GENERAL ELECTRIC COMPANY

AFFIDAVIT

I, R. Artigas, being duly sworn, depose and state as follows:

1. I am Manager, BWR Project Licensing, Safety and Licensing Operation, General Electric Company, and have been delegated the function of reviewing the information described in paragraph 2 which is sought to be withheld and have been authorized to apply for its withholding.

2. "Analog Trip System for Engineered Safeguard Sensor Trip Inputs - Edwin I. Hatch Nuclear Plant Units 1 and 2", NEDE-22154-1, Rev. 1, July 1983.

3. In designating material as proprietary, General Electric utilizes the definition of proprietary information and trade secrets set forth in the American Law Institute's Restatement Of Torts, Section 757. This definition provides:

   "A trade secret may consist of any formula, pattern, device or compilation of information which is used in one's business and which gives him an opportunity to obtain an advantage over competitors who do not know or use it.... A substantial element of secrecy must exist, so that, except by the use of improper means, there would be difficulty in acquiring information.... Some factors to be considered in determining whether given information is one's trade secret are: (1) the extent to which the information is known outside of his business; (2) the extent to which it is known by employees and others involved in his business; (3) the extent of measures taken by him to guard the secrecy of the information; (4) the value of the information to him and to his competitors; (5) the amount of effort or money expended by him in developing the information; (6) the ease or difficulty with which the information could be properly acquired or duplicated by others."

4. Some examples of categories of information which fit into the definition of proprietary information are:

   a. Information that discloses a process, method or apparatus where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

   b. Information consisting of supporting data and analyses, including test data, relative to a process, method or apparatus, the application of which provide a competitive economic advantage, e.g., by optimization or improved marketability;

c. Information which if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality or licensing of a similar product;

d. Information which reveals cost or price information, production capacities, budget levels or commercial strategies of General Electric, its customers or suppliers;

e. Information which reveals aspects of past, present or future General Electric customer-funded development plans and programs of potential commercial value to General Electric;

f. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection;

g. Information which General Electric must treat as proprietary according to agreements with other parties.

5. In addition to proprietary treatment given to material meeting the standards enumerated above, General Electric customarily maintains in confidence preliminary and draft material which has not been subject to complete proprietary, technical and editorial review. This practice is based on the fact that draft documents often do not appropriately reflect all aspects of a problem, may contain tentative conclusions and may contain errors that can be corrected during normal review and approval procedures. Also, until the final document is completed it may not be possible to make any definitive determination as to its proprietary nature. General Electric is not generally willing to release such a document to the general public in such a preliminary form. Such documents are, however, on occasion furnished to the NRC staff on a confidential basis because it is General Electric's belief that it is in the public interest for the staff to be promptly furnished with significant or potentially significant information. Furnishing the document on a confidential basis pending completion of General Electric's internal review permits early acquaintance of the staff with the information while protecting General Electric's potential proprietary position and permitting General Electric to insure the public documents are technically accurate and correct.

6. Initial approval of proprietary treatment of a document is made by the Subsection Manager of the originating component, the man most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within the Company is limited on a "need to know" basis and such documents at all times are clearly identified as proprietary.

7. The procedure for approval of external release of such a document is reviewed by the Section Manager, Project Manager, Principal Scientist or other equivalent authority, by the Section Manager of the cognizant Marketing function (or his delegate) and by the Legal Operation for

technical content, competitive effect and determination of the accuracy of the proprietary designation in accordance with the standards enumerated above. Disclosures outside General Electric are generally limited to regulatory bodies, customers and potential customers and their agents, suppliers and licensees only in accordance with appropriate regulatory provisions or proprietary agreements.

8. The document mentioned in paragraph 2 above has been evaluated in accordance with the above criteria and procedures and has been found to contain information which is proprietary and which is customarily held in confidence by General Electric.

9. The information contained herein is the result of extensive analyses performed at considerable cost to the General Electric Company. The development and verification of these methods, as well as their application and execution cost in excess of $2 million.

STATE OF CALIFORNIA)
COUNTY OF SANTA CLARA) ss:

R. Artigas, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 23th day of August, 1983.

R. Artigas
R. Artigas
General Electric Company

Subscribed and sworn before me this 23 day of August 1983.

OFFICIAL SEAL
KAREN S. VOGELHUBER
NOTARY PUBLIC - CALIFORNIA
SANTA CLARA COUNTY
My Commission Expires Dec. 21, 1984

Karen S. Vogelhuber
NOTARY PUBLIC, STATE OF CALIFORNIA

AJW:rm/A081919
8/19/83

TABLE OF CONTENTS

I. INTRODUCTION

The Edwin I. Hatch Nuclear Plant-Unit 2 (HNP-2) safety-related system instrumentation upgrade program, which includes incorporation of the analog transmitter trip system (ATTS), comprises many equipment modifications and installations to meet many of the requirements and criteria of:

- IE Bulletin 79-01B, Environmental Qualification of Class 1E Equipment*
- NUREG-0737, TMI Lessons Learned*
- NUREG-0696, Safety Parameters Display System Interfaces*

This submittal covers information on ATTS (Section III) and the proposed plant design modifications (Section IV) being performed at Plant Hatch through the installation of ATTS. The ATTS concept for use in the low low set system has been approved by the Nuclear Regulatory Commission (NRC) for Plant Hatch-Unit 2. This approval was granted by Amendment 33 to the Unit 2 Technical Specifications.

The purpose of this submittal is to provide the bases for the proposed modifications and to provide Georgia Power Company's (GPC) proposed revisions to the Technical Specifications. Because of space limitations on some figures located throughout this document, the Unit 2 designator was not included as part of the MPL identifier; that is, instead of 2G11-N004, G11-N004 was used. Therefore, unless otherwise noted, all MPL identifiers are Unit 2 specific.

ATTS and the associated plant modifications will be incorporated into the HNP-2 system logic during the next refueling outage. The installation is scheduled to be complete by the startup of Cycle 5.

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*Subject shown is not necessarily title of document.

II. REFERENCES

1. Edwin I. Hatch Nuclear Plant Unit 2, Docket No. 50-366, Proposed Plant Modification-Low Low Set Logic and Lowered MSIV Water Level, GPC Letter NED-83-108, Proposal for Technical Specifications Changes Which Support Cycle 4 Startup, February 23, 1983.

2. IE Bulletin 79-01B, Environmental Qualification of Class 1E Equipment.

3. IEEE 323-74, IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.

4. IEEE 344-75, IEEE Recommended Practices for Seismic Qualification of 1E Equipment for Nuclear Power Generating Stations.

5. NEDO-21617-A, General Electric Licensing Topical Report, Analog Transmitter/Trip Unit System for Engineered Safeguard Sensor Trip Inputs.

6. NUREG-0588, Revision 1, Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment.

7. NUREG-0696, Functional Criteria for Emergency Response Facilities.

8. Regulatory Guide 1.105, Instrument Setpoints.

9. Draft Standard Technical Specifications for General Electric Boiling Water Reactor (GE-STS) - BWR/4.

III. ANALOG TRANSMITTER TRIP SYSTEM (ATTS) INSTALLATION

A safety evaluation report was prepared to evaluate the licensing requirements of ATTS. From that evaluation, it was concluded that the incorporation of ATTS constitutes neither an unreviewed safety question, nor a significant hazards consideration. The following detailed description provides all of the instrumentation to be incorporated into ATTS during the next refueling outage. The general description of the proposed ATTS is provided in the NEDE-22154-1 General Electric (GE) Licensing Topical Report (Appendix 2). NEDE-22154-1 is a revision to NEDE-22154 which was provided to the NRC in Reference 1. The NEDE-221541 contains the system/component design changes and additional qualification test data. A description of qualification testing is documented in Chapter 4 of NEDE-22154-1, and in addition identifies Plant Hatch testing criteria. Actual qualification reports will be provided to the NRC upon request.

Since the equipment comprising the ATTS was demonstrated to be superior to the mechanical switches currently used at Plant Hatch, certain technical specifications surveillance requirements may be revised. These revisions take advantage of the sensor improvements and the decreased drift which was demonstrated for this new equipment. Section III discusses the proposed surveillance revisions for the complete ATTS installation. The proposed plant modifications associated with ATTS and their justifications are discussed in Section IV.

To keep the Nuclear Regulatory Commission (NRC) apprised of major Plant Hatch design modifications, Georgia Power Company (GPC) provided in Reference 1 a detailed description of the ATTS system to be installed. Since that time, several design modifications have occurred. This section, therefore, updates the information provided in Reference 1.

3A. ATTS DETAILED DESIGN INFORMATION

In NEDO-21617-A, dated December 1978, the NRC identified specific detailed information that each applicant (who uses that topical as a licensing basis) must provide relative to the specific application of the ATTS hardware into his plant. Since NEDE-22154-1 (Appendix 2) describes the same conceptual system as NEDO-21617-A but with different hardware, GPC is supplying the NRC with the same information (requested of those who use NEDO-21617-A as a licensing basis) regarding the Plant Hatch installation. Therefore, this section provides specific design criteria for the ATTS installation to be completed during refueling at the end of Cycle 4 for Plant Hatch-Unit 2.

3A.1 As identified in the NRC request section of NEDO-21617-A, GPC offers the information listed below.

3A.1.a The following information for each instrument loop that will be converted to the ATTS is provided in table 3.1:

- Variable name
- MPL number
- Engineered safeguards division
- Model number and vendor of the transmitter or RTD
- Device's associated rack.

3A.1.b The layout of each card file in each trip unit cabinet, showing the trip variable for each card file slot, is provided in figures 3.1 through 3.8. These figures were developed to illustrate the 12 card slots per card file and the three card files per cabinet arrangement.

3A.1.c The environmental service conditions for Plant Hatch are presented in tables 4-1 through 4-3 of NEDE-22154-1. The seismic curves identified in these tables are the Unit 2 floor response spectra curves generated for the various mounting locations of the ATTS hardware. Figures 3.9 through 3.21 correspond to or envelop the Plant Hatch safe shutdown earthquake curves. Table 4-5 and figures 4-5 through 4-12 of NEDE-22154-1 define the seismic levels to which the ATTS hardware was qualified. Comparison of the curves presented in figures 3.9 through 3.21 with the corresponding curves presented in figures 4-5 through 4-12 of NEDE-22154-1 shows that the ATTS hardware was seismically qualified to levels which exceed the Hatch requirements. Table 3.2 presents a summary of the various ATTS hardware along with the corresponding mounting locations and references to the applicable Plant Hatch seismic curves.

3A.1.d An interconnection diagram, showing the basic interconnections between the existing logic cabinets and the instrument cabinets and the new trip unit cabinets, is shown in figure 3.22. This interconnection diagram also presents the divisional separation of the cabinets.

3A.1.e The qualification program to the requirements of IEEE 323-74 and IEEE 344-75 is presently underway and is scheduled to be completed by the end of 1983. The details of the qualification program are presented in NEDE-22154-1. GPC will submit the final qualification reports (proprietary to GE) to the NRC upon request.

TABLE 3.1 (SHEET 1 OF 8)

INSTRUMENT LOOP INFORMATION

Variable Name[a]	Trip Unit MPL No.	Primary Sensor: Old MPL No.	Primary Sensor: New MPL No.	Engineering Safeguard Division	Existing Device: Manufacturer	Existing Device: Model No.	New Device: Generic Name	New Device: Manufacturer	New Device: Model No.[c]	Associated Rack[b]
1) Reactor Steam Dome Pressure High	B21-N620 A,B,C,D	NA	B21-N120 A,B,C,D	ECCS	NA	NA	Pressure Transmitter	Barton	763	A,C-H21-P404A/B B,D-H21-P405A/B
2) Low Low Set Control	B21-N621* A,B,C,D	NA	B21-N120 A,B,C,D	ECCS	NA	NA	Slave	NA	NA	NA
3) Low Low Set Control	B21-N622 A,B,C,D	NA	B21-N122 A,B,C,D	ECCS	NA	NA	Pressure Transmitter	Barton	763	A,C-H21-P404A/B B,D-H21-P405A/B
4) Steam Tunnel Temperature High	B21-N623 A,B,C,D	B21-N010 A,B,C,D	B21-N123 A,B,C,D	RPS	Fenwall	17002-40	RTD	Weed	1A0D	Local
5) Steam Tunnel Temperature High	B21-N624 A,B,C,D	B21-N011 A,B,C,D	B21-N124 A,B,C,D	RPS	Fenwall	17002-40	RTD	Weed	1A0D	Local
6) Steam Tunnel Temperature High	B21-N625 A,B,C,D	B21-N012 A,B,C,D	B21-N125 A,B,C,D	RPS	Fenwall	17002-40	RTD	Weed	1A0D	Local
7) Steam Tunnel Temperature High	B21-N626 A,B,C,D	B21-N013 A,B,C,D	B21-N126 A,B,C,D	RPS	Fenwall	17002-40	RTD	Weed	1A0D	Local
8) Reactor Vessel Pressure Low	B21-N641* B,C	B21-N021 B,C,	B21-N090 B,C	ECCS	Barton	288	Slave	NA	NA	NA
9) Reactor Vessel Steam Dome Pressure High	B21-N678 A,B,C,D	B21-N023 A,B,C,D	B21-N078 A,B,C,D	RPS	Barksdale	B2T-M12SS	Pressure Transmitter	Barton	763	A,B-H21-P404C/D C,D-H21-P405C/D

* Slave trip unit.

a. Transmitters having the same MPL No. are the same equipment providing several trip functions.

b. Separated for divisional considerations.

c. The complete RTD model number for all present ATTS applications is: 1A0D-611-1B-C-4-C-2-A2-0.

TABLE 3.1 (SHEET 2 OF 8)

INSTRUMENT LOOP INFORMATION

Variable Name[a]	Trip Unit MPL No.	Primary Sensor Old MPL No.	Primary Sensor New MPL No.	Engineering Safeguard Division	Existing Device Manufacturer	Existing Device Model No.	New Device Generic Name	New Device Manufacturer	New Device Model No.[c]	Associated Rack[b]
10) Reactor Vessel Steam Dome Pressure Low	B31-N679 A	B31-N018 A	B31-N079 A	RPS	Barksdale	B2T-M12SS	Differential Pressure Transmitter	Barton	764	H21-P404E
	B31-N679 D	B31-N018 B	B31-N079 D	RPS	Static-O-Ring	SN-A33-(X9)STT	Differential Pressure Transmitter	Barton	764	H21-P405E
11) Reactor Vessel Water Level Low (Level 3)	B21-N680 A,B,C,D	B21-N017 A,B,C,D	B21-N080 A,B,C,D	RPS	Barton	288A	Differential Pressure Transmitter	Barton	764	A,B-H21-P404C/D C,D-H21-P405C/D
12) Reactor Vessel Water Level Low Low Low (Level 1)	B21-N681 A,B	B21-N024** A,B	B21-N081 A,B	RPS	Yarway	4418C	Differential Pressure Transmitter	Barton	764	A,B-H21-P404C/D
	B21-N681 C,D	B21-N025** A,B	B21-N081 C,D	RPS	Yarway	4418C	Differential Pressure Transmitter	Barton	764	C,D-H21-P405C/D
13) Reactor Vessel Water Level Low Low (Level 2)	B21-N682* A,B	B21-N024** A,B	B21-N081 A,B	RPS	Yarway	4418C	Slave	NA	NA	NA
	B21-N682* C,D	B21-N025** A,B	B21-N081 C,D	RPS	Yarway	4418C	Slave	NA	NA	NA
14) Reactor Shroud Water Level Low (Level 0)	B21-N685 A,B	B21-N036 B21-N037	B21-N085 A,B	ECCS	Yarway	4418CE	Differential Pressure Transmitter	Barton	764	A-H21-P409 B-H21-P410
15) Main Steam Line A Flow High	B21-N686 A,B,C,D	B21-N006 A,B,C,D	B21-N086 A,B,C,D	RPS	Barton	288	Differential Pressure Transmitter	Barton	764	A,B-H21-P415A/B C,D-H21-P425A/B

* Slave trip unit.
** Not being deleted; only the safety function is being replaced by ATTS.

a. Transmitters having the same MPL No. are the same equipment providing several trip functions.

b. Separated for divisional considerations.

c. The complete RTD model number for all present ATTS applications is: IAOD-611-1B-C-4-C-2-A2-0.

