HATCH UNIT 2
MAIN STEAM ISOLATION VALVE LEAKAGE CLOSURE
SEISMIC EVALUATION

SEPTEMBER 13,1993

INTRODUCTION

The primary components to be relied upon for pressure boundary integrity in resolution of the BWR MSIV leakage issue are: (1) the main turbine condenser, (2) the main steam lines from the turbine stop and bypass valves, and (3) the main steam turbine bypass and drain line piping to the condenser. Earthquake experience has demonstrated that the welded steel pipe in these systems is seismically rugged. Based on post-earthquake reconnaissance, the BWROG seismic experience study has identified limited realistic seismic hazards, including support design attributes and proximity interaction issues, as potential sources of damage on a limited number of components. A review and evaluation were performed for Hatch Unit 2 to ensure that no such issues are present, thus providing reasonable assurance of the integrity of these systems and components. This document summarized the results of the review and evaluation.

1.0 SCOPE OF REVIEW

The Condenser forms the ultimate boundary of the leakage pathway. Boundaries were established upstream of the condenser by utilizing existing valves to limit the extent of the seismic verification walkdown. The boundaries are shown in Figure 1. The boundary valves were selected using the following criteria:

A) Normally closed valves that will not open and can be assured to remain closed (includes 1st and 2nd stage MSR drain valves 2N11-F020A&B, 2N38-F011A&B and miscellaneous manual drain valves)

B) Normally open valves that can be assured to close and remain closed (includes the Main Steam Stop Valves, the Main Steam By-Pass valves, the Off-Gas Preheater valves 2N62-F010A&B, Steam Jet Air Ejectors valves 2N32-F008A&B (first stage), 2N32-F044A&B (second stage), 2N32-F009A&B (third stage), the Reactor Feed Pump valves 2N11-F012A&B

C) Valves that may require operator action to assure closure (Moisture Separator Reheater valves 2N11-F004A&B, Steam Seal valve 2N11-F010)

Valves 2B21-F019 & F021 are the valves that must function following a LOCA and a seismic event. Valve 2B21-F021 is normally closed, but it has Class 1E power to open it in case of a loss of offsite power. 2B21-F019 is a normally closed isolation valve, and can be assumed to maintain its closed status as designed. Valve 2B21-F020 is normally open.

A seismic verification walkdown was performed to assure that the main condenser and steam piping systems that are not seismically designed fall within the bounds of the design characteristics of the seismic experience data base contained in Appendix D to the BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control System (NEDC-31858P, Rev. 1). The seismic experience data base piping and equipment designs have demonstrated good seismic performance, and the piping and equipment designs at Plant Hatch, Unit 2 are equivalent to that contained in the seismic experience data base. Conditions that might lead to piping configurations that are outside the bounds of this conventional piping were noted during the walkdown. Table 3 summarizes the identified conditions (termed "outliers"), and their resolution. Note that some outliers were resolved by demonstrating analytically that the outlier did not create hazards beyond the seismic inertial loading. These hazards include interaction, differential displacement, and failure/falling. Other outliers required corrective action as noted in the table.

Where analysis was used to resolve outliers, estimates of the realistic median-centered in-structure response spectra were employed. These estimated values are more representative of the actual response of the building during a seismic event than the original Design Basis Earthquake. Estimation involved a comparison of the Hatch Unit 2 Turbine Building and Control Building structural models, and frequency shifting and scaling based on the median-centered response spectra generated for the Control Building Seismic Margin Assessment for Unit 1.

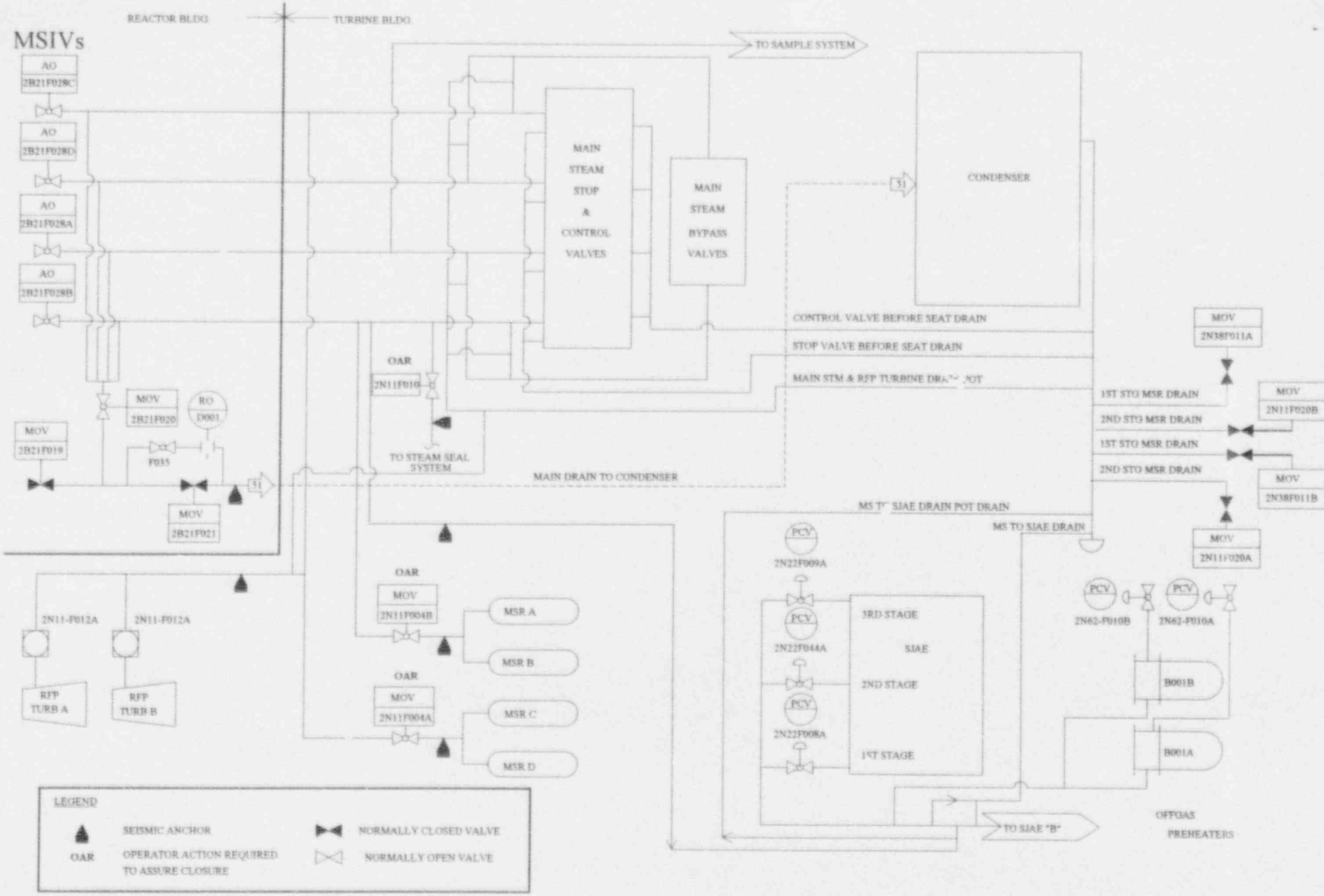


Figure 1: Schematic of Leakage Control Path

2.0 TURBINE BUILDING

Performance of the turbine building during a seismic event is of interest to the issue of MSIV leakage only to the extent that non-seismically designed structures and components should survive and not degrade the capabilities of the selected main steam and condenser pathways. A BWROG survey of this type of industrial structure has, in general, confirmed that excellent seismic capability exists. There are no known cases of structural collapse of either turbine buildings at power stations or structures of similar construction.

The design of the Hatch Unit 2 turbine building is dominated by the tornado loading, which was included in the original design. This loading exceeds seismic loading based on the Unified Building Code (UBC), the original design basis earthquake, or realistic, median-centered seismic data (see Section 2.1.6). Figure 2 shows a comparison of the base shear for the design wind load, the UBC design earthquake, and the estimated median-centered earthquake with a 0.15 g ground motion. Appendix D, Section 4.4, of NEDC-31858P summarizes the earthquake experience data as related to the Hatch Unit 2 turbine building.

