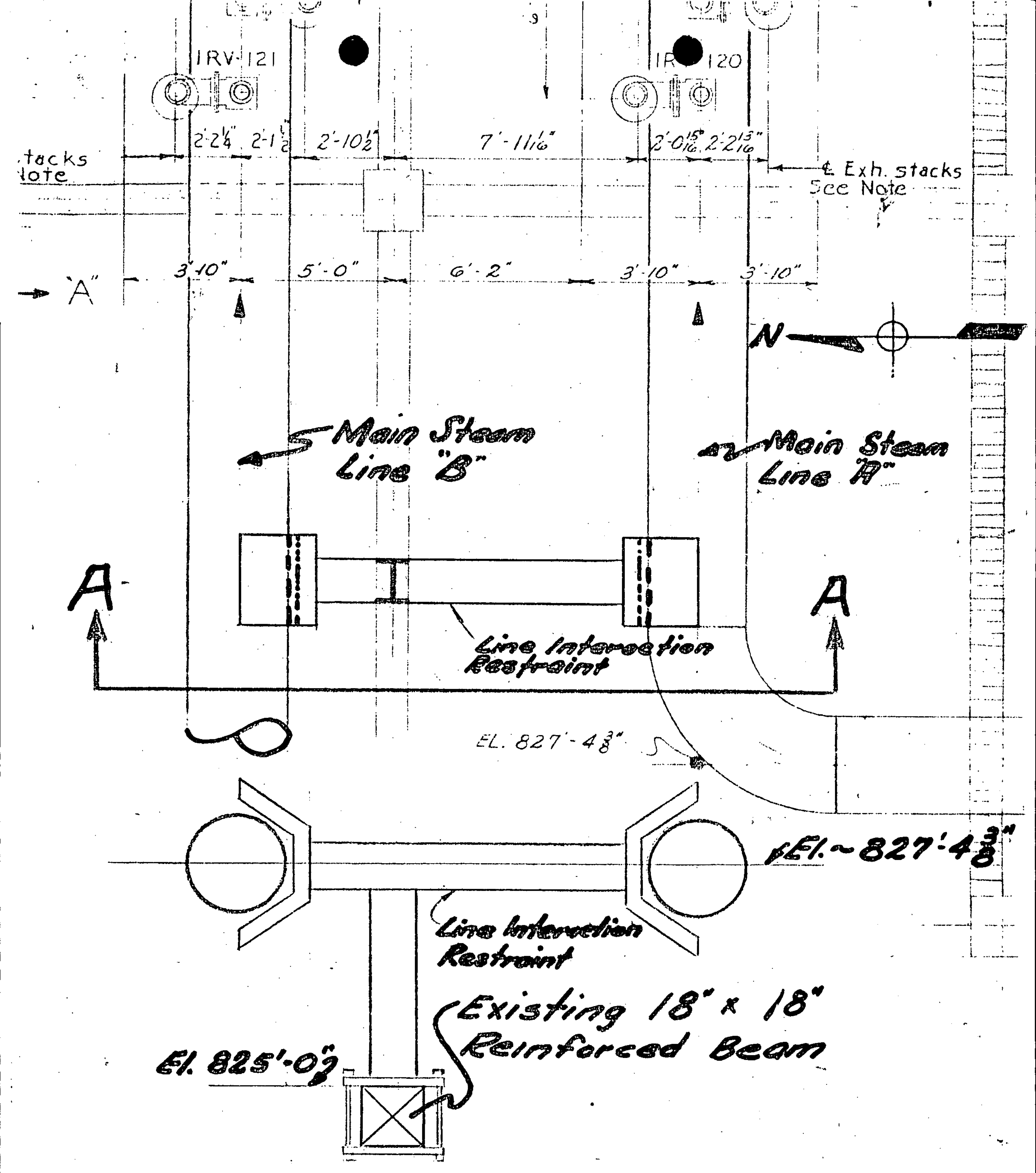
2. Normal thermal movements of either Main Steam line are not to be impaired

3. Mechanical loading and stress analysis aspects are:

   a) All load bearing members will be designed to 0.9 yield stress for primary stresses.

   b) A dynamic load factor of 3.0 will be used for applicable load bearing members of the restraint.

Figure 4.2-1.b_3 is a preliminary conceptual design of the proposed restraint.



SECTION A-A

OCONEE NUCLEAR STATION
MAIN STEAM SYSTEM
LINE INTERACTION RESTRAINT
(PRELIMINARY)
FIGURE 4.2-1.b3

Docket 50-269, -270, -287
Supplement 1 to Report No. OS-73.2
June 22, 1973

<u>Question 5</u>

Provide the design criteria for the proposed feedwater line restraints.