TABLE 3.1 (SHEET 3 OF 8)

INSTRUMENT LOOP INFORMATION

Variable Name[a]	Trip Unit MPL No.	Primary Sensor Old MPL No.	Primary Sensor New MPL No.	Engineering Safeguard Division	Existing Device Manufacturer	Existing Device Model No.	New Device Generic Name	New Device Manufacturer	New Device Model No.[c]	Associated Rack[b]
16) Main Steam Line B Flow High	B21-N687 A,B,C,D	B21-N007 A,B,C,D	B21-N087 A,B,C,D	RPS	Barton	288	Differential Pressure Transmitter	Barton	764	A,B-H21-P415A/B C,D-H21-P425A/B
17) Main Steam Line C Flow High	B21-N688 A,B,C,D	B21-N008 A,B,C,D	B21-N088 A,B,C,D	RPS	Barton	288	Differential Pressure Transmitter	Barton	764	A,B-H21-P415A/B C,D-H21-P425A/B
18) Main Steam Line D Flow High	B21-N689 A,B,C,D	B21-N009 A,B,C,D	B21-N089 A,B,C,D	RPS	Barton	288	Differential Pressure Transmitter	Barton	764	A,B-H21-P415A/B C,D-H21-P425A/B
19) Reactor Vessel Pressure Low	B21-N690 A,D,E,F	B21-N021 A,D,E,F	B21-N090## A,D,E,F	ECCS	Barksdale	B2T-M12SS	Pressure Transmitter	Barton	763	A,E-H21-P404A D,F-H21-P405A
	B21-N690 B,C	B21-N021 B,C	B21-N090 B,C	ECCS	Barton	288	Pressure Transmitter	Barton	763	B-H21-P410 C-H21-P409
20) Reactor Vessel Water Level Low Low Low (Level 1)	B21-N691 A,B,C,D	B21-N031 A,B,C,D	B21-N091# A,B,C,D	ECCS	Yarway	4418C	Differential Pressure Transmitter	Barton	764	A,C-H21-P404A B,D-H21-P405A
21) Reactor Vessel Water Level Low Low (Level 2)	B21-N692* A,B,C,D	B21-N031 A,B,C,D	B21-N091 A,B,C,D	ECCS	Yarway	4418C	Slave	NA	NA	NA
22) Reactor Vessel Water Level High (Level 8)	B21-N693 A,B	B21-N017 A,B	B21-N093 A,B	ECCS	Barton	288A	Differential Pressure Transmitter	Barton	764	Local
	B21-N693* C,D	B21-N017 C,D	B21-N095 A,B	ECCS	Barton	288A	Slave	NA	NA	NA

* Slave trip unit.

\# B21-N091A,B will provide reactor water level indication and replace by B21-N026A,B. B21-N091C,D will provide reactor water level recorder inputs and replace B21-N026C,D.

\## B21-N090A,D will provide reactor pressure recorder inputs and replace B21-N051A,B.

a. Transmitters having the same MPL No. are the same equipment providing several trip functions.

b. Separated for divisional considerations.

c. The complete RTD model number for all present ATTS applications is: 1AOD-611-1B-C-4-C-2-A2-0.

TABLE 3.1 (SHEET 4 OF 8)

INSTRUMENT LOOP INFORMATION

Variable Name[a]	Trip Unit MPL No.	Primary Sensor Old MPL No.	Primary Sensor New MPL No.	Engineering Safeguard Division	Existing Device Manufacturer	Existing Device Model No.	New Device Generic Name	New Device Manufacturer	New Device Model No.[c]	Associated Rack[b]
23) Reactor Vessel Water Level Low (Level 3)	B21-N695 A,B	B21-N042 A,B	B21-N095 A,B	ECCS	Yarway	4418C	Differential Pressure Transmitter	Barton	764	A-H21-P404B B-H21-P405B
24) Drywell Pressure High	C71-N650 A,B,C,D	C71-N002 A,B,C,D	C71-N050 A,B,C,D	RPS	Barksdale	D2H-M18SS	Differential Pressure Transmitter	Barton	764	Local
25) RHR Pump Discharge Pressure High	E11-N655 A,B,C,D	E11-N016 A,B,C,D	E11-N055 A,B,C,D	ECCS	Barksdale	B2T-M12SS	Pressure Transmitter	Barton	763	A,C-H21-P418B B,D-H21-P421B
26) RHR Pump Discharge Pressure High	E11-N656 A,C	E11-N020 A,C	E11-N056 A,C	ECCS	Barksdale	B2T-M12SS	Pressure Transmitter	Barton	763	H21-P418B
	E11-N656 B,D	E11-N020 B,D	E11-N056 B,D	ECCS	Static-O-Ring	5N-AA3-(X10)SITT	Pressure Transmitter	Barton	763	H21-P421B
27) RHR Pump Flow Low	E11-N682 A,B	E11-N021 A,B	E11-N082 A,B	ECCS	Barton	289	Differential Pressure Transmitter	Barton	764	A-H21-P418A B-H21-P421A
28) Drywell Pressure High	E11-N694 A,B,C,D	E11-N010 A,B,C,D	E11-N094 A,B,C,D	ECCS	Barksdale	D2H-M18SS	Differential Pressure Transmitter	Barton	764	Local
		E11-N011# A,B,C,D								
29) Core Spray Pump Discharge Flow Low	E21-N651 A,B	E21-N006 A,B	E21-N051 A,B	ECCS	Barton	289	Differential Pressure Transmitter	Barton	764	A-H21-P401 B-H21-P419

Deleted from plant. Functions of E11-N011A,B,C,D assigned to E11-N010A,B,C,D.

a. Transmitters having the same MPL No. are the same equipment providing several trip functions.

b. Separated for divisional considerations.

c. The complete RTD model number for all present ATTS applications is: 1AOD-611-1B-C-4-C-2-A2-0.

TABLE 3.1 (SHEET 5 OF 8)

INSTRUMENT LOOP INFORMATION

Variable Name[a]	Trip Unit MPL No.	Primary Sensor Old MPL No.	Primary Sensor New MPL No.	Engineering Safeguard Division	Existing Device Manufacturer	Existing Device Model No.	New Device Generic Name	New Device Manufacturer	New Device Model No.[c]	Associated Rack[b]
30) Core Spray Pump Discharge Pressure High	E21-N652 A,B	E21-N009 A,B	E21-N052 A,B	ECCS	Barksdale	B2T-M12SS	Pressure Transmitter	Barton	763	A-H21-P401 B-H21-P419
31) Core Spray Pump Discharge Pressure High	E21-N655 A,B	E21-N008 A,B	E21-N055 A,B	ECCS	Barksdale	B2T-M12SS	Pressure Transmitter	Barton	763	A-H21-P401 B-H21-P419
32) HPCI Pump Pressure High	E41-N650	E41-N027	E41-N050	ECCS	Barksdale	B2T-M12SS	Pressure Transmitter	Barton	763	H21-P414B
33) HPCI Pump Discharge Flow High, Low	E41-N651	E41-N006	E41-N051	ECCS	Barton	289	Differential Pressure Transmitter	Barton	764	H21-P414A
34) HPCI Pump Suction Pressure Low	E41-N653	E41-N010	E41-N053	ECCS	Static-O-Ring	6N-AA21-(X9)VSTT	Differential Pressure Transmitter	Barton	764	H21-P414B
35) HPCI Turbine Exhaust Diaphragm Pressure High	E41-N655 A,B,C,D	E41-N012 A,B,C,D	E41-N055 A,B,C,D	ECCS	Barksdale	D2H-M150SS	Differential Pressure Transmitter	Barton	764	A,C-H21-P434 B,D-H21-P414A
36) HPCI Turbine Exhaust Pressure High	E41-N656 B,D	E41-N017 A,B,	E41-N056 B,D	ECCS	Barksdale	B2T-M12SS	Pressure Transmitter	Barton	763	H21-P414B
37) HPCI Steam Line Differential Pressure High (+)	E41-N657 A,B	E41-N004 E41-N005	E41-N057 A,B	ECCS	Barton	288	Differential Pressure Transmitter	Barton	764	A-H21-P016 B-H21-P036

a. Transmitters having the same MPL No. are the same equipment providing several trip functions.

b. Separated for divisional considerations.

c. The complete RTD model number for all present ATTS applications is: 1AOD-611-1B-C-4-C-2-A2-0.

TABLE 3.1 (SHEET 6 OF 8)

INSTRUMENT LOOP INFORMATION

Variable Name[a]	Trip Unit MPL No.	Primary Sensor Old MPL No.	Primary Sensor New MPL No.	Engineering Safeguard Division	Existing Device Manufacturer	Existing Device Model No.	New Device Generic Name	New Device Manufacturer	New Device Model No.[c]	Associated Rack[b]
38) HPCI Steam Supply Pressure Low	E41-N658 A,B,C,D	E41-N001 A,B,C,D	E41-N058 A,B,C,D	ECCS	Barksdale	B2T-M12SS	Pressure Transmitter	Barton	763	A,C-H21-P016 B,D-H21-P036
39) HPCI Steam Line Differential Pressure High (-)	E41-N660* A,B	E41-N004 E41-N005	E41-N057 A,B	ECCS	Barton	288	Slave	NA	NA	NA
40) HPCI Torus Water Level High	E41-N662 B,D	E41-N015 A,B	E41-N062 B,D	ECCS	Magnetrol	3.5-751-MPG	Capillary Differential Pressure Transmitter	Barton	764 with Model 352 Capillary Sensors	Local
41) HPCI Equipment Ambient Temperature High	E41-N670 A,B	E41-N030** A,B	E41-N070 A,B	ECCS	Pyco	N145C3224P1 Type T	RTD	Weed	1A0D	Local
42) HPCI Pipe Room Ambient Temperature High	E41-N671 A,B	E41-N046** A,B	E41-N071 A,B	ECCS	Pyco	N145C3224P1 Type T	RTD	Weed	1A0D	Local
43) RCIC Pump Discharge Pressure High	E51-N650	E51-N020	E51-N050	ECCS	Barksdale	B2T-M12SS	Pressure Transmitter	Barton	763	H21-P417B
44) RCIC Pump Flow High and Low	E51-N651	E51-N002	E51-N051	ECCS	Barton	289	Differential Pressure Transmitter	Barton	764	H21-P417A
45) RCIC Turbine Exhaust Pressure High	E51-N656 A,C	E51-N009 A,B	E51-N056 A,C	ECCS	Barksdale	D2H-M80SS	Pressure Transmitter	Barton	763	H21-P417B
46) RCIC Steam Line ΔP High (+)	E51-N657 A,B	E51-N017 E51-N018	E51-N057 A,B	ECCS	Barton	288	Differential Pressure Transmitter	Barton	764	A-H21-P435 B-H21-P038

* Slave trip unit.

** Not being deleted; only the safety function is being replaced by ATTS.

a. Transmitters having the same MPL No. are the same equipment providing several trip functions.

b. Separated for divisional considerations.

c. The complete RTD model number for all present ATTS applications is: 1A0D-611-1B-C-4-C-2-A2-0.

TABLE 3.1 (SHEET 7 OF 8)

INSTRUMENT LOOP INFORMATION

	Variable Name(a)	Trip Unit MPL No.	Primary Sensor Old MPL No.	Primary Sensor New MPL No.	Engineering Safeguard Division	Existing Device Manufacturer	Existing Device Model No.	New Device Generic Name	New Device Manufacturer	New Device Model No.(c)	Associated Rack(b)
47)	RCIC Steam Supply Pressure Low	E51-N658 A,B,C,D	E51-N019 A,B,C,D	E51-N058 A,B,C,D	ECCS	Barksdale	B2T-M12SS	Pressure Transmitter	Barton	763	A,C-H21-P035 B,D-H21-P038
48)	RCIC Steam Line ΔP High (-)	E51-N660* A,B	E51-N017 E51-N018	E51-N057 A,B	ECCS	Barton	288	Slave	NA	NA	NA
49)	RCIC Equipment Ambient Temperature High	E51-N661 A,B	E51-N023** A,B	E51-N061 A,B	ECCS	Pyco	N145C3224P1 Type T	RTD	Weed	1A0D	Local
50)	Torus Ambient Temperature (no trip)	E51-N663 A,B,C,D	E51-N026** A,B,C,D	E51-N063 A,B,C,D	ECCS	Pyco	N145C3224P1 Type T	RTD	Weed	1A0D	Local
		E51-N664 A,B,C,D	E51-N027** A,B,C,D	E51-N064 A,B,C,D	ECCS	Pyco	N145C3224P1 Type T	RTD	Weed	1A0D	Local
51)	Torus Ambient Temperature High	E51-N666 A,B,C,D	E51-N025** A,B,C,D	E51-N066 A,B,C,D	ECCS	Pyco	N145C3224P1 Type T	RTD	Weed	1A0D	Local
52)	Torus Differential Temperature High	E51-N665 A,B,C,D	E51-N604** A,B,C,D	NA	ECCS	NA	NA	NA	NA	NA	NA
53)	RCIC Pump Suction Pressure Low	E51-N683	E51-N006	E51-N683	ECCS	Static-O-Ring	6N-AA21-(X9)VSTT	Differential Pressure Transmitter	Barton	764	H21-P417B
54)	RCIC Turbine Exhaust Diaphragm Pressure High	E51-N685 A,B,C,D	E51-N012 A,B,C,D	E51-N085 A,B,C,D	ECCS	Barksdale	D2H-M80SS	Differential Pressure Transmitter	Barton	764	A,C-H21-P417A B,D-H21-P437
55)	RWCU Room Temperature Inlet (no trip)	G31-N661 A,D,E,H, J,M	G31-N023* A,B,C,D, E,F	G31-N061 A,D,E,H, J,M	RPS	Pyco	N145C3224P1 Type T	RTD	Weed	1A0D	Local

* Slave trip unit.

** Not being deleted; only the safety function being replaced by ATTS.

a. Transmitters having the same MPL No. are the same equipment providing several trip functions.

b. Separated for divisional considerations.

c. The complete RTD model number for all present ATTS applications is: 1A0D-611-1B-C-4-C-2-A2-0.

TABLE 3.1 (SHEET 8 OF 8)

INSTRUMENT LOOP INFORMATION

Variable Name[a]	Trip Unit MPL No.	Primary Sensor Old MPL No.	Primary Sensor New MPL No.	Engineering Safeguard Division	Existing Device Manufacturer	Existing Device Model No.	New Device Generic Name	New Device Manufacturer	New Device Model No.[c]	Associated Rack[b]
56) RWCU Area Ventilation Differential Temperature High	G31-N663 A,D,E,H, J,M	G31-N602** A,B,C,D, E,F	NA	RPS	NA	NA	NA	NA	NA	NA
57) RWCU Room Outlet Ambient Temperature High	G31-N662 A,D,E,H, J,M	G31-N022** A,B,C,D, E,F	G31-N062 A,D,E,H, J,M	RPS	Pyco	N145C3224P1 Type T	RTD	Weed	1A0D	Local
58) Safety Relief Valve Open	NA	NA	B21-N302A B21-N302B B21-N302C B21-N302D B21-N302E B21-N302F B21-N302G B21-N302H B21-N302K B21-N302L B21-N302M	ECCS	NA	NA	Pressure Switch	PCI	A17-1P	Local

** Not being deleted; only the safety function is being replaced by ATTS.

a. Transmitters having the same MPL No. are the same equipment having several trip settings.

b. Separated for divisional considerations.

c. The complete RTD model number for all present ATTS applications is: 1A0D-611-1B-C-4-C-2-A2-0.