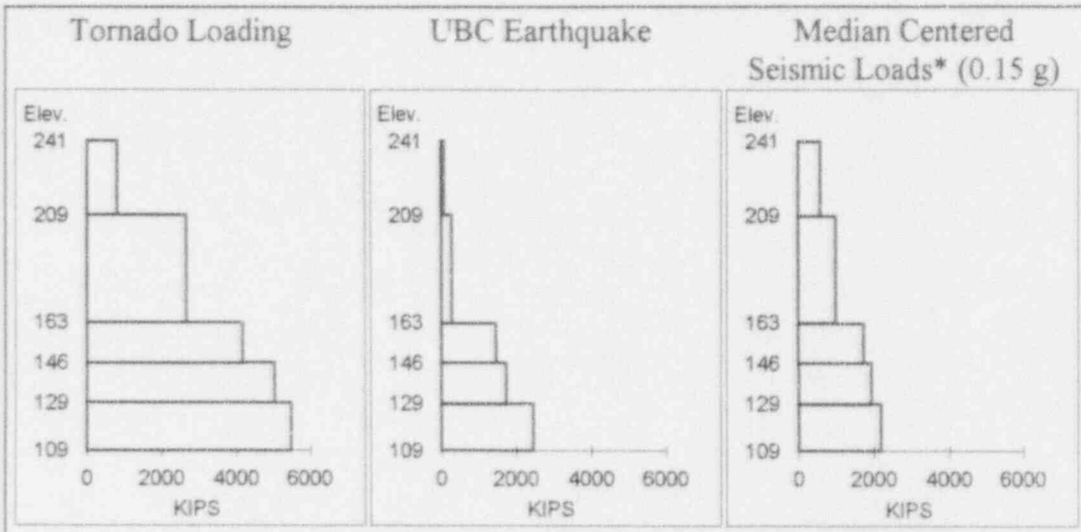


Figure 2: Comparison of Turbine Building Shear Loading
for Tornado and Seismic Loading Conditions
*See Section 1.0 for description of load estimation

Thus, the turbine building at Hatch Unit 2 is a seismically robust structure, and there is little risk of damage to the structure that would degrade the capability of the main steam and condenser fluid pathways. Specific parameters involved in the evaluation follow.

2.1 Design Basis

2.1.1 Lateral Force Resisting System Superstructure Type (above turbine floor) is a braced or rigid frame structure depending on the direction of lateral load consisting of the following:

a. Column lines TA and TI comprise vertical, alternating bays of cross-bracing resisting N-S wind or seismic lateral loading conditions.
b. E-W lateral forces are resisted by rigid frame bents from column line T15 to column line T22.
c. The end bays at column line T23 resist lateral loads by a braced frame.
d. The turbine floor and bottom cord bracing at elevation 225'-8" serve as diaphragms to distribute lateral loads to adjoining rigid frame bays in the E-W direction, or the braced bays in the N-S direction.

2.1.2 Lateral Force Resisting System Substructure Type (below turbine floor)

a. This reinforced concrete structure, elevation 112'-0" to 164'-0", consists of pilasters below rigid frame bents (above the turbine floor) carrying the loads to the foundation structure.
b. Concrete walls serve as shear walls for lateral loads in the N-S direction.

2.1.3 Seismic Design Codes

All category II structures are designed to conform to:

a. Paragraph 2314 of the 1970 Edition of the Uniform Building Code (UBC) (HNP-2-FSAR-3.2.1.)
b. American Institute of Steel Construction (AISC) Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, Seventh Edition.
c. American Concrete Institute (ACI) Building Code Requirements for Reinforced Concrete (ACI 318-63).
d. American Welding Society (AWS) Standard Code for Arc and Gas Welding in Building Construction (AWS D1.0-69 or AWS D2.0-69 as applicable).
e. Southern Standard Building Code

2.1.4 Seismic Design Basis

Note that wind loading was demonstrated to dominate design over the UBC requirements (see Sections 2.1.5 and 2.1.6).

a. UBC requirements for seismic zone I.
b. Comparison of lateral force magnitudes showed tornado loads governed over earthquake loads.
c. See section 2.1.6c for load combinations.

2.1.5 Wind Design Codes

a. UBC 1970 edition, Section 2308

b. American Society of Civil Engineers, paper no. 3269, Wind Design Requirements.
c. Items b through e of section 2.1.3 above.

2.1.6 Wind Design Basis

a. Wind Loads

The dynamic pressures used in the design of this plant are derived from ASCE Publication #3269 as it applies to the Hatch Nuclear Plant.

$q = 0.002558V^2$

where q is the velocity pressure in psf, and V is the wind velocity (mph). It was assumed that 80% of q is acting as pressure on windward side and 50% as suction on leeward side.

For roof slopes of less than 20°, suction on the roof was assumed as 60% of q. The total wind pressure p in psf is:

$p = 1.3\ q = 0.0033V^2$

<u>Wind Loads:</u>

	Class II Structures: 50 Year Recurrence				
Height (ft)	Velocity (mph) v	(psf) q	Pressure (psf) .8q	Suction (psf) .5q	Total (psf) p
0-50	100	26	21	13	34
50-100	125	40	32	20	52
150-400	156	62	50	31	81

(1) Siding and girts were designed for the wind pressures shown in the previous table. One third increase in allowable stresses was permitted. Siding and girts were designed to remain in place with tornado loading.
(2) Roof deck was designed for DL plus applicable LL and was checked for pressures acting upward normal to the surface equal to three-fourths of the values of q set forth in the table in paragraph 2.1.6a combined with DL only. For wind load and internal pressure, one third increase in allowable stress was permitted.

b. Tornado effects included in design consideration

(1) The velocity components applied as dynamic pressures normal to the walls of the building correspond to a 300 mph wind. ASCE paper 3269 was used to determine the proper drag and slope coefficients to be used for design. Maximum translational velocity of 60 mph was considered only for the purpose of computing depressurization.

(2) The tornado induces differential pressures between the inside and outside non-venting compartments, reaching a maximum bursting pressure of 3 psi (432 psf). For those compartments that are vented, either Hoecker's pressure profile for the Dallas Tornado of 1954, extrapolated to a 60 mph maximum translational velocity, or a simplified, idealized version of it where a constant pressure-drop rate of 1 psi/sec was assumed, was used.

(3) A missile impingement at any height on the structure equivalent to a 4" x 12" x 12' long wood plank (108 pounds) traveling end-on at 300 mph or a 4000 pound passenger auto flying through the air at 50 mph, at not more than 25 feet above the ground with a contact area of 20 sq. feet, was used.

(4) A torsional moment resulting from applying the wind specified in 2.1.6a above on one-half the structure and wind velocity equal to one-half that specified in 2.1.6a above applied to the other half of the building in the opposite direction was used.

(5) Blow-out panels designed to open at 50 psf internal differential pressure were considered in building design. The air flow and rate of depressurization depend upon the location and size of these panels. Exterior precast concrete panels were used either by themselves or in conjunction with poured-in-place concrete. These structural elements and their supports were designed to withstand the combined tornado loads discussed in 2.1.6b(1), (2), and (3) above without exceeding permissible stresses of 90% f_y for reinforcing steel and 85% f'_c of the concrete and 100% f_y for structural steel. Structural elements were allowed to dissipate internally stored energy through their strain energy absorption capacity, provided a ductility ratio of 20 was not exceeded and that the excess deflection would not impair that stability of the structure as a whole or in part.

c. Load Combinations

D = Dead Loads
L = Live Loads
W = Wind Loads
W'= Tornado Loads
E = Operating Basis Earthquake
E'= Design Basis Earthquake

Load Combinations	Allowable Stress
D + L + E	WSD - normal stress
D + L + W	WSD - 1/2 increase in stress
1.0D + 1.0L + 1.0W'	USD - See 2.1.6b(5)
1.0D + 1.0L + 1.0E'	USD

Live loads were allowed to be omitted, if the omission produced a more severe condition. Any other applicable loads were included in the load combinations.