<u>Answer</u>

Restraints for the Main Feedwater Reactor Building terminal end break are designed to the following criteria:

1. Restrain the lines to prevent pipe whip.

2. Limit the double ended break gap to a 0" gap insofar as possible based on thermal expansion tolerances.

3. Prevent jet impingement as the result of terminal end break.

4. Limit and direct flow of leakage away from vulnerable mechanical and electrical equipment.

5. Mechanical Loading and Stress Analysis aspects are:
   a) The guard pipe will be designed for full system design pressure and temperature, 1275 psig @ 475°F.
   b) All load bearing members will be designed to 0.9 yield stress for primary stresses.
   c) A dynamic load factor of 3.0 will be used for applicable load bearing members of the restraint.

Question 6

Provide an analysis of the station's ability, after design changes are completed, to mitigate a postulated feedwater line break in the turbine building in the area of the 4160 volt switchgear.

Answer

The consequences of the postulated double-ended break of main feedwater line A at the emergency feedwater connection inside the turbine building has been analyzed under the premise that the design changes as described in Section 4 of the high energy pipe break study, "Analysis of Effects Resulting from Postulated Piping Breaks Outside Containment for Oconee Nuclear Station, Units 1, 2, and 3," assure that a redundant emergency feedwater supply line is available to each steam generator for long-term core cooling. The extent of damage to other equipment is assumed to be as follows:

1. Feedwater valve FDW-33 is destroyed.

2. The pipe whip of feedwater line A severs emergency feedwater line connection to main feedwater line B and destroys feedwater valve FDW-42, thus eliminating the normal channels of main and emergency feedwater flow to either steam generator.

3. The 4160 volt switchgear 1TC, 1TD, and 1TE is lost due to direct water/steam impingement.

Without the additional emergency feedwater supply lines to each steam generator the immediate consequences of the accident are similar to those presented in Section 14.1.2.8.3, "Results of a Complete Loss of All Station Power Analysis," of the Final Safety Analysis Report. As further stated in that section, immediate operation of the emergency feedwater system is not of a critical nature, i.e., the reactor can sustain a complete loss of electric power without emergency cooling for about 23 minutes before the pressurizer is filled with reactor coolant and for an additional period of 83 minutes before boil-off of the coolant will start to uncover the core. However, with the addition of emergency feedwater to either steam generator prior to filling the pressurizer with reactor coolant, sufficient decay heat removal can be provided to assure core coverage and the reactor coolant system can be maintained for an extensive period of time in a hot shutdown condition. Once power is restored to the high pressure injection pumps, the reactor coolant system can then be cooled in an orderly manner with an adequate supply of borated water for coolant makeup and boron control.

The sequence of events and resulting consequences for the postulated feedwater line break with the availability of emergency feedwater are as follows:

1. Termination of all feedwater results in a reduction in secondary system heat removal capability. Feedwater line check valves prevent a secondary system blowdown through the feedwater line break.

2. Loss of electric power results in gravity insertion of control rods. Even if power is available after the break, increased reactor coolant system temperature and pressure result in a high pressure reactor trip within 15 seconds after the loss of feedwater.

3. Following reactor trip, turbine trip occurs with the closure of the turbine stop valves.

4. The main steam safety valves actuate after the turbine stop valves close to prevent excessive temperatures and pressures in the reactor coolant system. The safety valves close after about 20 seconds of steam relief if steam flow through the turbine bypass valves is available to relieve excess steam and provide for decay heat removal.

5. Thermal equilibrium is then re-established in the reactor coolant system, i.e., the heat removal rate provided by steam relief is equal to the core decay heat input.

6. Once the steam generator liquid inventories have been vaporized in about nine minutes, the RCS will begin to heat up with actuation of the pressurizer safety valves at 2515 psia within five minutes after the steam generators are dry.

7. Steam relief by the pressurizer safety valves will continue until emergency feedwater flow is established to either steam generator within 15 minutes after the break. Since the addition of emergency feedwater to either steam generator occurs within the 23 minute period described in FSAR Section 14.1.2.8.3, and is sufficient for decay heat removal, the pressurizer is prevented from filling with reactor coolant.

8. The operator can then re-establish thermal equilibrium and begin plant cooldown at this time by emergency feedwater control and steam relief to the condenser or the atmosphere.

9. Prior to plant cooldown, the operator must manually restore power to any one of three high pressure injection pumps. Power to pumps is not a part of the 4160 volt switchgear affected by the accident but comes directly from the 4160 volt main feeder buses (see Figure 8-3 of the FSAR). These actions can be easily accomplished within a 30 minute time period.