TABLE 3.2

CORRELATION BETWEEN EQUIPMENT AND MOUNTING LOCATIONS
(For Use With Seismic Figures 3.9 through 3.21)

	Description	Location	Figure
1)	Cabinets H11-P921 to H11-P928	Main Control Room - el 164 ft (Control Building)	3.10, 3.19
2)	Racks		
	MPL No.		
a)	H21-P401	Reactor Building - el 87 ft (NE Corner Room)	3.9, 3.10, 3.11
b)	H21-P402	Reactor Building - el 158 ft	3.15, 3.16, 3.17
c)	H21-P404 A-E	Reactor Building - el 158 ft	3.15, 3.16, 3.17
d)	H21-P405 A-E	Reactor Building - el 158 ft	3.15, 3.16, 3.17
e)	H21-P409	Reactor Building - el 130 ft	3.12, 3.13, 3.14
f)	H21-P410	Reactor Building - el 130 ft	3.12, 3.13, 3.14
g)	H21-P414 A/B	Reactor Building - el 87 ft (HPCI Room)	3.9, 3.10, 3.11
h)	H21-P415 A/B	Reactor Building - el 130 ft	3.12, 3.13, 3.14
i)	H21-P016 (Wall)	Reactor Building - el 82 ft (HPCI Room)	3.12*, 3.13*, 3.14*
j)	H21-P417 A/B	Reactor Building - el 87 ft (RCIC Corner Room)	3.9, 3.10, 3.11
k)	H21-P418 A/B	Reactor Building - el 87 ft (NE Corner Room)	3.9, 3.10, 3.11
l)	H21-P016 (Wall)	Reactor Building - el 87 ft (SE Corner Room)	3.9*, 3.10*, 3.11*
m)	H21-P421 A/B	Reactor Building - el 87 ft (SE Corner Room)	3.9, 3.10, 3.11
n)	H21-P425 A/B	Reactor Building - el 130 ft	3.12, 3.13, 3.14
o)	H21-P434 (Wall)	Reactor Building - el 87 ft (HPCI Room)	3.12*, 3.13*, 3.14*
p)	H21-P435 (Wall)	Reactor Building - el 130 ft	3.15*, 3.16*, 3.17*
q)	H21-P036 (Wall)	Reactor Building - el 130 ft	3.15*, 3.16*, 3.17*

TABLE 3.2 (Continued)

	Description	Location	Figure
	r) H21-P437 (Wall)	Reactor Building - el 87 ft (RCIC Corner Room)	3.12*, 3.13*, 3.14*
	s) H21-P038 (Wall)	Reactor Building - el 87 ft (Torus Room)	3.12*, 3.13*, 3.14*
3)	Pressure Switches, RTDs	Use Multiple Locations-Plant Hatch Spectrum Peak Envelope Curves	3.20, 3.21
4)	Locally Mounted Transmitters	Use Multiple Locations - Plant Hatch Spectrum Peak Envelope Curves	3.20, 3.21

*The next higher elevation seismic curves should be used for the wall racks.

FIGURE 3.1

ASSIGNED TRIP UNIT LOCATIONS IN ECCS CARD FILES FOR CABINET H11-P925[d]

1	2	3	4	5	6	7	8	9	10	11	12
E51-N658A[a] E51-N058A[b] RCIC STM LN[c] LOW PR	E51-N658C E51-N058C RCIC STM LN LOW PR	E51-N657A E51-N057A RCIC STM LN HI ΔP(+)	E51-N660A SLAVE RCIC STM LN HI ΔP(-)	E51-N685A E51-N085A RCIC TB EX DIA HI PP	E51-N685C E51-N085C RCIC TB EX DIA HI PR	E51-N656A E51-N056A RCIC TB EX HI PR	E51-N656C E51-N056C RCIC TB EX HI PR	E51-N651 E51-N051 RCIC PUMP HI FLOW		E51-N683 E51-N083 RCIC PUMP SUC LOW PR	E51-N650 E51-N050 RCIC PUMP HI PR
E41-N658A[a] E41-N058A[b] HPCI STM LN[c] LOW PR	E41-N658C E41-N058C HPCI STM LN LOW PR	E41-N657A E41-N057A HPCI STM LN HI ΔP(+)	E41-N660A SLAVE HPCI STM LN HI ΔP(-)	E41-N655A E41-N055A HPCI TB EX DIA HI PR	E41-N655C E41-N055C HPCI TB EX DIA HI PR				B21-N620A B21-N120A LLS SCRAM PR PERM	B21-N621A SLAVE LLS CONTROL	B21-N622A B21-N122A LLS CONTROL
B21-N691A[a] B21-N091A[b] WTR LVL 1[c] CS/ADS/RHR DIESEL	B21-N692A SLAVE WTR LVL 2 HPCI/RCIC DIESEL	B21-N693A B21-N093A WTR LVL 8 RCIC	B21-N691C B21-N091C WTR LVL 1 CS/ADS/RHR	B21-N692C SLAVE WTR LVL 2 HPCI/RCIC	B21-N693C SLAVE WTR LVL 8 RCIC	B21-N695A B21-N095A WTR LVL 3 ADS	B21-N685A B21-N085A WTR LVL 0 RHR	E11-N682A E11-N082A RHR PUMP A/C LOW FLOW	B21-N620C B21-N120C LLS SCRAM PR PERM	B21-N621C SLAVE LLS CONTROL	B21-N622C B21-N122C LLS CONTROL

a. Trip unit MPL No.
b. Sensor MPL No.
c. Loop function.
d. Blank spaces denote empty file slots.

FIGURE 3.2[a]

ASSIGNED TRIP UNIT LOCATIONS IN ECCS CARD FILES FOR CABINET H11-P926[b]

1	2	3	4	5	6	7	8	9	10	11	12
E51-N658B E51-N058B RCIC STM LN LOW PR	E51-N658D E51-N058D RCIC STM LN LOW PR	E51-N657B E51-N057B RCIC STM LN HI ΔP(+)	E51-N660B SLAVE RCIC STM LN HI ΔP(-)	E51-N685B E51-N085B RCIC TB EX DIA HI PR	E51-N685D E51-N085D RCIC TB EX DIA HI PR	E41-N656B E41-N056B HPCI TB EX HI PR	E41-N656D E41-N056D HPCI TB EX HI PR	E41-N651 E41-N051 HPCI PUMP HI FLOW		E41-N653 E41-N053 HPCI PUMP SUC LOW PR	E41-N650 E41-N050 HPCI PUMP HI PR
E41-N658B E41-N058B HPCI STM LN LOW PR	E41-N658D E41-N058D HPCI STM LN LOW PR	E41-N657B E41-N057B HPCI STM LN HI ΔP(+)	E41-N660B SLAVE HPCI STM LN HI ΔP(-)	E41-N655B E41-N055B HPCI TB EX DIA HI PR	E41-N655D E41-N055D HPCI TB EX DIA HI PR	E41-N662B E41-N062B HPCI TORUS HI WTR LVL	E41-N662D E41-N062D HPCI TORUS HI WTR LVL		B21-N620B B21-N120B LLS SCRAM PR PREM	B21-N621B SLAVE LLS CONTROL	B21-N622B B21-N122B LLS CONTROL
B21-N691B B21-N091B WTR LVL 1 CS/ADS/RHR DIESEL	B21-N692B SLAVE WTR LVL 2 HPCI/RCIC	B21-N693B B21-N093B WTR LVL 8 HPCI	B21-N691D B21-N091D WTR LVL 1 CS/ADS/RHR DIESEL	B21-N692D SLAVE WTR LVL 2 HPCI/RCIC	B21-N693D SLAVE WTR LVL 8 HPCI	B21-N695B B21-N095B WTR LVL 3 ADS	B21-N685B B21-N085B WTR LVL 0 RHR	E11-N682B E11-N082B RHR PUMP B/D LOW FLOW	B21-N620D B21-N120D LLS SCRAM PR PERM	B21-N621D SLAVE LLS CONTROL	B21-N622D B21-N122D LLS CONTROL

a. See figure 3.1 for description of nomenclature.
b. Blank spaces denote empty file slots.

FIGURE 3.3[a]

ASSIGNED TRIP UNIT LOCATIONS IN ECCS CARD FILES FOR CABINET H11-P927[b]

1	2	3	4	5	6	7	8	9	10	11	12
E11-N655A E11-N055A RHR PUMP A HI PR	E11-N655C E11-N055C RHR PUMP C HI PR	E11-N656A E11-N056A RHR PUMP A HI PR	E11-N656C E11-N056C RHR PUMP C HI PR	E11-N694A E11-N094A DRYWELL HI PR	E11-N694C E11-N094C DRYWELL HI PR	B21-N690A B21-N090A VESSEL LOW PR	B21-N690C B21-N090C VESSEL LOW PR	B21-N641C SLAVE VESSEL LOW PR	B21-N690E B21-N090E VESSEL LOW PR		
E21-N651A E21-N051A CS PUMP A LOW FLOW		E21-N655A E21-N055A CS PUMP A HI PR	E21-N652A E21-N052A CS PUMP A HI PR			E51-N663A E51-N063A RCIC TORUS AMB LO(NT)[c]	E51-N664A E51-N064A RCIC TORUS AMB HI(NT)[c]	E51-N665A NA RCIC TORUS HI ΔT	E51-N663C E51-N063C HPCI TORUS AMB LO(NT)[c]	E51-N664C E51-N064C HPCI TORUS AMB HI(NT)[c]	E51-N665C NA HPCI TORUS HI ΔT
E41-N670A E41-N070A HPCI EQUIP AMB HI TMP	E41-N671A E41-N071A HPCI PIPE RM AMB HI TMP		E51-N661A E51-N061A RCIC EQUIP AMB HI TMP			E51-N666A E51-N066A RCIC TORUS AMB HI TMP	E51-N666C E51-N066C HPCI TORUS AMB HI TMP				

a. See figure 3.1 for description of nomenclature.
b. Blank spaces denote empty file slots.
c. (NT) denotes instrument with no trip function.

FIGURE 3.4[a]

ASSIGNED TRIP UNIT LOCATIONS IN ECCS CARD FILES FOR CABINET H11-P928[b]

1	2	3	4	5	6	7	8	9	10	11	12
E11-N655B E11-N055B RHR PUMP B HI PR	E11-N655D E11-N055D RHR PUMP D HI PR	E11-N656B E11-N056B RHR PUMP B HI PR	E11-N656D E11-N056D RHR PUMP D HI PR	E11-N694B E11-N094B DRYWELL HI PR	E11-N694D E11-N094D DRYWELL HI PR	B21-N690B B21-N090B VESSEL LOW PR	B21-N641B SLAVE VESSEL LOW PR	B21-N690D B21-N090D VESSEL LOW PR		B21-N690F B21-N090F VESSEL LOW PR	
E21-N651B E21-N051B CS PUMP B LOW FLOW		E21-N655B E21-N055B CS PUMP B HI PR	E21-N652B E21-N052B CS PUMP B HI PR			E51-N663B E51-N063B RCIC TORUS AMB LO(NT)[c]	E51-N664B E51-N064B RCIC TORUS AMB HI(NT)[c]	E51-N665B NA RCIC TORUS HI ΔT	E51-N663D E51-N063D HPCI TORUS AMB LO(NT)[c]	E51-N664D E51-N064D HPCI TORUS AMB HI(NT)[c]	E51-N665D NA HPCI TORUS HI ΔT
E41-N670B E41-N070B HPCI EQUIP AMB HI TMP	E41-N671B E41-N071B HPCI PIPE RM AMB HI TMP		E51-N661B E51-N061B RCIC EQUIP AMB HI TMP			E51-N666B E51-N066B RCIC TORUS AMB HI TMP	E51-N666D E51-N066D HPCI TORUS AMB HI TMP				

a. See figure 3.1 for description of nomenclature.
b. Blank spaces denote empty file slots.
c. (NT) denotes instrument with no trip function.

FIGURE 3.5[a]

ASSIGNED TRIP UNIT LOCATIONS IN RPS CARD FILES FOR CABINET H11-P921[b]

1	2	3	4	5	6	7	8	9	10	11	12
B21-N686A B21-N086A MN ST LN A HI FLOW	B21-N687A B21-N087A MN ST LN B HI FLOW	B21-N688A B21-N088A MN ST LN C HI FLOW	B21-N689A B21-N089A MN ST LN D HI FLOW	B21-N680A B21-N080A WTR LVL 3 RPS	B21-N681A B21-N081A WTR LVL 1 MSIV	B21-N682A SLAVE WTR LVL 2 ISOL	B21-N678A B21-N078A VESSEL HI PR	B31-N679A B31-N079A VESSEL LOW PR	C71-N650A C71-N050A DRYWELL HI PR		
						B21-N623A B21-N123A STM TUNN HI TMP	B21-N624A B21-N124A STM TUNN HI TMP		B21-N625A B21-N125A STM TUNN HI TMP	B21-N626A B21-N126A STM TUNN HI TMP	
			G31-N661A G31-N061A RWCU RM IN TMP(NT)[c]	G31-N662A G31-N062A RWCU RM OUT AMB HI TMP	G31-N663A NA RWCU AREA HI ΔT	G31-N661E G31-N061E RWCU RM IN TMP(NT)[c]	G31-N662E G31-N062E RWCU RM OUT AMB HI TMP	G31-N663E NA RWCU AREA HI ΔT	G31-N661J G31-N061J RWCU RM IN TMP(NT)[c]	G31-N662J G31-N062J RWCU RM OUT AMB HI TMP	G31-N663J NA RWCU AREA HI ΔT

a. See figure 3.1 for description of nomenclature.
b. Blank spaces denote empty file slots.
c. (NT) denotes instrument with no trip function.

FIGURE 3.6[a]

ASSIGNED TRIP UNIT LOCATIONS IN RPS CARD FILES FOR CABINET H11-P922[b]

1	2	3	4	5	6	7	8	9	10	11	12
B21-N686B B21-N086B MN ST LN A HI FLOW	B21-N687B B21-N087B MN ST LN B HI FLOW	B21-N688B B21-N088B MN ST LN C HI FLOW	B21-N689B B21-N089B MN ST LN D HI FLOW	B21-N680B B21-N080B WTR LVL 3 RPS	B21-N681B B21-N081B WTR LVL 1 MSIV	B21-N682B SLAVE WTR LVL 2 ISOL	B21-N678B B21-N078B VESSEL HI PR		C71-N650B C71-N050B DRYWELL HI PR		
						B21-N623B B21-N123B STM TUNN HI TMP	B21-N624B B21-N124B STM TUNN HI TMP		B21-N625B B21-N125B STM TUNN HI TMP	B21-N626B B21-N126B STM TUNN HI TMP	

a. See figure 3.1 for description of nomenclature.
b. Blank spaces denote empty file slots.

FIGURE 3.7[a]

ASSIGNED TRIP UNIT LOCATIONS IN RPS CARD FILES FOR CABINET H11-P923[b]

1	2	3	4	5	6	7	8	9	10	11	12
B21-N686C B21-N086C MN ST LN A HI FLOW	B21-N687C B21-N087C MN ST LN B HI FLOW	B21-N688C B21-N088C MN ST LN C HI FLOW	B21-N689C B21-N089C MN ST LN D HI FLOW	B21-N680C B21-N080C WTR LVL 3 RPS	B21-N681C B21-N081C WTR LVL 1 MSIV	B21-N682C SLAVE WTR LVL 2 ISOL	B21-N678C B21-N078C VESSEL HI PR		C71-N650C C71-N050C DRYWELL HI PR		
						B21-N623C B21-N123C STM TUNN HI TMP	B21-N624C B21-N124C STM TUNN HI TMP		B21-N625C B21-N125C STM TUNN HI TMP	B21-N626C B21-N126C STM TUNN HI TMP	

a. See figure 3.1 for description of nomenclature.
b. Blank spaces denote empty file slots.