Φ Factors

Φ = 0.90 for concrete in flexure
Φ = 0.85 for diagonal tension, bond and anchorage
Φ = 0.75 for spirally reinforced compression members
Φ = 0.70 for tied compression members

Reinforced concrete members were proportioned and reinforced so as to have a ductility factor of 4 or more. Ductility factor is defined as the ratio between the maximum displacement and the yield displacement of the simple system.

3.0 MAIN TURBINE CONDENSER

The H tch Unit 2 condenser is a two-shell, single pass unit, w th heat transfer surface a ea of 280,000 ft per shell. In Table 4-3 of Appendix D of NEDC-31858P, Rev. 1, the design attributes of this condenser are compared to the two condensers in the earthquake experience database most representative of the BWR type condensers: Moss Landing Units 6 & 7, and Ormond Beach Units 1 & 2. Note that the Hatch condenser is composed of two structurally independent shells, which may be independently compared to the earthquake experience condensers.

The condenser shells of the Hatch and the database condensers are 0.75" thick ASTM A-285C. The overall size and weight of the Hatch condenser are generally enveloped by the database condensers, as shown in Figure 3. The overall dimensions of the Hatch condenser are well represented by the earthquake experience data as well (Figure 4).

In summary, the Hatch Unit 2 condenser design and anchorage are similar to those at facilities in the earthquake experience database that have experienced earthquakes in excess of the Hatch Unit 2 design basis earthquake (see Figure 5). Appendix D, Section 4.1, of NEDC-31858P, Rev. 1, contains details of the earthquake experience for condensers. Specific data used in the evaluation are as follows.

3.1 Design Basis

3.1.1 Design Code:

Heat Exchange Institute (HEI) Standards.

3.1.2 Hydrostatic Test Requirement:

Shell - Completely filled with water and inspected for leakage.

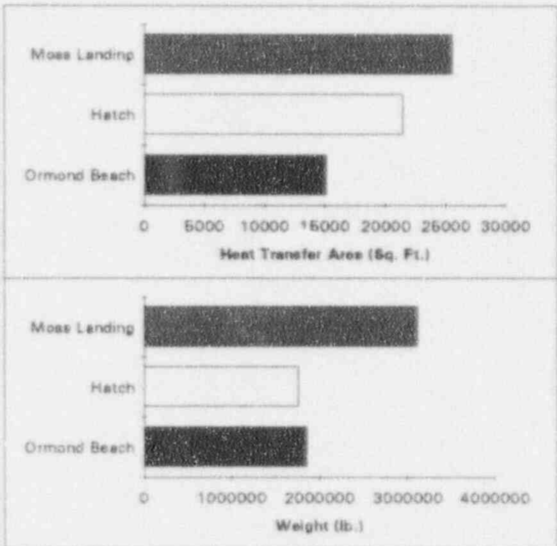
Waterboxes - Design pressure was 55 psig, test pressure was 60 psig per HEI. Latest edition of HEI, however, requires hydrostatic test pressure to be 1.5 times design pressure.

3.1.3 Anchorage

(a) Vertical-Uplift Capacity

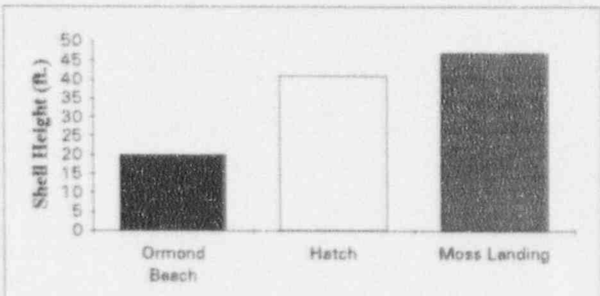
Material changes of the tubes and tubesheets in the Unit 2 main condensers produced a net weight reduction of 425K and caused the condensers to be in an uplift condition during operation. New hold-down bolts were installed to handle these new uplift forces.

Figure 3: Size Comparison of the Hatch Unit 2 Condenser
with Representative Condensers from Earthquake Experience Database

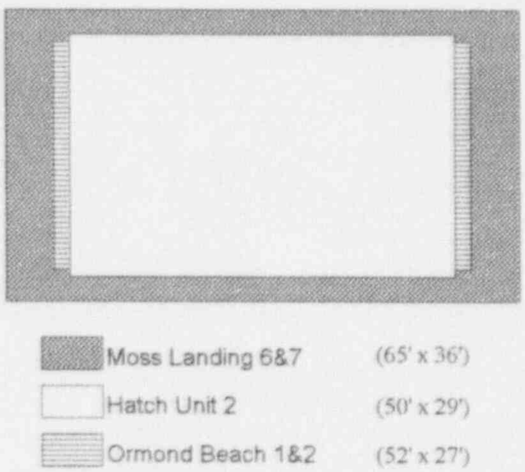


(b) Horizontal-types and shear areas

Figure 4: Dimensional Comparison of Hatch Unit 2 Condenser
and Representative Condensers from the Earthquake Experience Database



(a) Height Comparison



(b) Shell Footprint Comparison

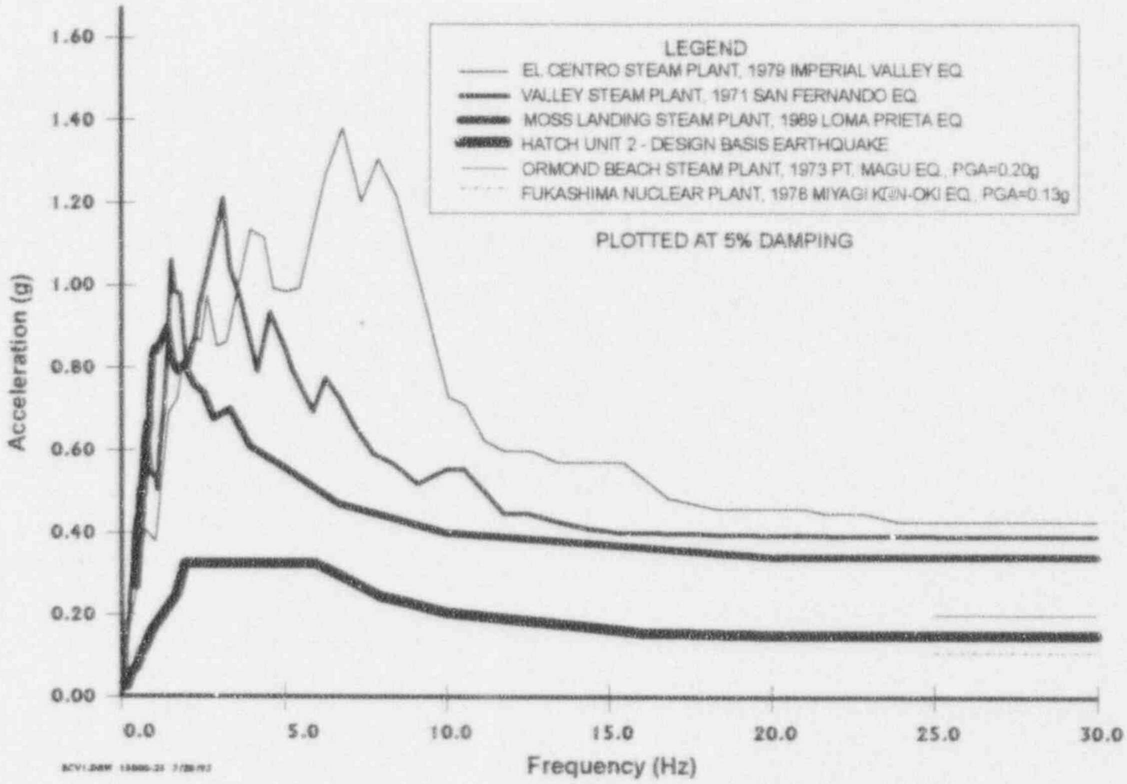


Figure 5: Comparison of Hatch Ground Response Spectrum to Data Base Spectra

Additional uplift loads due to seismic considerations have been evaluated. Realistic median-center estimates of the in-structure response spectrum peak were used in the evaluation. The existing condenser anchorage system has the capacity to withstand the uplift forces during a seismic event.