10. The operator utilizes high pressure injection flow for makeup and boron control during plant cooldown.

The postulated feedwater line break results in a reactor trip followed by reactor coolant system heatup prior to the orderly control of the transient by the operator so that the core can always be maintained in a subcritical condition. Also, the reactor coolant system pressure does not exceed code design limits at any time during the transient. Therefore, the core integrity is maintained during this event and an orderly cooldown to cold shutdown condition is accomplished by the installation of the redundant emergency feedwater supply lines.

Question 7

Qualify the remote probability of a Reactor Building terminal end pipe break for the east Main Steam line, and describe any actions and circumstances supporting the acceptability of Duke's design and analysis.

Answer

Postulated pipe break 1.b as described in Table 2.1-1 and Figure 2.1-1.b is highly unlikely based on the following considerations. (See MDS Report No. OS-73.2)

1. As described in 1.2.2, the Main Steam System is cold pulled such that thermal stresses are essentially eliminated while the system is in normal operation. The maximum thermal stress at the terminal end is about 1100 psi or 4% of the ANSI B31.1.0 Code (1967) allowable stress during operation. In addition, all stress criteria established by the AEC are satisfied.

2. Quality of the Main Steam System approaches that of a nuclear power piping system as can be noted from 1.2.3. In essence, all materials are uniquely traceable, all welds are 100% x-ray quality, and the system is seismically designed.

3. Overpressure capability based on actual wall thickness and ANSI B31.1.0 (1967) Code equations is 20% as established in 1.2.2.

Circumstances and actions supporting the acceptability of Duke's design and analysis are as follows:

1. As can be noted from figures 4.2-1.b_1 and 4.2-1.b_2, the blowdown from the postulated Main Steam break will probably not produce full pressure in the penetration room as the flow is directed toward the north wall and away from the penetration room. In addition, the break location is on the fringe area of the penetration room.

2. Pressure relief of the penetration room is assured by removing the north wall and extending the vent area eastward by removing the existing restroom between columns 64 and 65. The north wall of the restroom will also be removed and light blowout panels of 1 lb/ft^2 or less will form the new wall for all vent areas.

3. Duke will increase the inservice inspection to include the metal surface inspection of the postulated break area every 5 years to detect any surface defects.

Question 8

Have the effects of critical crack impingements been analyzed?

Answer

As described in 2.1.1, the consequences of critical cracks were analyzed and accounted for in the report. (See MDS Report No. 73.2)

Docket 50-269, -270, -287
Supplement 1 to Report No. OS-73.2
June 22, 1973

Question 9

Provide a detailed schedule for the completion of proposed station modifications.

Answer

The following station modifications have been reviewed and estimated dates of completion are presented for all three units. Until modifications are completed on Unit 2, the interim inspection described for Unit 1 will also be accomplished for Unit 2.

Unit 1 Schedule

Modification	Complete Design	Materials Delivery	Complete Const.
Penetration Room Blowout Panels	7-1-73	9-1-73	10-1-73
Reinforce Pene. Room Walls	7-1-73	9-1-73	10-15-73
Shield LP Inj. Line (Radiation)	7-1-73	10-1-73	11-1-73
Cable Tray Impingement Deflector	7-1-73	7-15-73	8-1-73
Emerg. FDW Bypass Piping	7-1-73	8-15-73	11-1-73
valves	Complete	10-1-73	11-1-73
hangers & restraints	7-23-73	10-1-73	11-1-73
Main Feedwater Restraints	6-22-73	10-1-73	11-1-73
Main Steam Restraints	7-10-73	10-10-73	11-1-73

Unit 2 Schedule

Modification	Complete Design	Materials Delivery	Complete Const.
Penetration Room Blowout Panels	8-1-73	9-1-73	10-1-73
Reinforce Butt Room Walls	8-1-73	9-1-73	10-15-73
Shield LP Inj. Line		N/A	
Cable Tray Impingement Deflector	7-15-73	8-1-73	9-1-73

Docket 50-269, -270, -287
Supplement 1 to Report No. OS-73.2
June 22, 1973

Modification		Complete Design	Materials Delivery	Complete Const.
Emerg. FDW Bypass	Piping	7-1-73	8-15-73	11-1-73
	valves	Complete	10-1-73	11-1-73
	hangers & restraints	8-15-73	11-15-73	12-15-73
Main Feedwater Restraints		6-22-73	10-1-73	11-1-73
Main Steam Restraints		7-10-73	10-10-73	11-1-73

Unit 3 Schedule

As described in 4.4, all Unit 3 modifications will be complete prior to unit startup.