FIGURE 3.8[a]

ASSIGNED TRIP UNIT LOCATIONS IN RPS CARD FILES FOR CABINET H11-P924[b]

1	2	3	4	5	6	7	8	9	10	11	12
B21-N686D B21-N086D MN ST LN A HI FLOW	B21-N687D B21-N087D MN ST LN B HI FLOW	B21-N688D B21-N088D MN ST LN C HI FLOW	B21-N689D B21-N089D MN ST LN D HI FLOW	B21-N680D B21-N080D WTR LVL 3 RPS	B21-N681D B21-N081D WTR LVL 1 MSIV	B21-N682D SLAVE WTR LVL 2 ISOL	B21-N678D B21-N078D VESSEL HI PR	B31-N679D B31-N079D VESSEL LO PR	C71-N650D C71-N050D DRYWELL HI PR		
						B21-N623D B21-N123D STM TUNN HI TMP	B21-N624D B21-N124D STM TUNN HI TMP		B21-N625D B21-N125D STM TUNN HI TMP	B21-N626D B21-N126D STM TUNN HI TMP	
			G31-N661D G31-N061D RWCU RM IN TMP(NT)[c]	G31-N662D G31-N062D RWCU RM OUT AMB HI TMP	G31-N663D NA RWCU AREA HI ΔT	G31-N661H G31-N061H RWCU RM IN TMP(NT)[c]	G31-N662H G31-N062H RWCU RM OUT AMB HI TMP	G31-N663H NA RWCU AREA HI ΔT	G31-N661M G31-N061M RWCU RM IN TMP(NT)[c]	G31-N662M G31-N062M RWCU RM OUT AMB HI TMP	G31-N663M NA RWCU AREA HI ΔT

a. See figure 3.1 for description of nomenclature.
b. Blank spaces denotes empty file slots.
c. (NT) denotes instrument with no trip function.