Each condenser footing has 4 - 2 1/4" ASTM A36 anchors that project through a 2" thick sole plate. The sole plate has 1 1/2" thick x 50" long shear plates that extend 4" down into a 4' high reinforced concrete pier. The concrete pier is anchored into the turbine building base slab. Shear forces are transferred from the anchor bolts into the sole plate and are carried through the reinforced pier down into the base slab. Figure 6 shows the location of the anchors and the orientation of the slots for the anchors at the piers.

The piers were originally designed to carry a horizontal force of 156K, which occurs during initial start up. There are no lateral loads present when the condenser is operational. There is thermal growth of the condenser from a fixed point at the base; however the condenser footings are designed with slotted holes so that these forces are not transmitted into the piers (Figure 6).

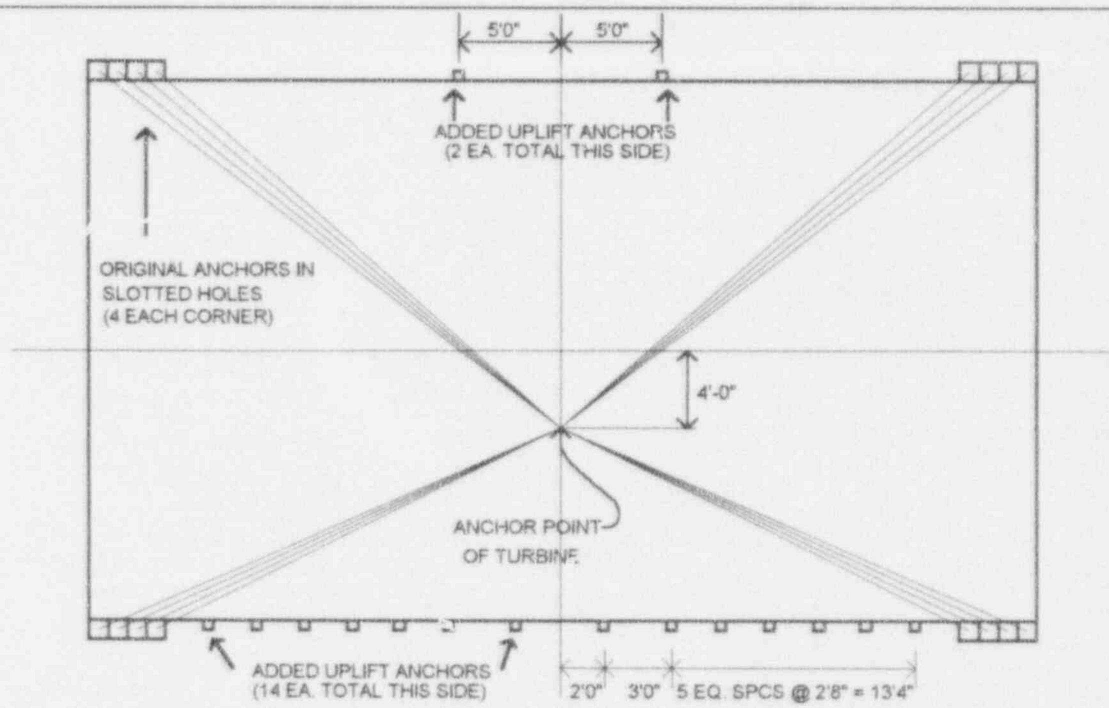


Figure 6a: Layout of Anchors

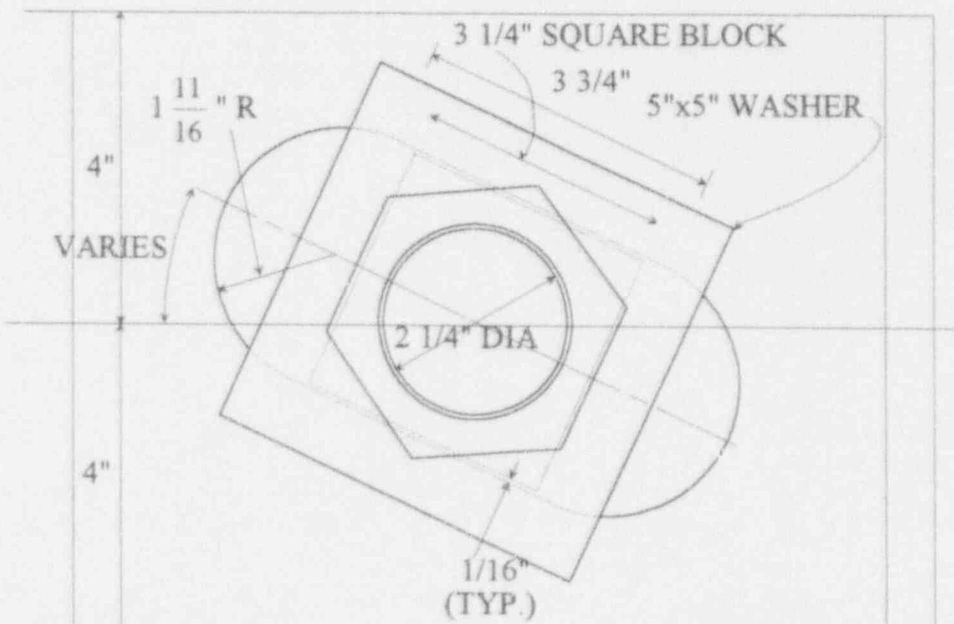


Figure 6b: Orientation of Slot for the 16 Original Anchors

An evaluation of the condenser support system was performed. It was determined that the horizontal shear capacity is sufficient to withstand the lateral forces that are present during an SSE event, based on estimated median centered earthquake demand. Figure 5 shows the anchor layout.

The shear area divided by the demand was used to compare anchorage in the earthquake experience data base (see Figure 7). The values for Hatch Unit 2 condenser are: lower bound, 0.000156 in^2 per g; upper bound, 0.000234 in^2 per g, higher than for most BWR condensers, and significantly higher than the selected data base sites (see NEDC 31858P, Rev. 1, Figure 4-10 and 4-11).

3.1.4 Manufacturer: Foster Wheeler Energy Corporation.

3.1.5 Size, Weight, Dimensions:

Size - 560,000 ft^2 total, 280,000 ft^2 per shell.
Weight - (Based on new titanium tubes and tubesheets)
Dry Weight Per Shell 775,000 lbs.
Operating Weight Per Shell 1,360,000 lbs.
Dimensions - Per Shell
Length: tubesheet to tubesheet 50'-0"; overall incl. waterboxes 66'-7"
Width: 29'-0"
Height: 23'-9"
Height: incl. neck and exp. joint, 41'-2"; base slab to turb. conn. 45'-6"

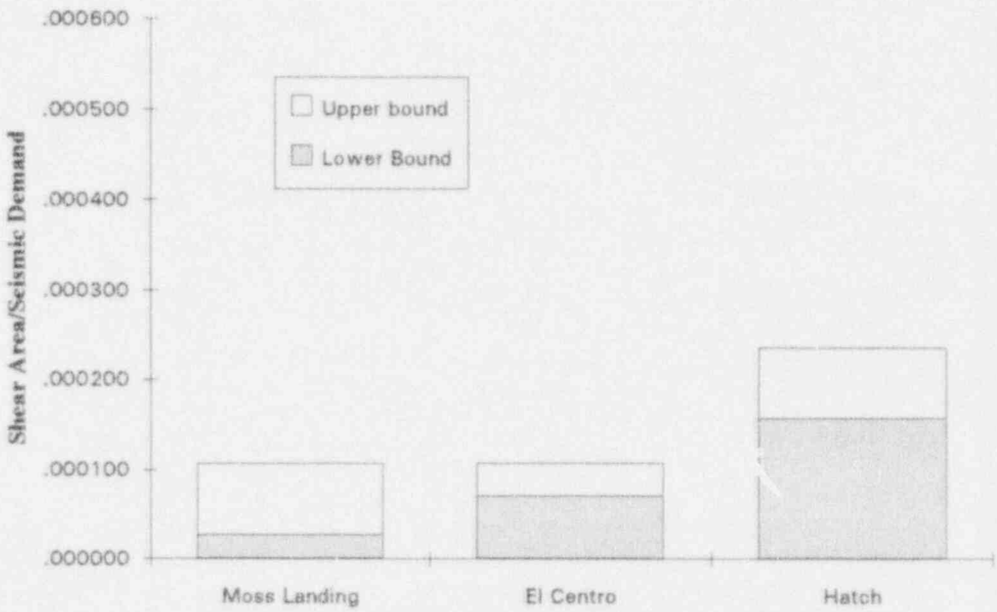


Figure 7a: Anchorage Capacity-to-Demand Ratio: Comparison of Hatch Unit 2 Condenser to Selected Data Base Sites: Parallel to Turbine Generator Axis
(data from NEDC 31858P, Rev. 1, Appendix D, Figure 4-10 and 4-11)

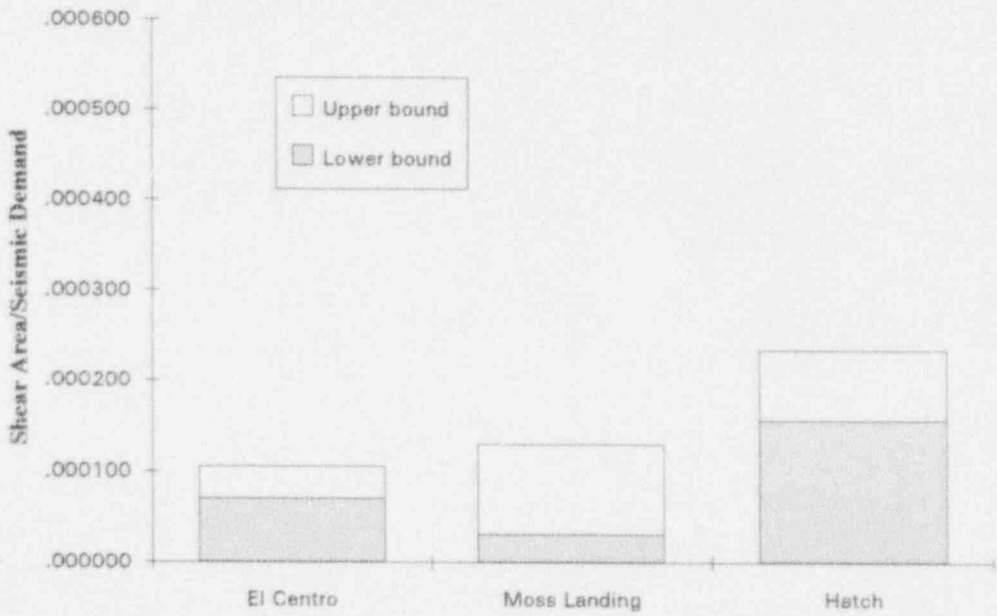


Figure 7b: Anchorage Capacity-to-Demand Ratio: Comparison of Hatch Unit 2 Condenser to Selected Data Base Sites: Perpendicular to Turbine Generator Axis
(data from NEDC 31858P, Rev. 1, Appendix D, Figure 4-10 and 4-11)

Total footprint for both shells - Length 66'-7"
Width 65'-8½"

3.1.6 Type: Base supported, rectangular, twin shell, single pass.

3.1.7 Shell Material and Thickness:

Material - ASTM A-285 Gr. C, Flange Quality
Thickness - 3/4"

3.1.8 Tube/Sheet Design:
Material - Titanium
Thickness - 1.125"
Tubes - 1" OD titanium, 22 BWG, 21,480 tubes per shell, 42,960 tubes total.
Support Plate Spacing - 40" - Spacing design was based on original Admiralty tubes. Stakes were added when the condenser was re-tubed with titanium.

3.1.9 Hotwell Capacity: 3 minutes, 63,600 gallons total.

3.1.10 Expansion Joint Design: 12 inch, stainless steel, convolute type.

4.0 MAIN STEAM AND DRAIN LINE/BYPASS PIPING

Portions of main steam and drain line/bypass piping designs that have not been seismically analyzed were reviewed to demonstrate that piping and supports fall within the bounds of design characteristics found in selected conventional power plant steam piping. These conventional power plant steam piping designs demonstrated good seismic performance and were shown to be comparable to the steam piping design for Hatch Unit 2. This included (1) a review of design codes and standards used to insure adequate dead load support margin and ductile support behavior where subject to lateral loads, and (2) a walkdown to verify that small diameter piping and instrumentation is free of impact interactions from falling and proximity or differential motion hazards.

Analyzed lines included the Main Steam Line (from the MSIV to the turbine stop valve) and the main steam bypass (to the bypass valves), the drain line portion in the reactor building, and portions of various main steam branch connections to the seismic anchor downstream of the isolation valves for the branch. Design methods for these analyzed lines are consistent with seismic Category I qualification methods for Plant Hatch and design margins are expected to be adequate to assure good seismic performance.

For lines designed by rule or by approximate methods such as the drain path (in the turbine building) and interfacing piping, it was demonstrated that these systems are composed of welded steel pipe and standard support components, well represented in conventional plants in the earthquake database. Further, it was demonstrated that adequate design margins exist for typical or bounding support designs.

In summary, the piping for the main steam and bypass was seismically designed in accordance with ASME B&PV Code, Section III. Thus, although it has thinner walls than most piping of its size in the earthquake experience data base, its seismic capability is evident. The main drain and associated piping are similar to the piping found in commercial piping systems in the earthquake experience data base that have experienced earthquakes in excess of the Hatch Unit 2 design basis earthquake (see Figure 5). Minor interaction issues identified in the walkdown that could be potential sources of damage were evaluated, and, where necessary, action has been initiated to eliminate the potential (see Table 3). Specific data used in the evaluation is summarized below. For the main drain and interconnected piping, it was demonstrated that adequate design margins exist to provide reasonable assurance that piping position retention will be maintained by the system dead weight supports under normal and earthquake loading.

4.1 Main Steam and Turbine Bypass

These systems were analyzed in accordance with the ASME code, Section III, Class 2, using response spectrum analysis techniques. The analysis model included the main steam (to the turbine), the bypass line, and significant branch piping up to the seismic anchor. For the steam seal the moisture separator reheater, steam jet air ejector, and the reactor feed pump branches, the anchor is downstream of the isolation valve. Thus, detailed seismic design analysis was performed for these portions of the systems. Margin for the main steam and turbine bypass is basically the design margin inherent in the seismic design codes.

4.1.1 Design Basis

4.1.1.1 Piping Design Code: ASME III, Class 2, 1974 and B31.1, 1973

4.1.1.2 Piping Design:

a. Design Temperature: 545°F
   Design Pressure: 1250 psi
b. Size, schedule, and D/t

Pipe Size (NPS)	Thickness (inch)	D/t
24	1.531	16
24	1.218	20
16	1.031	16
10.75	0.718	15
10.75	0.593	18
4.5	0.437	10
4.5	0.337	13
6.625	0.562	12
6.625	0.432	15
14	0.937	15

c. Typical Support Spacing: B31.1 suggested span
d. Support Types: Springs, struts, snubbers, box types, etc.
e. Design Loading: Weight, thermal expansion, seismic, steam hammer
f. Analysis Method: Linear elastic analysis, seismic spectrum analysis, steam hammer time history
g. Seismic and Dynamic Design Basis - Response spectrum analysis using floor response spectra based on the design basis earthquake (DBE) from the FSAR (0.15 g maximum ground motion - see Figure 5 for comparison to experience data base ground motion)

4.1.1.3 Pipe Support Design Code: AISC, ANSI B31.1

4.1.1.4 Margin Assessment:

Design methods for these analyzed lines are consistent with seismic Category I qualification methods for Plant Hatch and design margins are expected to be adequate to assure good seismic performance.

4.1.2 Main Steam and Turbine Bypass Supplemental Verification Walkdown Results

See Table 2.

4.2 Main Steam Drain to Condenser

The main steam drain to the condenser is of welded pipe, and was analyzed by rule and approximate methods. The main drain and associated piping are similar to the piping found in commercial piping systems in the earthquake experience data base that have experienced earthquakes in excess of the Hatch Unit 2 design basis earthquake (see Figure 5). Minor interaction issues identified in the walkdown that could be potential sources of damage were evaluated, and, where necessary, action has been initiated to eliminate the potential (see Table 3). Specific data used in the evaluation is summarized below. For these lines, it was demonstrated that adequate design margins exist to provide reasonable assurance that piping position retention will be maintained by the system dead weight supports under normal and earthquake loading.