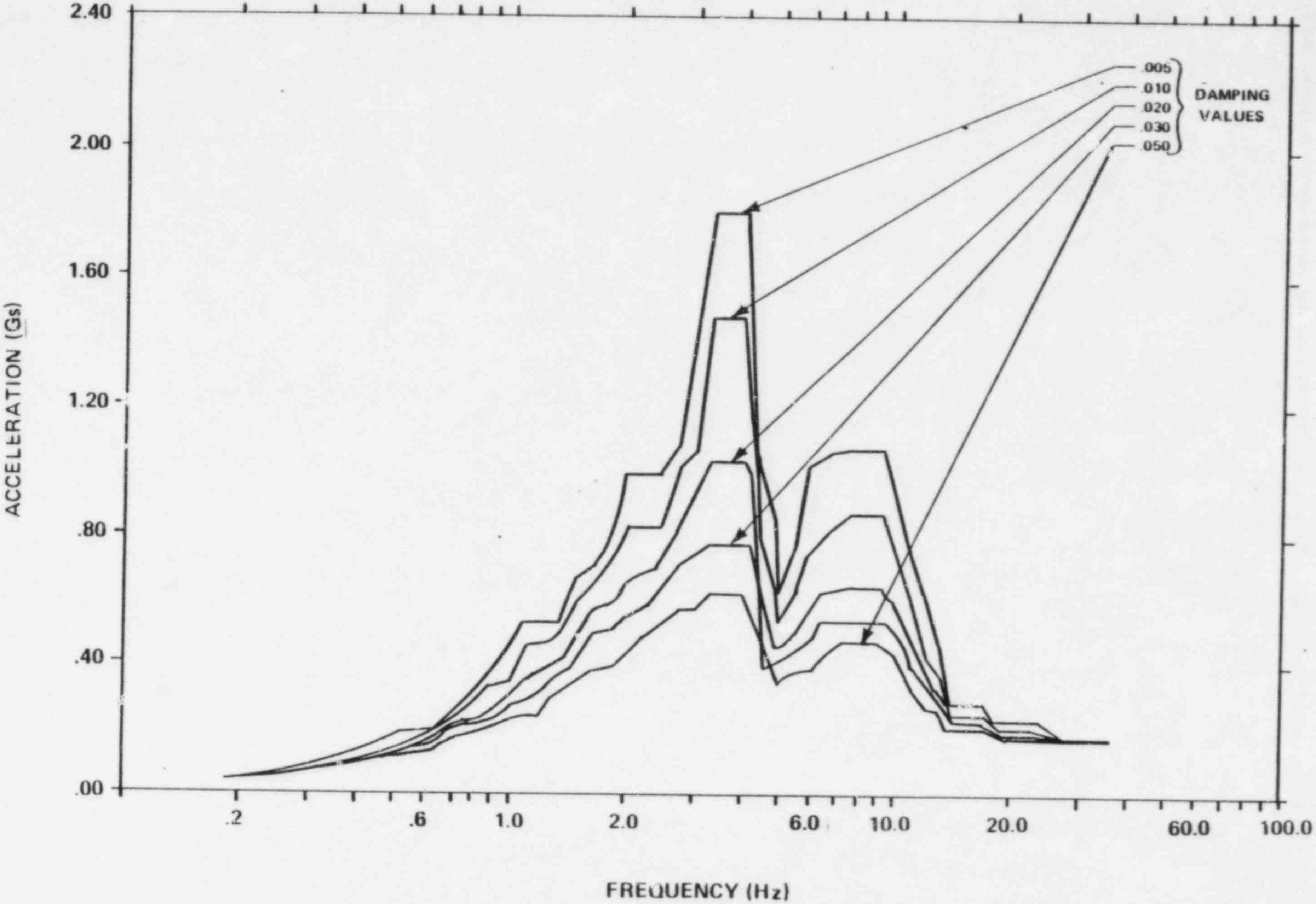


FIGURE 3.9 HNP-2 REACTOR BUILDING EAST-WEST SSE
el 87 ft

9389-2

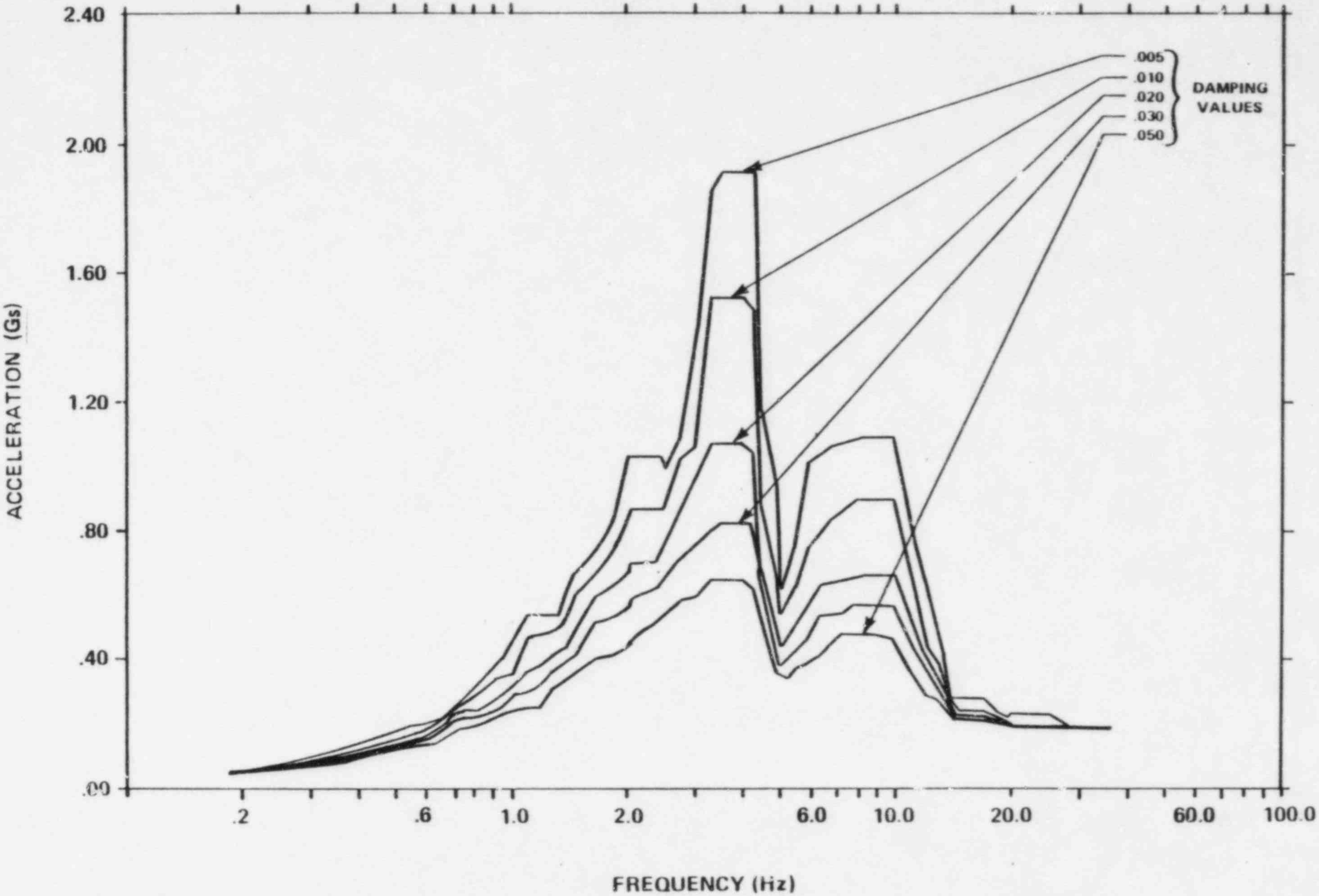


FIGURE 3.10 HNP-2 REACTOR BUILDING NORTH-SOUTH SSE
el 87 ft

9389-2

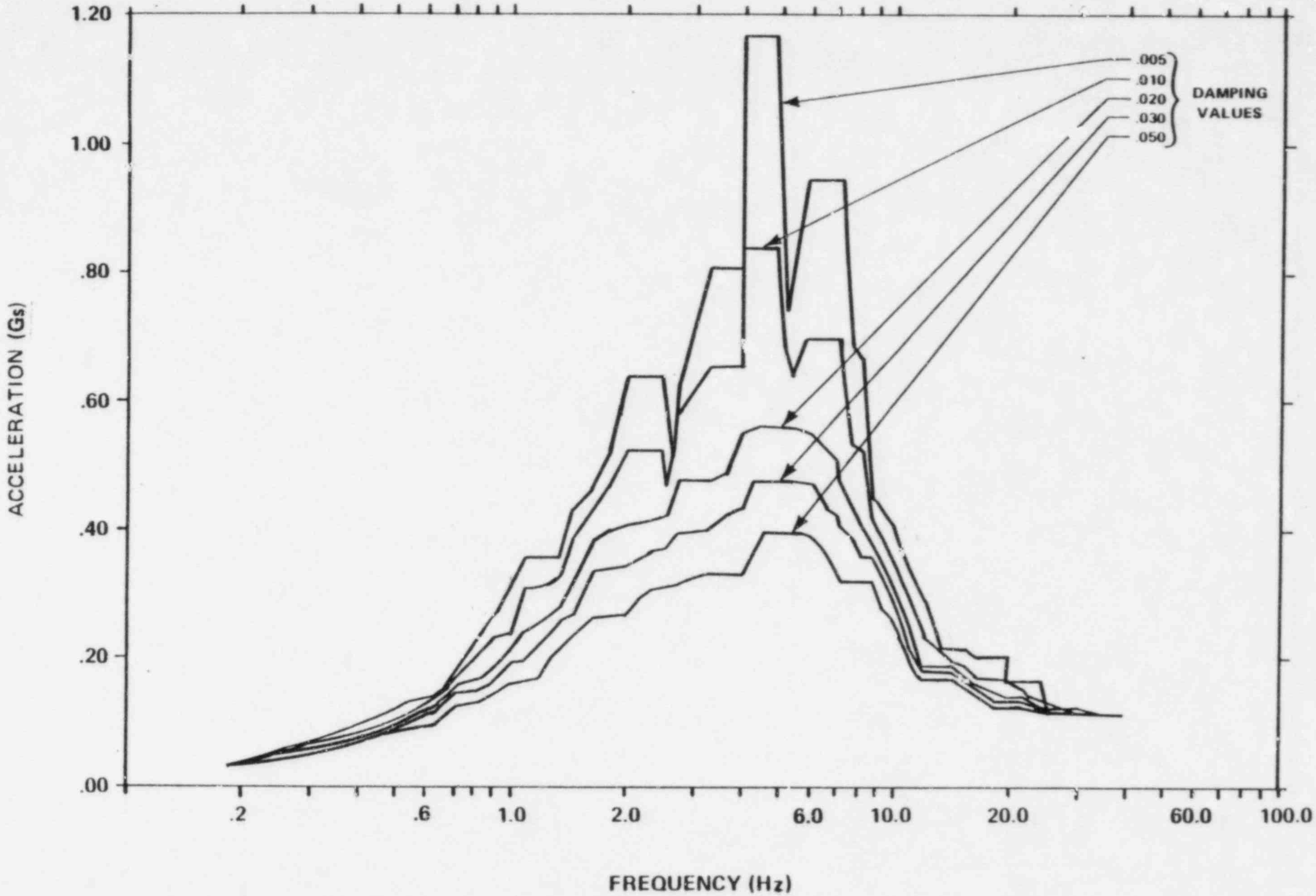


FIGURE 3.11 HNP-2 REACTOR BUILDING VERTICAL SSE
el 87 ft

9389-2

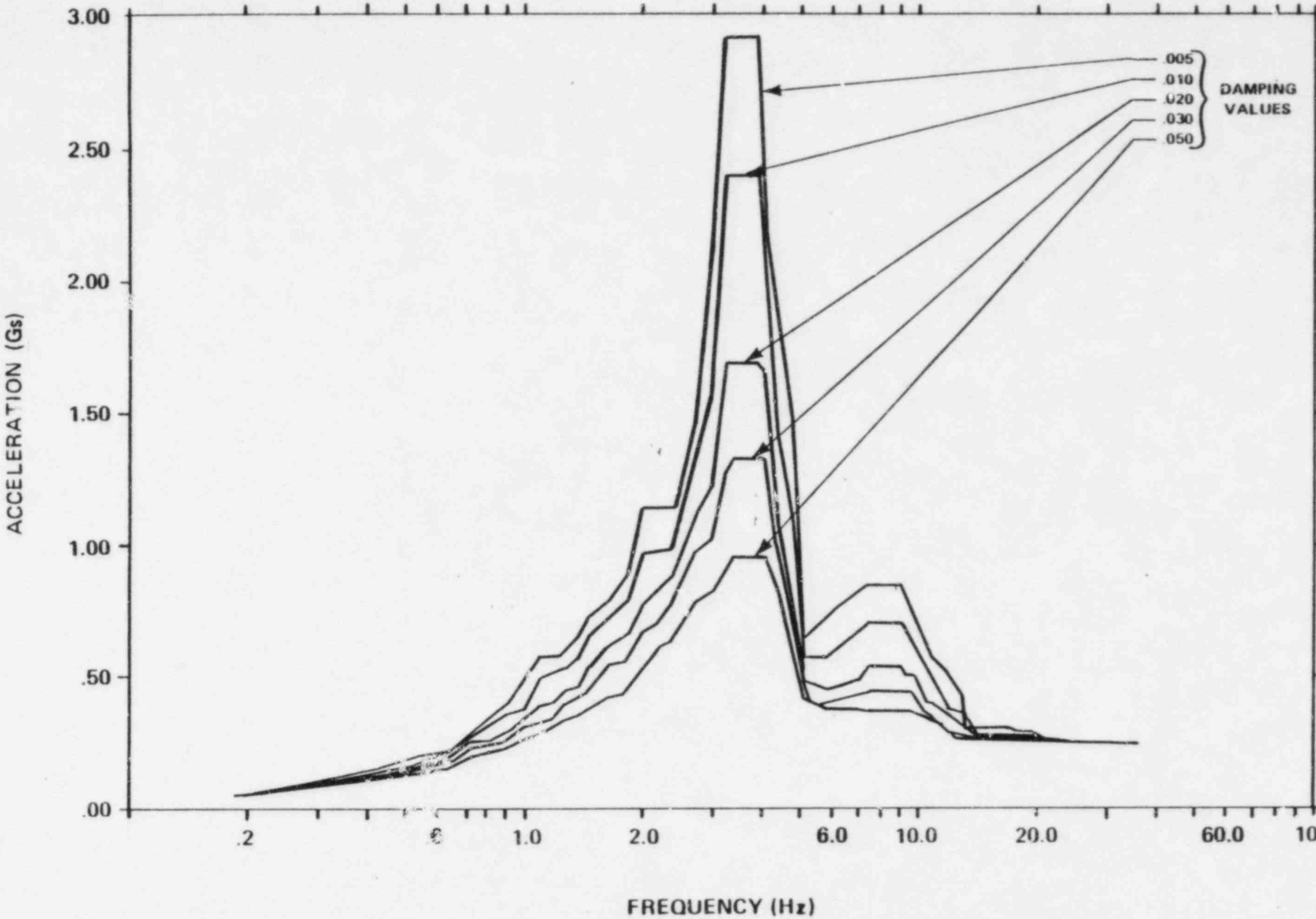


FIGURE 3.12 HNP-2 REACTOR BUILDING EAST-WEST SSE
el 130 ft

9389-2

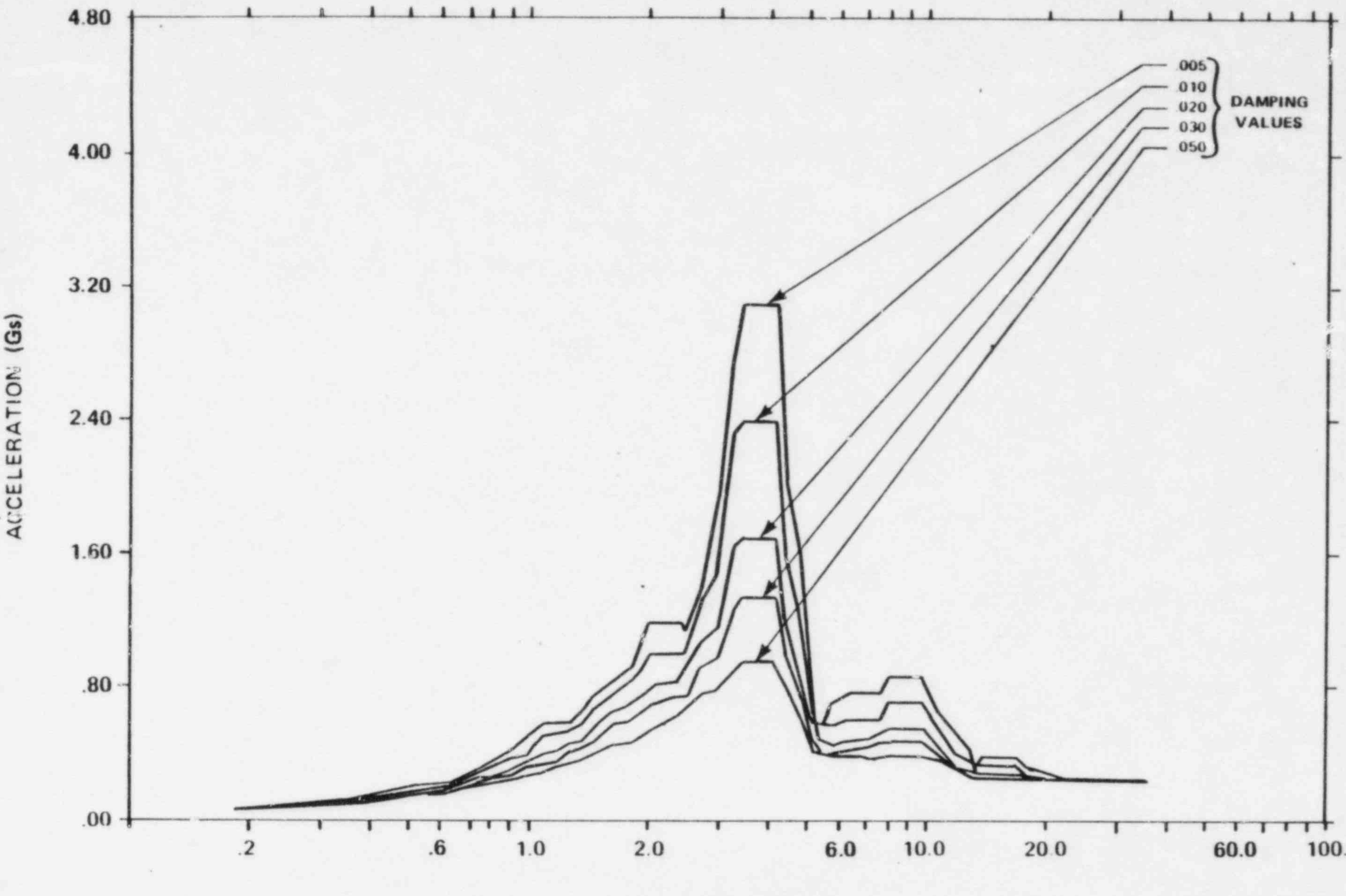


FIGURE 3.13 HNP-2 REACTOR BUILDING NORTH-SOUTH SSE
el 130 ft

9389-2

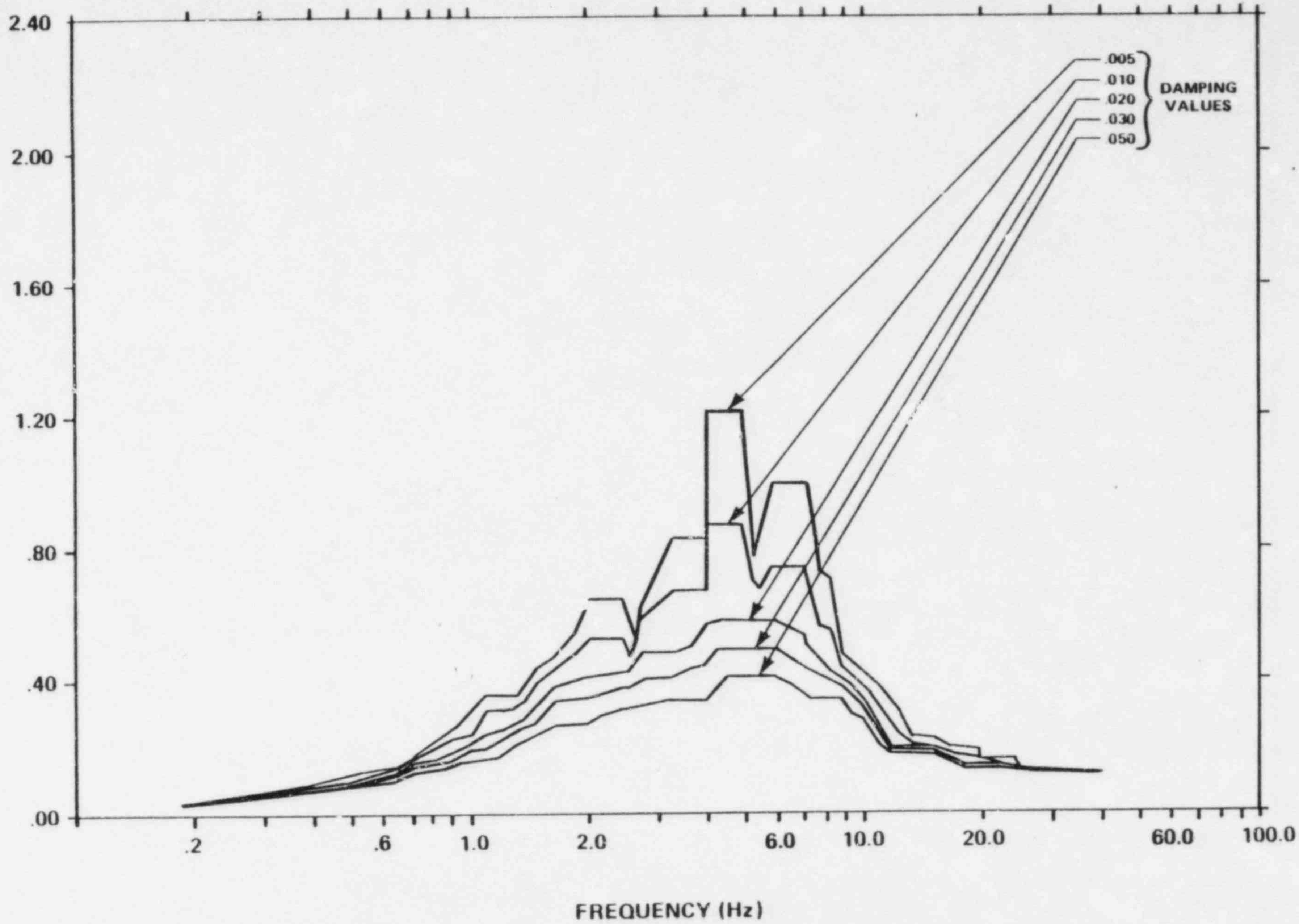


FIGURE 3.14 HNP-2 REACTOR BUILDING VERTICAL SSE
el 130 ft

9389-2

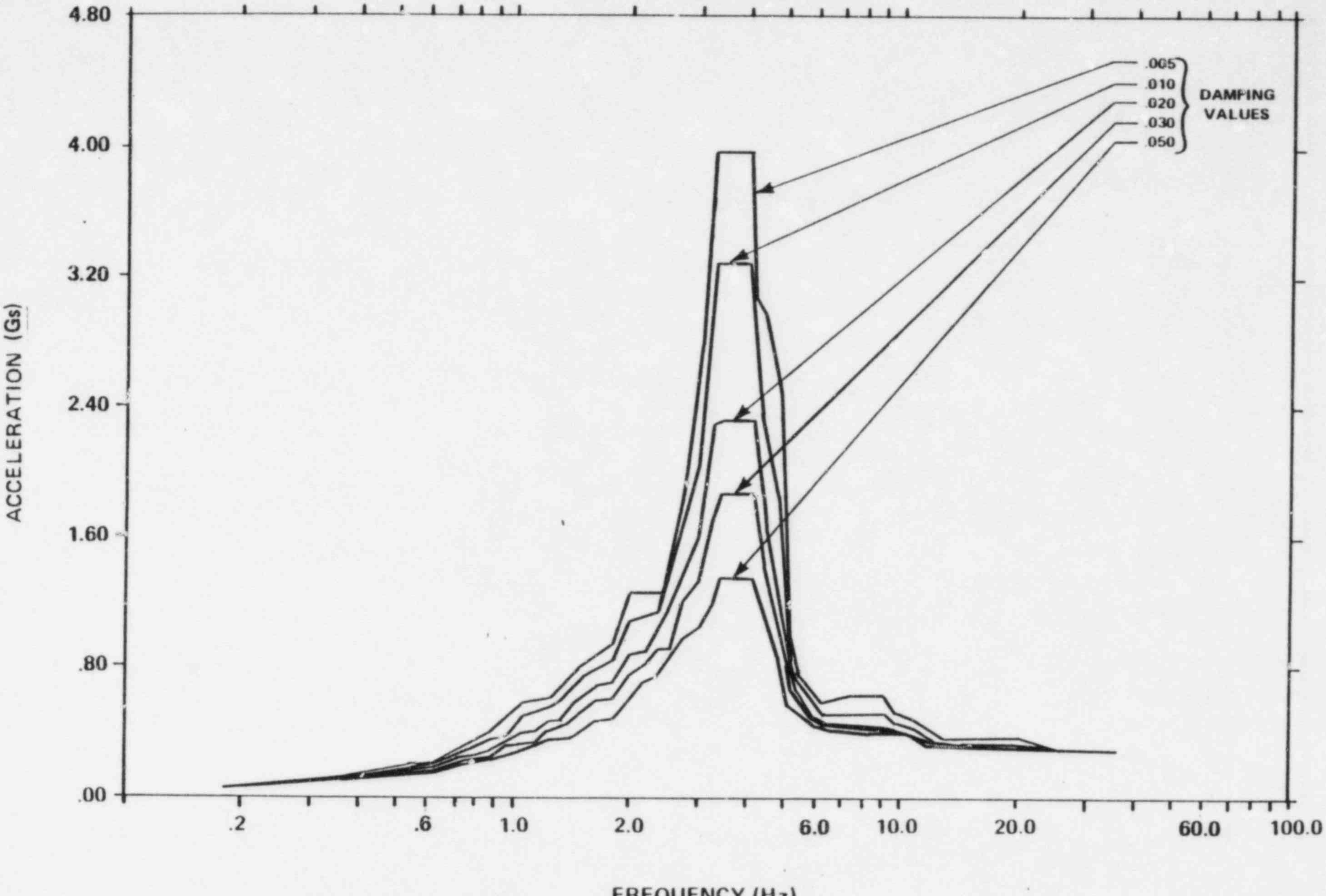


FIGURE 3.15 HNP-2 REACTOR BUILDING EAST-WEST SSE
el 158 ft

9389-2

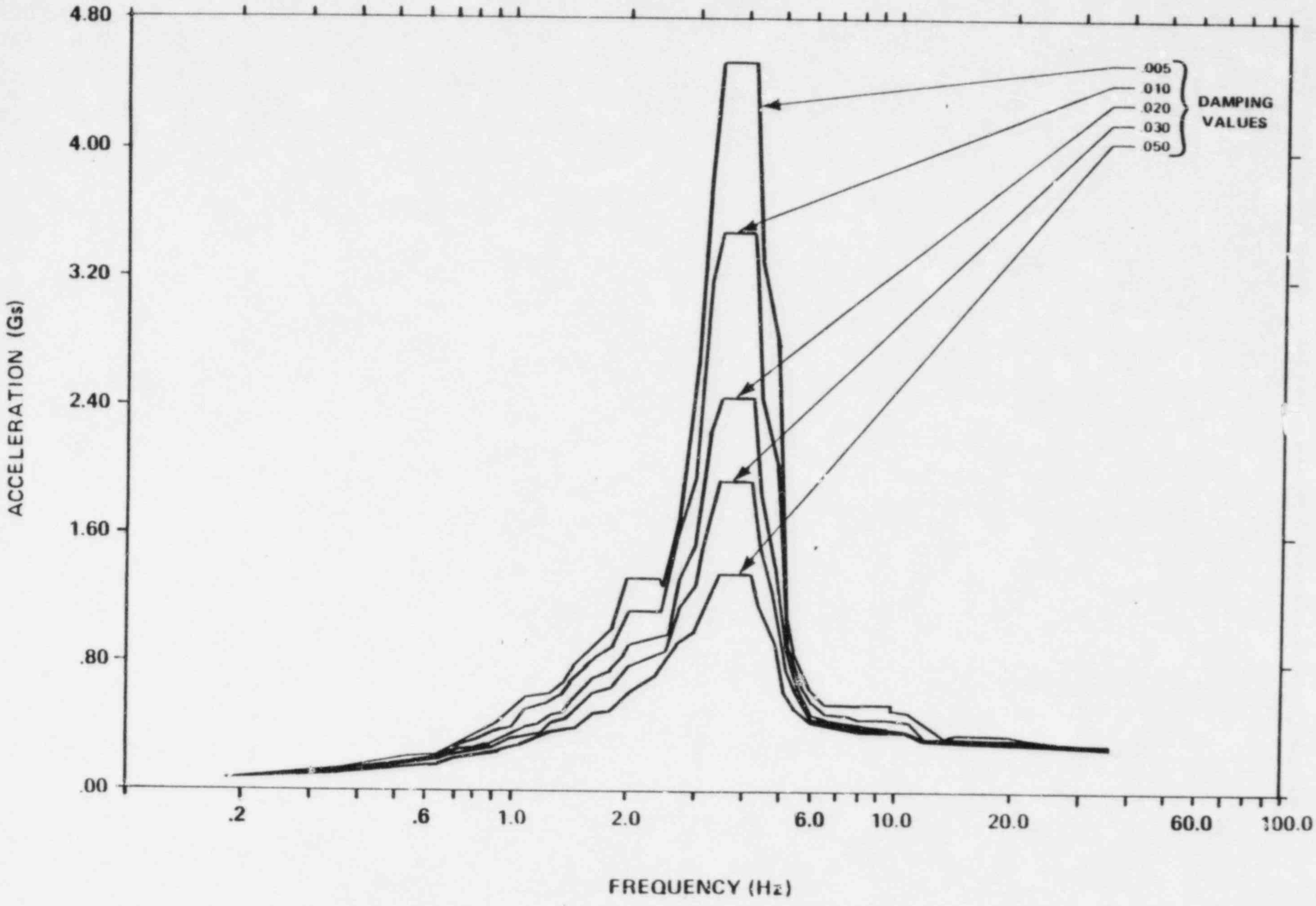


FIGURE 3.16 HNP-2 REACTOR BUILDING NORTH-SOUTH SSE
el 158 ft

9389-2

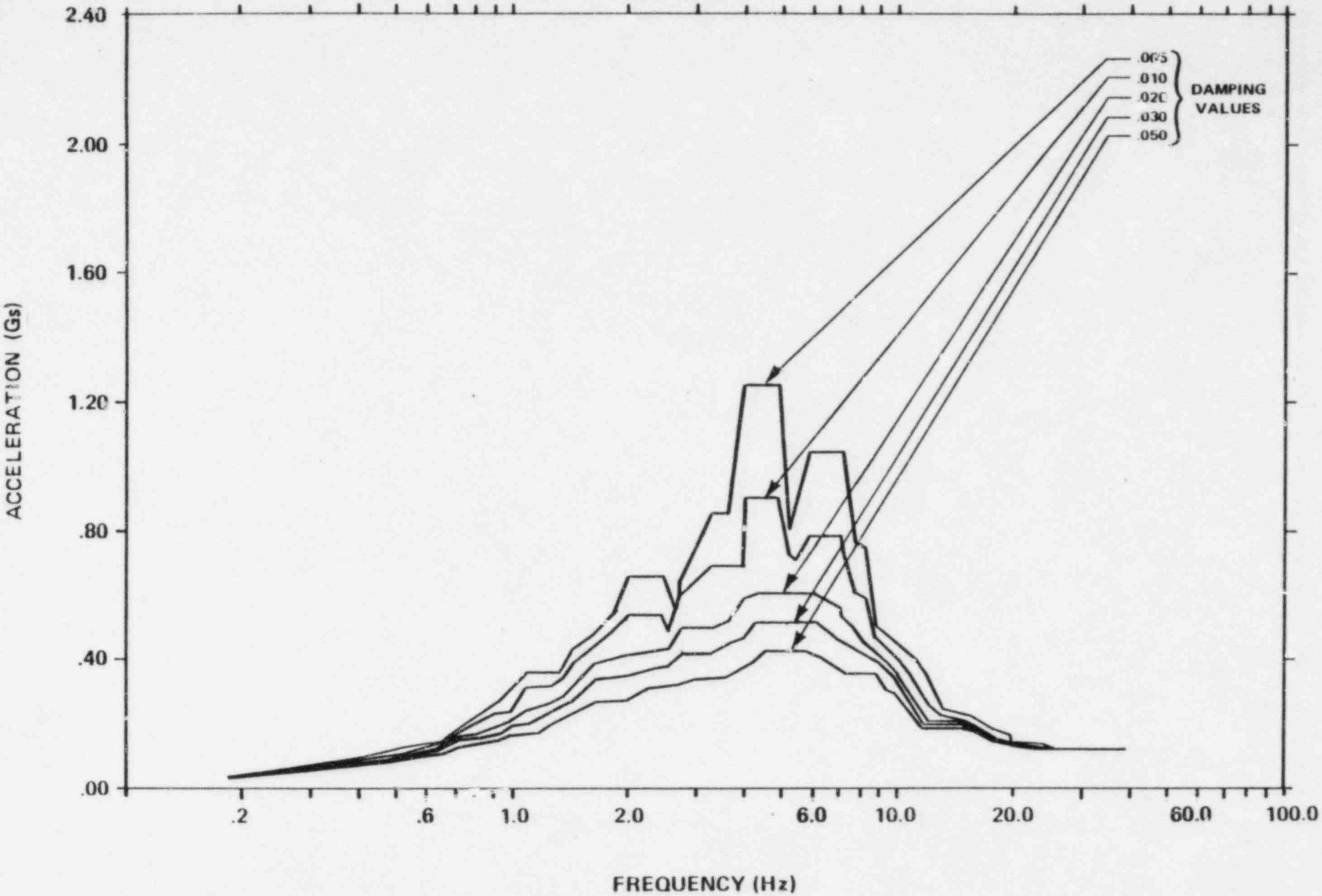


FIGURE 3.17 HNP-2 REACTOR BUILDING VERTICAL SSE
el 158 ft

9389-2

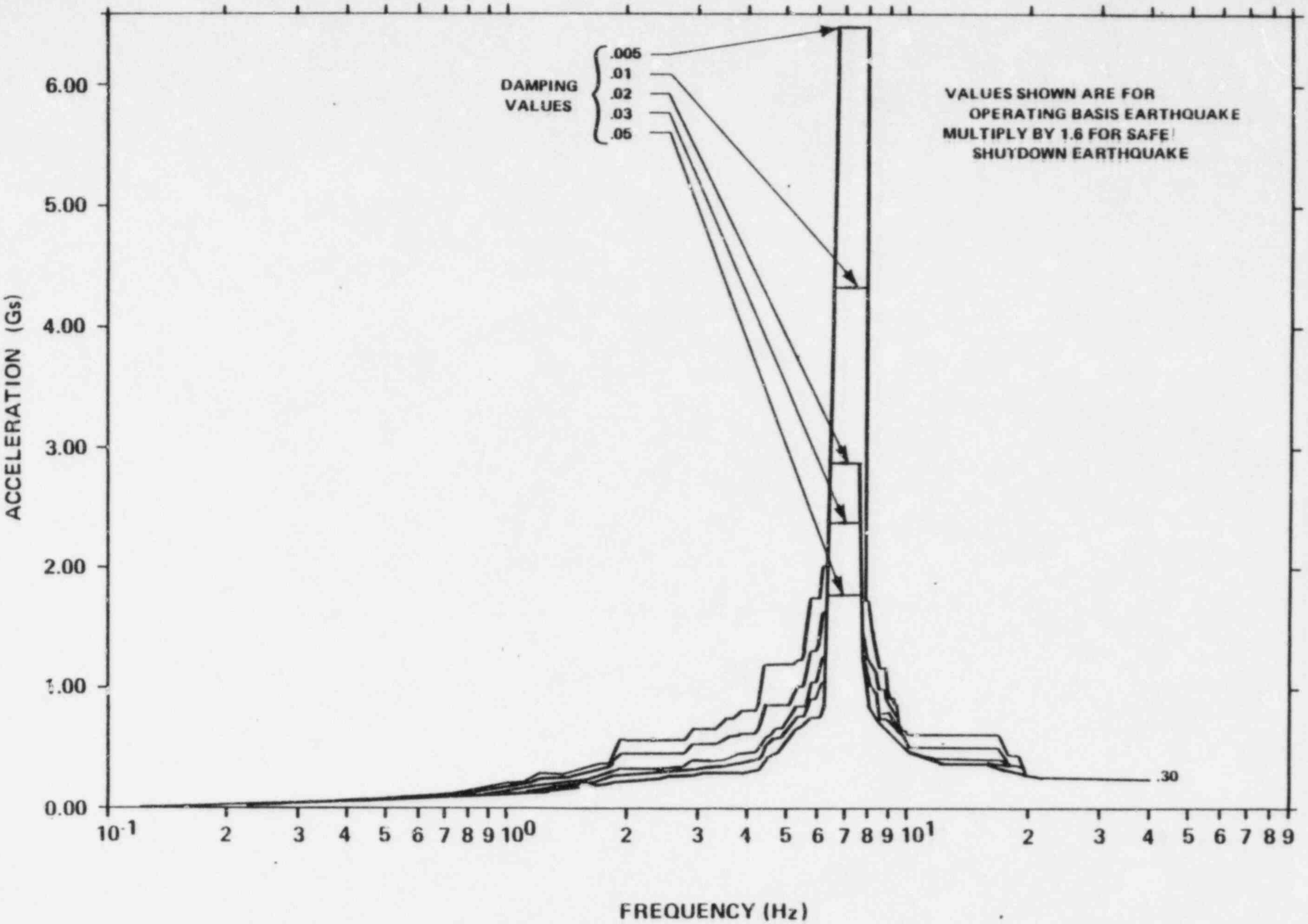


FIGURE 3.18 HNP-2 CONTROL BUILDING HORIZONTAL OBE
el 164 ft

9389-2

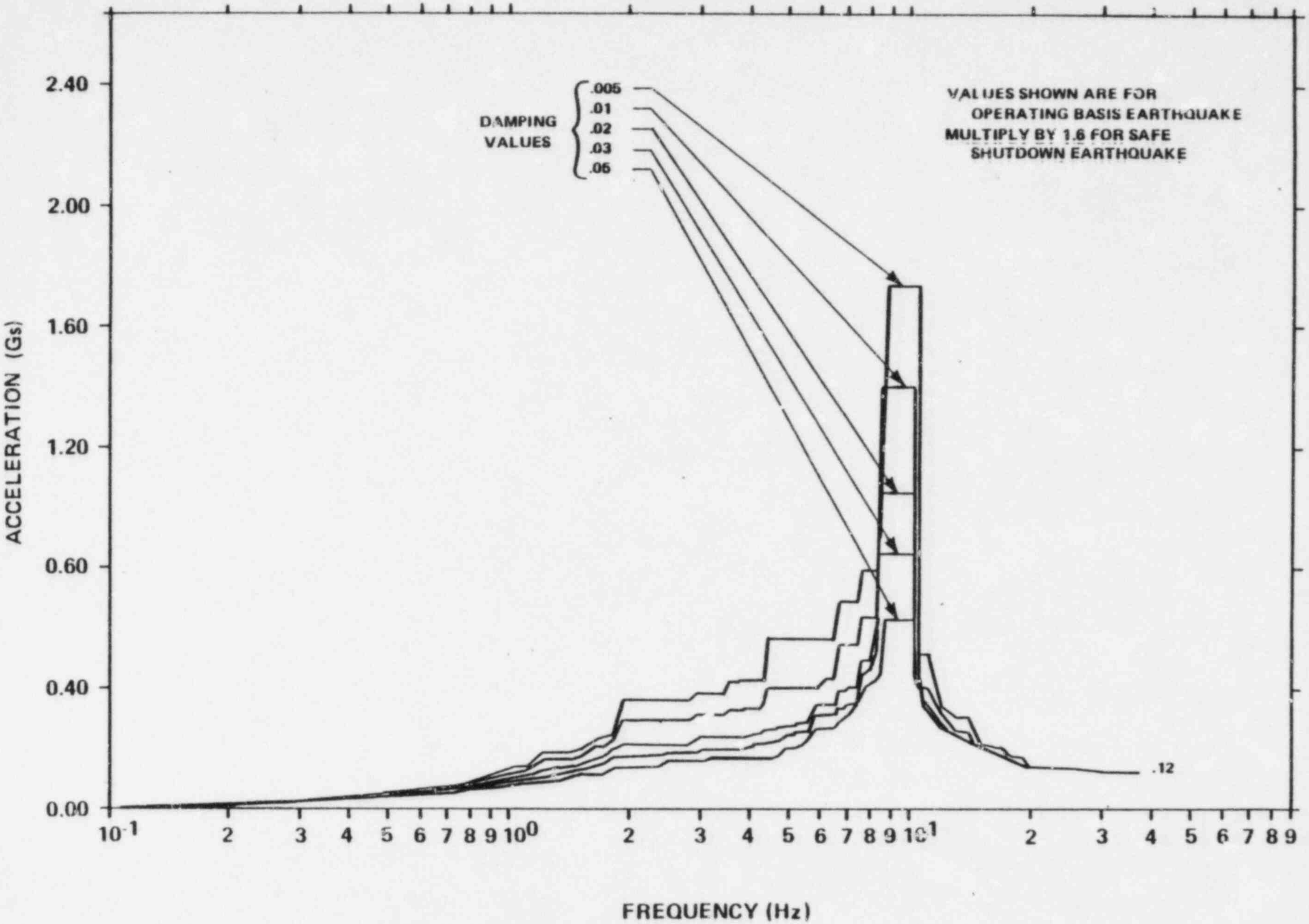


FIGURE 3.19 HNP-2 CONTROL BUILDING VERTICAL OBE
el 164 ft

9389-2

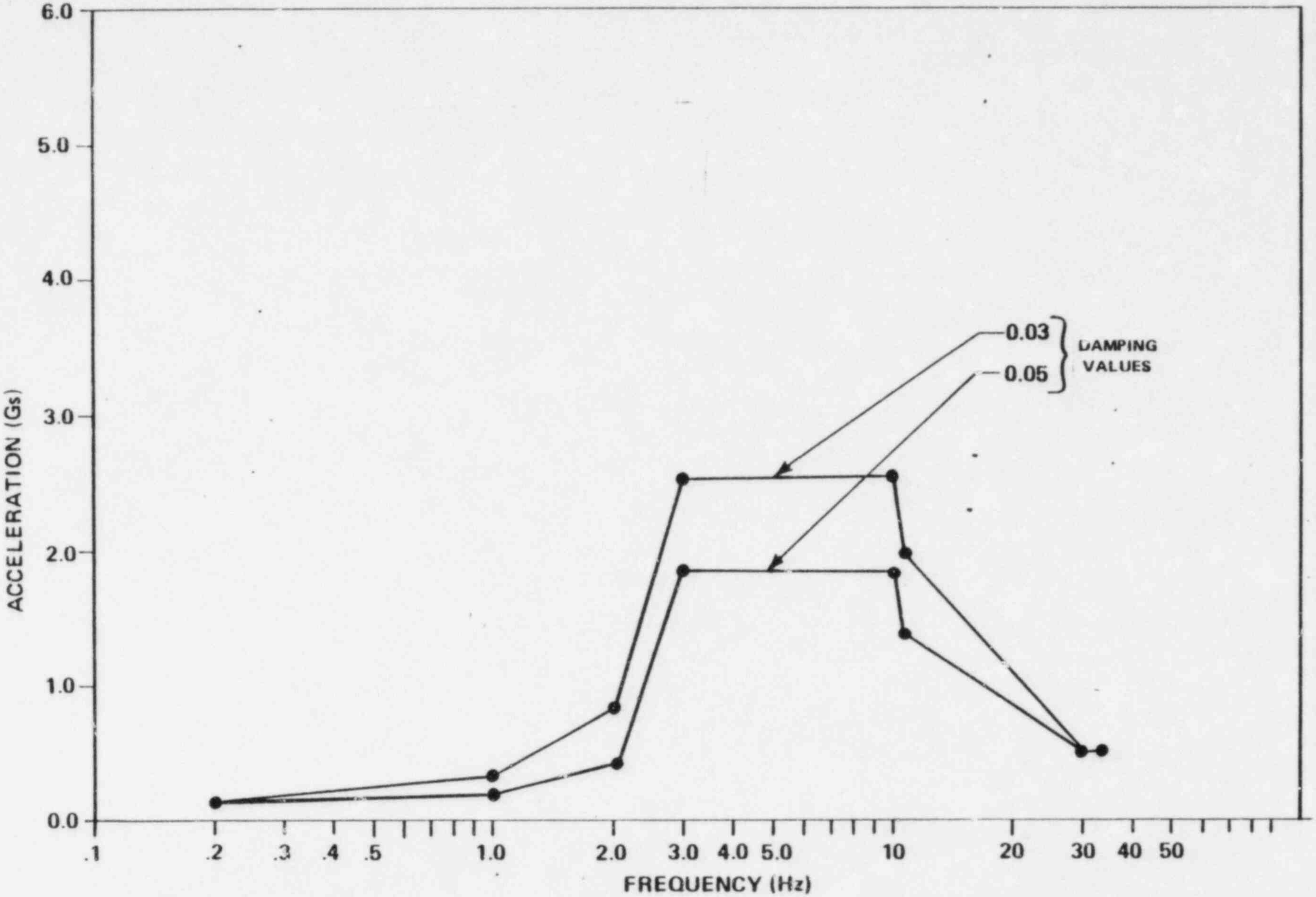


FIGURE 3.20 HNP-2 SSE SPECTRUM PEAK ENVELOPE 3- AND 5- PERCENT DAMPING (VERTICAL)