4.2.1 Design Basis

4.2.1.1 Piping Design Code: ASME III, Class 2, 1974 and ANSI B31.1

4.2.1.2 Piping Design:

a. Design Temperature and Pressure: 575°F and 1250 psi; 562°F and 1146 psi
b. Size, Schedule and D/t: NPS 3, schedule 160, D/t = 8, and NPS 1, schedule 160, D/t = 5

c. Typical Support Spacing: B31.1 suggested spans
d. Support Types: Rigid struts, rods
e. Design Loading: Weight, thermal expansion, hydro, seismic
f. Analysis Methods: Linear elastic analysis
g. Seismic and Dynamic Design Basis: Response spectrum analysis using the Design Basis Earthquake from the FSAR (inside Reactor Building); linear elastic analysis (Turbine Building - dead weight and thermal only)

4.2.1.3 Pipe Support Design Code: AISC and MSS SP58

4.2.1.4 Margin Assessment

This assessment is to demonstrate the Main Steam Drain Line design provides adequate margins when subject to weight and seismic load, thus providing reasonable assurance that the position retention of the line will be maintained during a seismic event. In conjunction with the field verification, this assessment has provided assurance that the supports will behave in a ductile manner and that the lines are free of known seismic hazards. Further, it demonstrates that the Hatch designs will perform in a manner similar to piping and supports that have observed good seismic performance in past strong ground motion earthquakes.

The methodology utilized to demonstrate the margins inherent in the piping support designs is the Conservative Deterministic Failure Margin (CDFM) Seismic Margins. This methodology is based on the following procedures for capacity and demand estimation:

- The earthquake response spectrum is conservatively defined as 84% non-exceedance
- The estimated structural and piping response is median centered
- The component support capacity is conservatively estimated

This combination of conservatively defined seismic demand, median centered response to the seismic demand, and conservative estimate of capacity is considered to result in a high confidence of a low probability of failure (HCLPF), which provides the desired reasonable assurance of performance.

4.2.1.4.1 Seismic Demand

Seismic demand is estimated based on the Hatch Unit 1 median centered margins earthquake response spectra developed for the control building, and on the similarity of the turbine building and control building structures. Scaling and frequency shifting was performed using the early fixed-base analyses response spectra for the two buildings and the median centered margins response spectra for the Unit 1 control building. The resulting estimated maximum vertical floor response spectrum peak is 0.75 g (5% damping).

4.2.1.4.2 Piping System Response Estimation

The system response estimation is a median centered best estimate of the appropriate loadings:

- Loadings Combination: Operating Mechanical Loads + Dead Weight + Seismic
- Component Standard Supports Designed by Load rating: TL x 0.7 $\frac{S_u}{S_u^*}$

where,

TL	=	Support test load $\leq$ load under which support fails to perform its intended function
S_u	=	Material ultimate strength at temperature
S_u*	=	Material ultimate strength at test temperature

Operating mechanical loads for this system are thermal expansion loads. Piping systems designed utilizing rod supports typically do not impose constraints on thermal expansion, and no thermal loads are identified in the support designs. Design dead weight support loads are consistent with tributary area weight procedures.

The seismic response of the line is median centered and utilizes a factored load coefficient methodology to determine seismic loads. The load coefficient utilized is a factor of one (1) times the peak vertical spectral response acceleration.

4.2.1.4.3 Pipe Support Component Capacities

The supplemental field verification determined that the support types used are considered to have good seismic performance. The system is predominantly supported utilizing dead load rods. These designs are constructed from standard support catalog items and typically consist of clamps, threaded rod, weldless eye nuts, turnbuckles, clevis and welded lug attachments to either concrete or to steel structures. These types of supports are designed to resist vertical load in tension. Design capacities are provided by manufacturers' ratings.

Load capacity ratings for component standard supports are typically based on test and utilize a factor of safety of 5 in accordance with MSSP-58. The load on which the load capacity data (LCD) is based is therefore a factor of five higher that the catalog load rating. The margins capacities for the component support items are taken as the LCD x 5 x 0.7.

Evaluation of bolted anchorages to concrete follows the procedures established in the Margins Methodology report EPRI NP-6041. Concrete anchor bolts are evaluated using data from the EPRI NP-5228.

4.2.1.4.4 Margins Evaluation Results and Conclusions

A support for the Main Drain to Condenser was selected based on the results of the supplemental field verification, which identified the support system as having less than standard concrete anchor capacity due to the proximity of adjacent anchors. Therefore, it may be considered to have less capacity than typical supports on this system. HCLPF values for the support and anchorage are significantly higher than the 0.15g plant design basis. On the basis of the evaluation of this support and the configuration of the other supports in the system, the HCLPF values for the supports in this system are typically in the range of 0.75g to 1.5g. On the basis of this evaluation, we conclude that the system design has adequate margin to insure position retention. Furthermore, based on the supplemental field walkdown inspection, the piping systems and their supports are similar to piping system and support designs that have experienced strong ground motion and demonstrated good seismic performance.

4.2.2 Main Steam Drain to Condenser Supplemental Verification Walkdown Results

See Table 3.

5.0 INTERCONNECTED SYSTEMS

The interconnected systems are composed of welded steel piping and standard support components, well represented in the earthquake experience database. These systems, analyzed by rule and approximate methods, are similar to the piping found in commercial piping systems in the earthquake experience data base that have experienced earthquakes in excess of the Hatch Unit 2 design basis earthquake (see Figure 5). Minor interaction issues identified in the walkdown that could be potential sources of damage were evaluated, and, where necessary, action has been initiated to eliminate the potential (see Table 3). Specific data used in the evaluation is summarized below. For these lines, it was demonstrated that adequate design margins exist to provide reasonable assurance that piping position retention will be maintained by the system dead weight supports under normal and earthquake loading.

5.1 Design Basis

Table 2 shows the design parameters for the interconnected piping associated with the main steam, main steam bypass, main drain, and condenser.

5.1.1 Margin Assessment for Interconnected Systems

Same as for Main Steam Drain to Condenser, Section 4.2.1.4.

5.1.2 Supplemental Verification Walkdown Results for Interconnected Systems

See Table 3.

Table 2: Interconnected System Design Parameters

System Designation	Piping Design	Temp. (^oF)	Pres. (psig).	Size Size	Sch	D/t	Supports Spacing	Types	Des. Code	Loading	Analysis Method	Seismic Des. Basis	
												To Anchor	Remainder
Main Steam to Steam Jet Air Ejectors	ANSI B31.1	562	1146	6"	80	15	ANSI B31.1	Rod hangers, concrete anchors, bolted connections	AISC, MSS SP58	DW, Thermal, Hydro	Linear Elastic	RS anal. using DBE	None
Main Steam to Reactor Feed Pump Turbine	ANSI B31.1	562	1146	4"	80	13	ANSI B31.1	Rigid struts, snubbers	AISC, MSS SP58	DW, Thermal, Hydro	Linear Elastic	RS anal. using DBE	None
Steam Jet Air Ejectors to Off-Gas Preheaters	ANSI B31.1	575	1425	3"	160	8	ANSI B31.1	Rod hangers, concrete anchors, bolted connections	AISC, MSS SP58	DW, Thermal, Hydro	Linear Elastic	None	None
Main Steam to Moisture Seperator Reheaters	ANSI B31.1	575 & 562	1250 & 1146	8" & 10"	80	17, 18	ANSI B31.1	Springs, struts, snubber, box type	AISC, MSS SP58	DW, Thermal, Hydro, Seismic	Linear Elastic, RS Analysis, steam hammer time history	RS anal. using DBE	NA
Instrument Header to Pressure Transducers	ANSI B31.1	575	1250	1" & 4"	160 & 120	5 & 10	ANSI B31.1	Mostly rod hangers	AISC, MSS SP58	DW, Thermal, Hydro	Linear Elastic	None	None
Main Steam to Sample System	ANSI B31.1	583	1387	1/2" tubing	.065 wall	8	ANSI B31.1	standard tubing clips	AISC, MSS SP58	DW, Thermal, Hydro	Linear Elastic	None	None
Miscellaneous Drains	ANSI B31.1	562	1162	1", 2", 3"	160	5, 7, 8	ANSI B31.1	Rods, U-bolts, boxed supports, etc.	AISC, MSS SP58	DW, Thermal, Hydro	Linear Elastic	None	None