9389-2

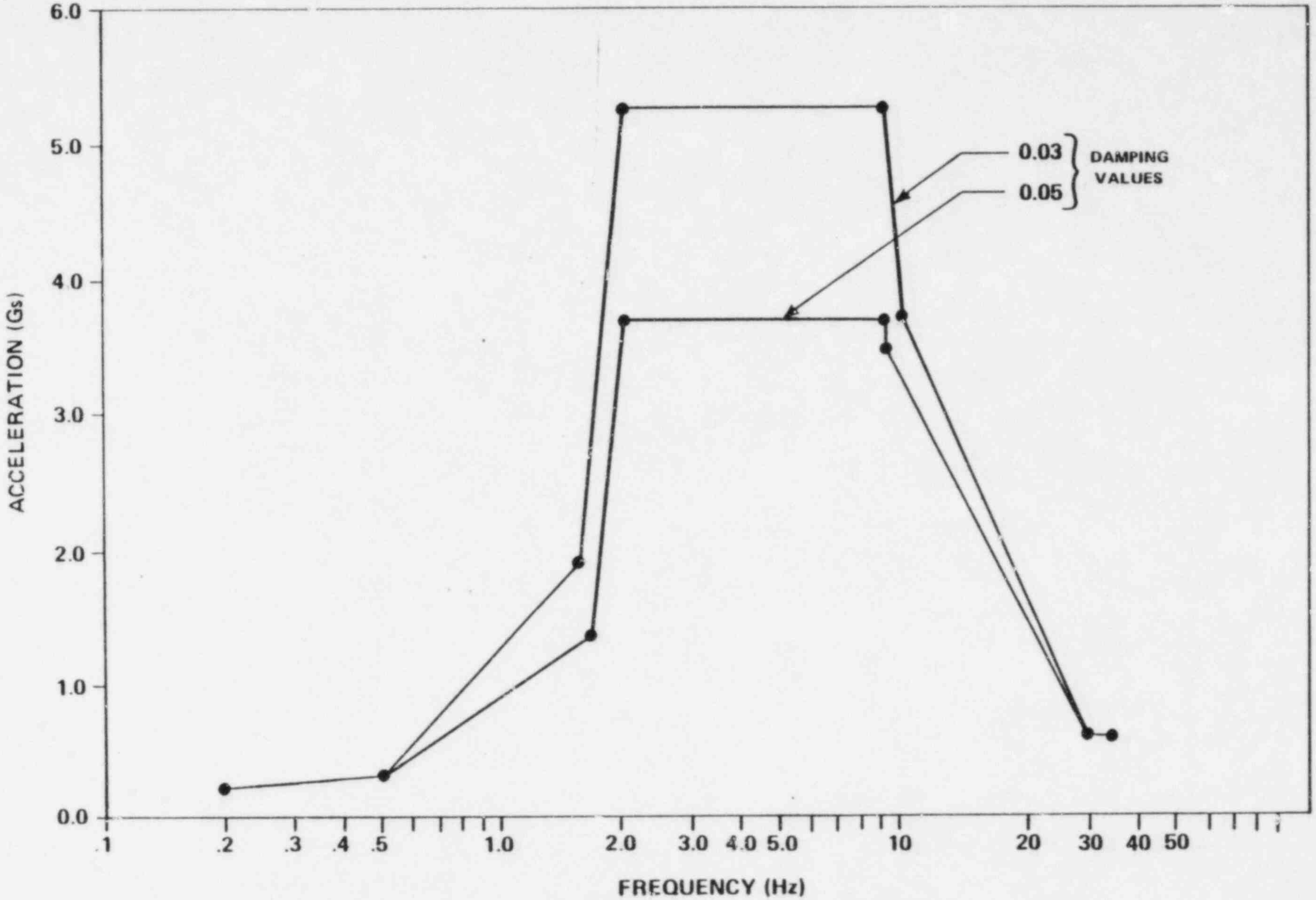


FIGURE 3.21 HNP-2 SSE SPECTRUM PEAK ENVELOPE 3- AND 5- PERCENT DAMPING (HORIZONTAL)

9389-2

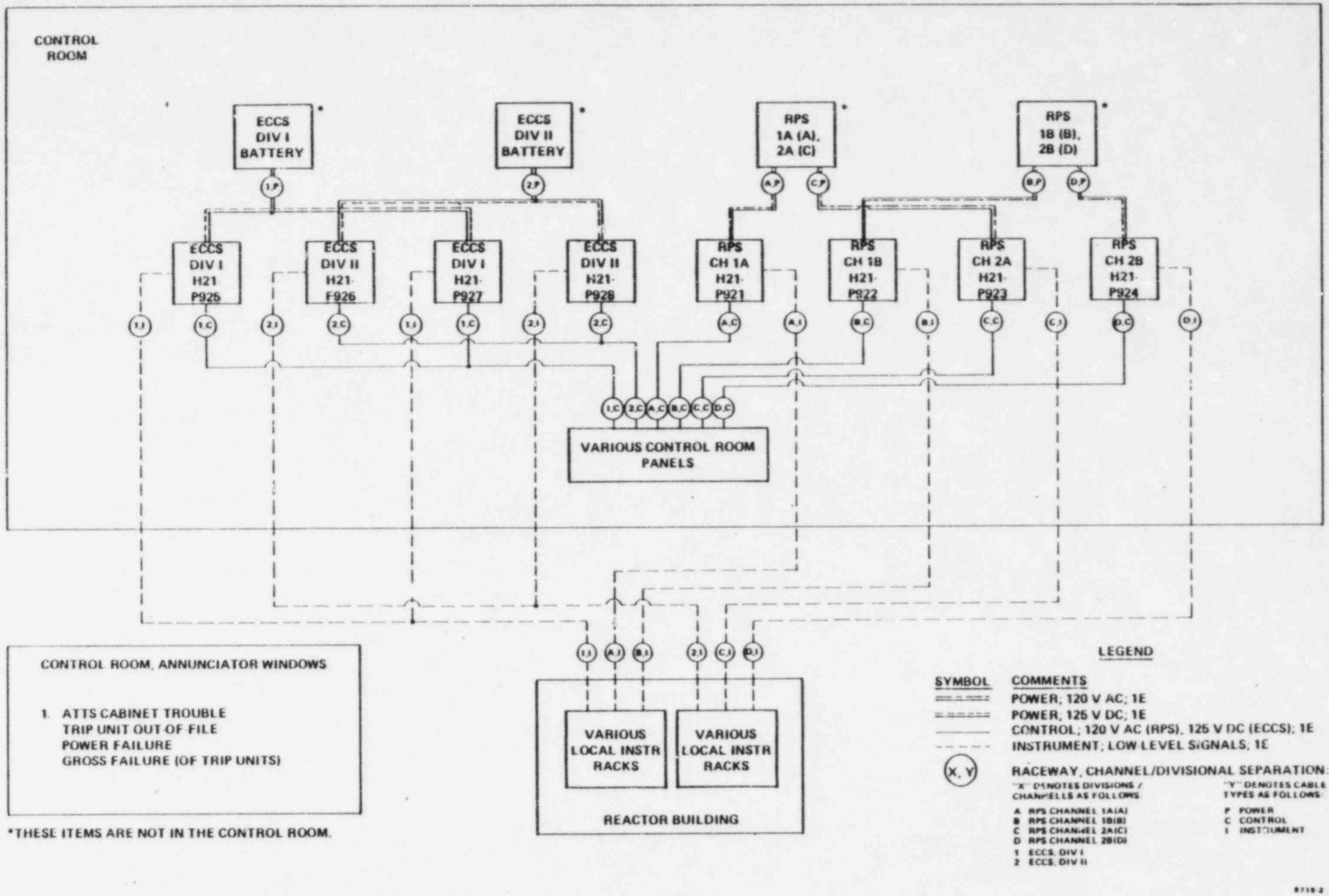


FIGURE 3.22 BASIC CABLE ARRANGEMENT AND SEPARATION LAYOUT FOR PLANT HATCH ATTS

3B. BASES FOR TECHNICAL SPECIFICATIONS REVISIONS FOR ATTS EQUIPMENT

With the incorporation of ATTS into the HNP-2 design, two types of Technical Specifications revisions are desirable:

- Nomenclature changes
- Modifications to the surveillance frequency of ATTS equipment.

The bases for the Technical Specifications changes over and above these ATTS changes are included in Section IV.

3B.1 The bases for these changes are as follows:

3B.1.a Nomenclature Changes to the Technical Specifications

The installation of ATTS replaces mechanical switches with transmitters. In the past, the Technical Specifications identified the switches as such under the instrument description. Therefore, the Technical Specifications require a revision to reflect this change.

3B.1.b Modifications of the Surveillance Frequency

As evidenced by NEDO-21617-A, the NRC previously approved the following surveillance frequencies for ATTS equipment:

- Once per shift for channel check
- Once per month for channel functional test
- Once per operating cycle for channel calibration.

The revisions contained in the enclosed proposed changes to the Technical Specifications reflect the above surveillance frequencies.

Additional bases for the surveillance frequency revisions are contained within Chapter 6 of NEDE-22154-1 (Appendix 2).

3B.2 The proposed revisions illustrated in this submittal are the revisions that will be incorporated into the Plant Hatch-Unit 2 system during the next refueling outage.

3C SUMMARY

From the preceding discussions, it may be concluded that the proposed incorporation of the ATTS equipment into the Plant Hatch design and the proposed nomenclature and surveillance requirement changes to the Technical Specifications do not introduce an unreviewed safety question or a significant hazards consideration. This proposed change takes advantage of the sensor improvements and the decreased drift which was demonstrated for this new equipment.

IV. PROPOSED PLANT MODIFICATIONS TO BE INCORPORATED INTO THE ATTS DESIGN

4A. INTRODUCTION

Section IV provides the bases for the proposed Technical Specifications revisions, in addition to an overview of the evaluation performed to conclude that the proposed modifications constitute neither an unreviewed safety question, nor a significant hazards consideration.

Within this section three values are discussed: the analytical limit, the trip setpoint/allowable value, and the trip setpoint. The analytical limit is the value used in the plant safety analyses. The trip setpoint/allowable value was developed using the criteria of Regulatory Guide 1.105, taking into account instrument inaccuracies, and is the value that will be listed in the Technical Specifications. The actual trip setpoint will take into consideration instrument drift.

4B. SAFETY EVALUATION REPORT

4B.1 Reactor Water Low Low (Level 2) Trip Setpoint Modification

The proposed Technical Specifications trip setpoint/allowable value for the reactor vessel water level 2 signal is $\geq$ -55 in. Reactor vessel water level 2 is for the initiation of high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC), and the recirculation pump trip.

Using Appendix K models, the Final Safety Analysis Report (FSAR) emergency core cooling system (ECCS) analysis was performed using a nominal analytical limit of -38 in. for level 2 HPCI actuation. However, the level 2 analytical limit is not a significant parameter in the Appendix K calculations. Therefore, as explained below, the ECCS calculations are insensitive to the variation in HPCI actuation water level so that a lower water level for level 2 has no significant effect on the ECCS system performance.

Sensitivity studies of the effect the level 2 analytical limit has on the ECCS analysis have shown that level 2 may be lowered all the way to the level 1 analytical limit (-152.5 in.) without affecting the results of that analysis. However, it has not been recommended that the level 2 analytical limit be lowered to -152.5 in. The proposed analytical limit of -58 in. was selected to provide the best flexibility and protective margin for the plant.

For small breaks, the limiting single failure is the HPCI failure. Even if HPCI is assumed operable, the HPCI actuation at -58 in. rather than -38 in. has an insignificant effect on core heatup.

For the proposed change in the level 2 analytical limit to -58 in., the MAPLHGR[a] limit will not be changed. The requirements of 10 CFR 100 will still be met because:

- Reactor power level or inventory is not changed.
- Engineering standards are not changed.
- The probability of radioactive release will not be increased.

The trip setpoint/allowable value of $\geq$ -55 in. was developed from the newly established analytical limit using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into consideration instrument drift.

Therefore, this modification constitutes neither an unreviewed safety question, nor a significant hazards consideration. No limiting conditions of operation (LCOs) or surveillance frequencies are changed by this modification. Appendix 1 (page A1-2) provides the results of the significant hazards review.

a. Maximum average planar linear heat-generation rate.

4B.2 Deletion of High Drywell Pressure Signal for Residual Heat Removal (Shutdown Cooling Mode), RPV Head Spray Valves, and Reactor Water Cleanup System Isolation

High Drywell pressure has been used as a signal to isolate reactor water cleanup (RWCU) and the shutdown cooling mode of RHR. Small steam leaks in the drywell can cause a high drywell pressure signal which would prohibit an acceptable normal shutdown procedure by preventing operation of the RHR and RWCU systems during the shutdown cooling mode. To resolve this operational concern, the high drywell pressure signal would be deleted from the isolation logic for the RHR shutdown cooling suction and discharge valves, as well as the reactor pressure vessel (RPV) head spray isolation valves and RWCU isolation valves.

The use of high drywell pressure as an isolation signal has little effect in preventing coolant losses due to an RHR or RWCU pipe break inside the drywell since the inboard isolation valves are located as close as possible to the drywell wall. Such pipe breaks do not present a site boundary dose problem since the leaked fluid and associated radioactivity are completely retained within the primary containment boundary. The high drywell pressure signal for the RPV head spray isolation valves will also be deleted. Since the RPV head spray valves are used as part of the shutdown cooling procedures, this change is consistent with above mentioned proposed RHR (shutdown cooling mode) System Modification.

This change does not affect the Appendix K calculation results presented in the FSAR. The requirements of 10 CFR 100 will still be met. This modification, therefore, constitutes neither an unreviewed safety question, nor a significant hazards consideration. Appendix 1 (page A1-3) provides the results of the significant hazards review.

This modification has been implemented on other boiling water reactor (BWR)/4s.