Table 3: Outlier Identification and Resolution

System Description	Outlier Description	Outlier Type (Potential Failure Mode)					Resolution Status	Required Action
		A	F	P	D	V		
Condenser	Review for anchorage, experience bounding	X					Condenser found to be well represented in database. Anchorage adequate.	
	Anchor bolt nut missing; others loose	X					Not acceptable as-is	Replace nut; tighten nuts
	12" HP Drain manifold w/ 8 connections is cantilevered about 12 ft. from condenser		X				Acceptable as-is by analysis	
	Potentially damaging interaction of drain line on end of 12" manifold and pipe			X			Not acceptable as-is	Relocate drain line
Main Steam By-Pass	Potential interaction w/ hydraulic lines (check consequences)			X			Acceptable as-is; loss of hyd. results in closure of bypass valve	
	Check the valve function for closure on loss of actuator power		X				Acceptable as-is: valve closes on loss of actuator power	
Main steam drain to condenser in the TB	Anchor bolt spacing violation	X					Acceptable as-is by analysis	
	Penetration of Turbine Bldg. wall is in a structural joint				X		Acceptable as-is: drawings indicate no potential interaction	
Main Steam and MS Drain Inside RB	Valve 2B21-F021 has extended MOV beyond screening guidelines					X	Acceptable as-is by analysis	
	Valve 2B21-F021 is close to 3" tube steel column			X			Acceptable as-is by analysis	
Main Steam to SJAE	Spring support at elbow	X					Not acceptable as-is	Redesign to include lateral restraint
	Box support appears to have anchors pulled out	X					Not acceptable as-is	Check bolts and take appropriate action
	Potentially damaging interaction between drain line and cable tray				X		Not acceptable as-is	Reroute drain piping
	Masonry wall supports tubing to instrument rack in Rooms A&B		X				Acceptable as-is by analysis: masonry wall evaluated	
	Potentially damaging interaction between boundary valve actuator diaphragms in SJAE Room A and a 3" line & flange			X			Acceptable as-is by analysis	

Table 3: Outlier Identification and Resolution (continued)

System Description	Outlier Description	Outlier Type (Potential Failure Mode)					Resolution Status	Required Action
		A	F	P	D	V		
Main Steam to SJAE (continued)	SJAE Anchorage evaluation required	X					Acceptable as-is by analysis	
	Masonry wall near entrance to this room.						Acceptable as-is by analysis: masonry wall evaluated	
	Missing anchor on SJAE	X					Not acceptable as-is	Install missing anchor
Main Steam RFPT Drain Pot to 12" HP Drain Manifold	Potentially damaging interaction with adjacent larger pipe			X			Acceptable as-is by analysis	
	Collapse of RFPT 72" discharge line (expansion joint, rod hung, mitered elbow) could impact lines (see Note 2)			X			Acceptable as-is based on analysis of RFPT discharge	
Main Steam Bypass Drain Pots to 12" HP Drain Manifold	Three one-way stanchions	X					Not acceptable as-is	Modify stanchion to provide lateral restraint
	Stanchion with no contact	X					Not acceptable as-is	Shim to close 1" gap
	Valve 2N22-F394 has extended MOV beyond screening guidelines					X	Acceptable as-is by analysis	
2N22-F011B to 12" HP Drain Manifold	Valve has extended MOV beyond screening guidelines					X	Acceptable as-is by analysis	
	Three supports are ganged pickups (see Note 1)	X					Not acceptable as-is	Disconnect from condenser vent pipe and re-hang
2N22-F020A to 12" HP Drain Manifold	Valve has extended MOV beyond screening guidelines					X	Acceptable as-is by analysis	
	One rod hanger broken or disconnected	X					Not acceptable as-is	Repair support
2N22-F020B to 12" HP Drain Manifold	Valve has extended MOV beyond screening guidelines					X	Acceptable as-is by analysis	
2N22-F011A to 12" HP Drain Manifold	Valve has extended MOV beyond screening guidelines					X	Acceptable as-is by analysis	
Stop Valve Before-Seat Drains to 12" HP Drain Manifold	Valve has extended MOV beyond screening guidelines					X	Acceptable as-is by analysis	
	Potentially damaging interaction with 36" duct			X			Acceptable as-is by analysis	

Table 3: Outlier Identification and Resolution (continued)

System Description	Outlier Description	Outlier Type (Potential Failure Mode)					Resolution Status	Required Action
		A	F	P	D	V		
Control Valve Before-Seat Drains to 12" HP Drain Manifold	Angle seat support yielded	X					Acceptable as-is by analysis	
2" line from SJAE to Off-gas Preheaters 2A & 2B	Masonry wall near entrance to this room.			X			Acceptable as-is by analysis: masonry wall evaluated	
	The preheater and recombiner are spring-supported	X					Acceptable as-is	
	Potential for damaging interaction between vessels and pipe			X			Acceptable as-is by analysis	
Main Steam from TB wall to Stop Valves	Potentially damaging interaction between 1" MS Sample tap off MS and MS By-Pass line			X			Acceptable as-is by analysis	
	Orthogonal supports to MS connected to Control & Turbine Buildings				X		Acceptable as-is by analysis	
	Single support connected to Turbine Bldg. and Control Bldg.				X		Not acceptable as-is	Analysis determined brace to TB column should be removed
Main Steam Sample line	Two tube clamps missing	X					Not acceptable as-is	Reinstall tube clamps
	Sample tube rack attached to masonry wall			X			Acceptable as-is by analysis: masonry wall evaluated	
	Heat exchanger in Sample Panel requires anchorage evaluation	X					Acceptable as-is by analysis	
	Anchorage of panel to floor visibly corroded and deteriorated	X					Not acceptable as-is	Repair as necessary
	Panel 2P33-P205 in proximity to masonry wall			X			Acceptable as-is by analysis: masonry wall evaluated	

Note 1: This support involved piping from several systems; only one entry included in the table.

Note 2: Collapse of the RFPT discharge could damage piping in several systems; only one entry is included in the table.

Key to Outlier Types in Table 3:

A Anchorage or Support Capacity
F Failure and Falling
P Proximity and Impact
D Differential Displacement
V Valve Operator Screening

Enclosure 2

Edwin I. Hatch Nuclear Plant - Unit 2
Request to Revise Technical Specifications:
Increase in Allowable MSIV Leakage Rate and
Deletion of the MSIV Leakage Control System

10 CFR 50.92 Evaluation

Proposed Change 1

This proposed change increases the allowable leak rate specified in Technical Specification 3.6.1.2.c and associated Action from 11.5 standard cubic feet per hour (scfh) for any one main steam isolation valve (MSIV) when tested at 28.8 psig to 100 scfh for any one MSIV when tested at 28.8 psig. Any MSIV that exceeds 100 scfh will be restored to 11.5 scfh.

Basis for Proposed Change 1

Georgia Power Company (GPC) has reviewed the proposed change and determined it does not involve a significant hazards consideration based on the following:

1. The change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed amendment does not involve a change to structures, components, or systems which would affect the probability of an accident previously evaluated in the Hatch Unit 2 Final Safety Analysis Report (FSAR). It results in acceptable radiological consequences for the design basis loss of coolant accident (LOCA) which was previously evaluated in section 15.1.39 of the FSAR. The MSIV leakage contribution to control room and offsite doses are bounded by the LOCA as described in section 15.1.39 of the FSAR. Therefore, the proposed amendment will not significantly increase the consequences of other analyzed accidents.

   Plant specific radiological analyses have been performed to assess the effects of the proposed increase in the allowable MSIV leak rate in terms of control room, technical support center (TSC), and offsite doses following a postulated design basis LOCA. These analyses utilize the hold-up volumes of the main steam piping and condenser as an alternate method for treating MSIV leakage. The radiological analyses use standard conservative assumptions for the release of source terms consistent with Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," Revision 2, dated April 1974.

   The analysis demonstrates dose contributions from the proposed MSIV leakage rate limit of 100 scfh per MSIV (or 100 scfh per steam line) result in an acceptable increase to the LOCA doses previously evaluated against the regulatory limits for the offsite, control room, and TSC

doses as contained in 10 CFR 100 and 10 CFR 50, Appendix A (General Design Criterion 19). The LOCA doses previously evaluated (recalculated) are discussed in section 15.1.39 of the FSAR. The revised LOCA doses are the LOCA doses previously evaluated in the FSAR (recalculated doses to be incorporated into the FSAR) plus the MSIV leakage doses calculated assuming use of the alternate treatment method. Table 2 of Enclosure 1 shows the previously calculated doses and the newly calculated doses.

It is important to note the resulting doses are dominated by the organic iodine fractions which occur because of the conservative source term assumptions used in this analysis. For MSIV leakage of 100 scfh per MSIV, more than 80 percent of the offsite, control room, and TSC iodine doses are due to the organic iodine from Regulatory Guide 1.3 source term and organic iodine converted from the elemental iodine deposited in main steam piping systems. If the actual iodine composition from the fuel release (cesium iodine) is used in the calculations essentially all of this organic iodine dose would be eliminated.

The TSC doses due to MSIV leakage are especially conservative. It is not expected that there will be any radioactive releases to the TSC due to MSIV leakage during the first 30 minutes following a LOCA event since it would take considerable time for the MSIV leakage to travel through the main steam lines and main steam line drain system to the condenser, into the turbine building, and finally to the atmosphere and TSC. However, it was conservatively assumed the 30 day integrated dose of 4.38 rem due to MSIV leakage of 100 scfh could be received by personnel who start the filter system 30 minutes after the LOCA occurs and immediately enter the TSC. The dose calculations were made using control room occupancy factors specified in SRP 6.4.

2. The proposed change will not create the possibility of a new or different kind of accident from any accident previously analyzed. The BWROG evaluated MSIV leakage performance and concluded MSIV leakage rates up to 200 scfh will not inhibit the capability and isolation performance of the valves to isolate the primary containment. There is no new modification which could impact the MSIV operability. The LOCA has been analyzed using the main steam piping and condenser as a treatment method to process MSIV leakage at the proposed maximum rate of 100 scfh. Therefore, the proposed change will not create any new or different kind of accident from any accident previously analyzed in the FSAR.

3. Operation of Plant Hatch in accordance with the proposed change will not involve a significant reduction in the margin of safety. The allowable leak rate limit specified for the MSIVs is used to quantify a maximum amount of bypass leakage assumed in the LOCA radiological analysis. Results of the analysis are evaluated against the dose requirements contained in 10 CFR 100 for the offsite doses and 10 CFR 50, Appendix A (General Design Criterion 19) for the control room and TSC doses.

   The margins of safety are not significantly adversely affected because the dose levels remain well below the limits of 10 CFR 100 and General Design Criterion 19. Therefore, the proposed amendment does not involve a significant reduction of the margin of safety at Plant Hatch.

<u>Proposed Change 2</u>

This proposed change to delete Technical Specification 3/4.6.1.4 and Bases section 3/4.6.1.4 involves eliminating the MSIV leakage control system (LCS) requirements from the Technical Specifications.

<u>Basis for Proposed Change 2</u>

GPC has reviewed the proposed change and determined it does not involve a significant hazards consideration based on the following:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. As described in section 6.5 of the FSAR, the LCS is manually initiated about 20 minutes following a design basis LOCA. Since the LCS is operated only after an accident has occurred, this proposed amendment has no effect on the probability of an accident. The proposed change results in acceptable radiological consequences of the design basis LOCA previously evaluated in section 15.1.39 of the FSAR.

   Plant Hatch has an inherent MSIV leakage treatment capability. GPC proposes to use the main steam line drains and condenser as an alternate to the LCS. GPC will incorporate this alternate method in the Operating Procedures and Emergency Operating Procedures.

   Plant specific radiological analyses have been performed to assess the effects of MSIV leakage in terms of control room, TSC, and offsite doses following a postulated design basis LOCA. These analyses utilize the hold-up volumes of the main steam piping and condenser as an alternate

treatment method for the MSIV leakage. The analysis demonstrates the proposed change results in an acceptable increase in the radiological consequences of a LOCA previously evaluated in the FSAR. Since the MSIV leakage contribution to control room, TSC, and offsite doses is bounded by the LOCA as described in section 15.1.39 of the FSAR, the proposed change will not involve a significant increase in the consequences of an accident previously analyzed.

2. The proposed change does not create the possibility for a new or different kind of accident from any accident previously analyzed. The purpose of the CS is to reduce the untreated MSIV leakage when isolation of the primary coolant system and containment are required. Radiological dose contributions due to MSIV leakage are bounded by a LOCA. The LOCA has been analyzed using the main steam piping and condenser as a treatment method to process MSIV leakage at the proposed maximum rate of 100 scfh and determined to be within the regulatory requirements. Therefore, the proposed change does not create the possibility of a new or different kind of accident.

3. The proposed change to delete Technical Specifications 3/4.6.1.4, and Bases section 3/4.6.1.4 does not involve a significant reduction in the margin of safety. The intended function of the LCS for treatment of MSIV leakage will be performed by using the more effective alternate path via the main steam drain lines and condenser. This treatment method is effective for treatment of MSIV leakage over an expanded leakage range. Except for the requirement to assure certain valves are opened to establish a proper flow path from the MSIVs to the condenser, the proposed method is passive and does not require any logic controls or interlocks. On the other hand, the LCS consists of complicated logic controls and sensitive equipment which must be maintained at significant cost and radiation exposure. The radiological effects on the margin of safety are discussed above for Change 1. The safety significance of the LCS in terms of public risk was addressed in NUREG/CR-4330 which contains the evaluation for eliminating the LCS and disabling the systems currently installed at BWRs. The conclusion was that the increased public risk is less than 1 percent. Therefore, the proposed change does not involve a significant reduction in the margin of safety at Plant Hatch.

Proposed Change 3

This proposed change deletes LCS valves 2E32-F001B, 2E32-F001F, 2E32-F001K, and 2E32-F001P from Technical Specifications Table 3.6.3-1.

Basis for Proposed Change 3

Georgia Power Company has reviewed the proposed change and determined it does not involve a significant hazards consideration based on the following:

1. The proposed change does not involve an increase in the probability or consequences of an accident previously analyzed in the FSAR. The proposed change is consistent with the proposed deletion of the LCS. It has no effect on the consequences of an accident since the LCS lines will be disconnected, capped and welded, assuring the integrity of the primary containment is maintained. Welding and examination procedures will be in accordance with the ASME Section XI repair and replacement requirements for Plant Hatch.

2. The proposed change will not create the possibility of a new or different kind of accident from any accident previously analyzed. The LCS lines connected to the main steam lines will be permanently closed to assure the primary containment integrity, isolation, and leak testing capability are not compromised, therefore eliminating the possibility for any new or different kind of accident.

3. The proposed change to delete the LCS isolation valves from Table 3.6.3-1 does not involve a significant reduction in the margin of safety, since permanent closure of the LCS lines assures the primary containment integrity, isolation, and leak testing capability are not compromised.

Proposed Change 4

This change involves the revision of the Index and the pages containing Technical Specifications 3/4.6.1.2.c (and associated Actions) and 3/4.6.1.4, and Bases section 3/4.6.1.4 to rearrange the sections and page numbers as appropriate. In addition, an editorial change unrelated to proposed changes 1 through 4 revises Index page XII to reflect that Bases section 3/4.6.3 is on page B 3/4 6-4b rather than page B 3/4 6-4.

Basis for Proposed Change 4

Georgia Power Company has reviewed the proposed change and determined it does not involve a significant hazards consideration based on the following:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident. The change is administrative in nature and has no effect on any accident. The change provides new section and page numbers and latest Technical Specifications amendment designations due to Changes 1, 2, and 3 above. An unrelated page number error is also corrected in the Index to reflect the proper page number for Technical Specification 3/4.6.3.

2. The proposed change does not create the possibility of a different kind of accident from any analyzed previously. The proposed change is administrative in nature and has no potential for creating an accident.

3. The proposed change will not involve a significant reduction in the margin of safety. The proposed change is administrative in nature and will have no bearing on the margin of nuclear safety at Plant Hatch.