4B.3 Lowered Water Level Trip Setpoint for Isolation of Reactor Water Cleanup System and Secondary Containment, and Starting of Standby Gas Treatment System (SGTS)

Reactor scram from normal power levels (above 50 percent of rated) usually results in a reactor vessel water level transient due to void collapse that causes isolation of the RWCU system at reactor water level 3. The result is typically the dropping of the cleanup filter cake, added radwaste processing, loss of ability to remove water from the reactor vessel immediately after scram, and other undesirable operational problems. These results adversely affect plant availability and operability. By lowering the isolation setpoint to reactor water level 2, these problems may be resolved without any adverse safety impact. The lowering of the level trip for isolation of RWCU from reactor water level 3 to reactor water level 2 will not have any adverse effect on plant transient and accident analysis. For any reactor pressure coolant boundary line breaks inside the primary containment, the LOCA design basis accident (DBA) analysis shows that the ECCS is capable of mitigating all break sizes including and up to the recirculation line break. For a RWCU line break outside the primary containment, the break detection is provided by the high differential flow, area high temperature, or area high ventilation differential temperature rather than by water level variation.

By lowering the SGTS actuation and secondary containment isolation from reactor water level 3 to reactor water level 2, a potential for spurious trips is reduced. The ECCS analysis design basis assumes that the SGTS will initiate at the same time as the ECCS which initiates at reactor water level 2.

This modification has been implemented on other BWR/4s. The requirements of 10 CFR 100 are still met. These changes to the plant constitute neither an unreviewed safety question, nor a significant hazards consideration. Appendix 1 (page A1-4) provides the results of the significant hazards review.

4B.4 Deletion of Ambient Temperature Loops in Leak Detection System

Typically, the leak detection system uses ambient and differential temperatures to detect the small high-temperature leaks. In the earlier design, the ambient temperature trip was provided by an independent temperature element and trip device, and the differential temperature trip was provided by two independent temperature elements and a ΔT trip device as shown on figure 3.1. By using the ATTS, the ambient temperature trip may be obtained from one leg of the differential temperature trip shown on figure 3.2. In figures 3.1 and 3.2, it can be seen that K1 will provide the high ambient temperature trip, and K2 will provide the differential temperature trip. With this arrangement, the sensitivity of leak detection may be changed slightly, dependent on heating, ventilation, and air-conditioning design; but it will not defeat the intended function of the system. This arrangement is suitable for the small rooms containing leak detection temperature monitoring as part of the isolation logic, because only large rooms, such as the turbine building, need the spatial location of sensors to adequately protect the room against leaks. This scheme will allow the deletion of several unnecessary temperature loops in the RWCU system.

The RWCU temperature and differential temperature sensors sense the temperature in the two pump rooms and the heat exchanger room. Each room has a redundant set of temperature instrumentation that provides input to the RWCU isolation logic. By using the hot leg of the differential temperature sensor for the high ambient trip as shown on figure 3.2, several devices may be deleted without any loss of protective function.

For these modifications, single-failure criteria will be maintained.

The following six temperature loops are proposed for deletion:

MPL No.	
Sensor	Switch
G31-N016A	G31-N600A
G31-N016B	G31-N600B
G31-N016C	G31-N600C
G31-N016D	G31-N600D
G31-N016E	G31-N600E
G31-N016F	G31-N600F

The proposed Technical Specifications revisions will reference the trip unit loop from which the ambient temperature trip is taken in place of the existing ambient temperature trip instrument. Included in these proposed changes are new surveillance frequencies which correspond with the surveillance requirements of the ATTS. This modification does not constitute an unreviewed safety question or a significant hazards consideration. No LCO requirements will change as a result of this modification. Appendix 1 (page A1-5) provides the results of the significant hazards review.

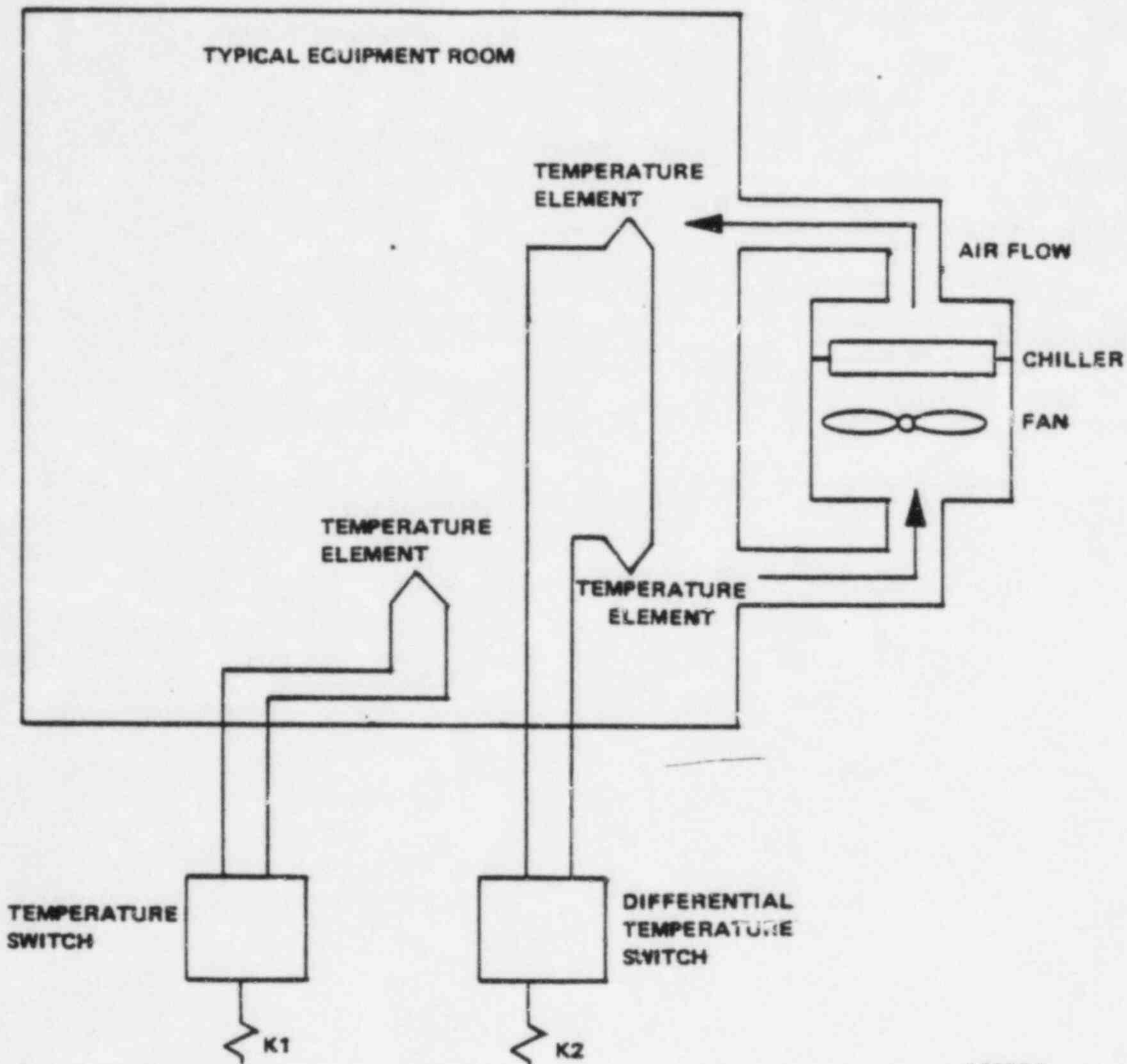


FIGURE 4.1 EXISTING TYPICAL AMBIENT AND DIFFERENTIAL TEMPERATURE SENSORS.

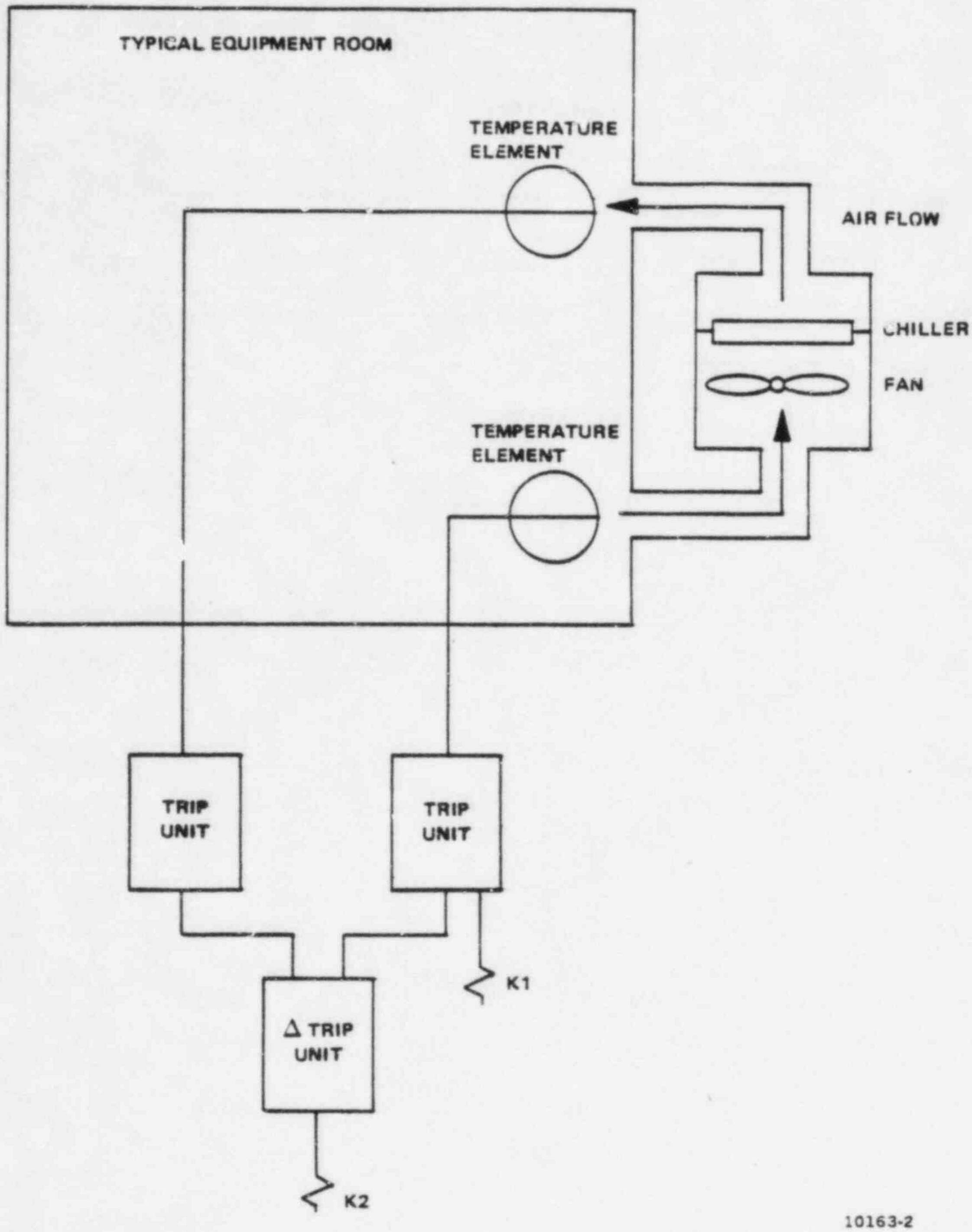


FIGURE 4.2 TYPICAL ATTS AMBIENT AND DIFFERENTIAL TEMPERATURE SENSORS

4B.5 Deletion of Drywell Pressure Sensors E11-N011A, B, C, D

The original design of Plant Hatch has the high drywell pressure signals for the ECCS coming from eight sensing devices. For example, E11-N011A, B, C, D (existing MPL numbers) provide signals to RHR, core spray, and HPCI; E11-N010A, B, C, D (existing MPL numbers) provide signals to ADS. This configuration is inconsistent with the inputs for the reactor water levels 1 and 2 trips which are provided by only four sensing devices, namely B21-N031A, B, C, D (existing MPL numbers). To make drywell pressure sensor configuration consistent with that for the water levels 1 and 2 sensors, drywell pressure sensors E11-N010A, B, C, D may be used to provide signals for all four systems of the ECCS and still maintain single-failure criteria. Plant safety margin is not being reduced since the level of redundancy to serve a trip function is maintained.

This change deletes instruments E11-N011A, B, C, D and transfers their associated trip function to instruments E11-N010A, B, C, D. Since these instruments (E11-N010A, B, C, D) are being incorporated into the ATTS modification, the instrument number was changed to E11-N694A, B, C, D.

It is proposed that the surveillance frequencies be modified to those of instruments E11-N010A, B, C, D. It should be noted that the surveillance frequencies of both E11-N010A, B, C, D, and E11-N011A, B, C, D are the same in the existing Unit 2 Technical Specifications. As a result of the above discussion, this modification represents neither an unreviewed safety question, nor a significant hazards consideration. Appendix 1 (page A1-6) provides the results of the significant hazards review.

4B.6 Trip Setpoint/Allowable Value Setpoint Modifications

4B.6.1 The instruments to be incorporated into the ATTS possess less drift and greater accuracy than the existing instruments in use at Plant Hatch. Therefore, new calculations were performed to determine the setpoint value for each instrument. The setpoint calculations were made using the criteria of Regulatory Guide 1.105. The Plant Hatch analytical limits were used (where applicable) to develop the allowable values and trip setpoints. Unless identified in the text, the analytical limits used to develop these setpoints are the values used in the design basis of Plant Hatch. The values that are proposed to be inserted into the Technical Specifications are the calculated allowable values. The Technical Specifications also have a column to indicate the trip setpoint. To avoid inconsistency in the specifications, this table reflects that the trip setpoint is more conservative than the allowable value. The setpoints used at Plant Hatch will take into consideration instrument drift and will be developed from the allowable values. The proposed Technical Specifications revisions include modification of the trip setpoints/allowable values for the following instruments:

Reactor Protection System (RPS) Trip Function	Trip Unit MPL No.
1. Main steamline flow - high	B21-N686A,B,C,D B21-N687A,B,C,D B21-N688A,B,C,D B21-N689A,B,C,D
2. Main steamline tunnel temperature - high	B21-N623A,B,C,D B21-N624A,B,C,D B21-N625A,B,C,D B21-N626A,B,C,D
3. Reactor vessel steam dome pressure - high	B21-N678A,B,C,D
4. Reactor vessel water level - level 3	B21-N680A,B,C,D
5. Reactor vessel steam dome pressure - low	B31-N679A,D
6. Reactor vessel water level - level 1	B21-N681A,B,C,D
7. Drywell pressure - high	C71-N650A,B,C,D
8. RWCU room ambient temperature - high	G31-N662A,D,E,H,J,M
RWCU room differential temperature - high	G31-N663A,D,E,H,J,M G31-N661A,D,E,H,J,M

ECCS Trip Function	Trip Unit MPL No.
1. Reactor vessel water level - level 1	B21-N691A,B,C,D
2. Reactor vessel water level - level 8	B21-N693B,D
3. Reactor shroud water level - level 0	B21-N685A,B
4. Reactor vessel pressure - low	B21-N690A,B,C,D
5. Reactor vessel pressure - low	B21-N690E,F B21-N641B,C
6. Reactor vessel water level - level 3	B21-N695A,B
7. Drywell pressure - high	E11-N694A,B,C,D
8. RHR pump discharge pressure - high	E11-N655A,B,C,D E11-N656A,B,C,D
9. Core spray pump discharge pressure - high	E21-N655A,B E21-N652A,B
10. HPCI steam supply pressure - low	E41-N658A,B,C,D
11. HPCI steamline differential pressure - high	E41-N657A,B E41-N660A,B
12. HPCI turbine exhaust diaphragm pressure - high	E41-N655A,B,C,D
13. Suppression chamber water level - high	E41-N662B,D
14. Emergency area cooler temperature - high	E41-N670A,B
15. HPCI pipe penetration room ambient temperature - high	E41-N671A,B
16. RCIC steamline pressure - low	E51-N658A,B,C,D
17. RCIC turbine exhaust diaphragm pressure - high	E51-N685A,B,C,D
18. RCIC steamline differential pressure - high	E51-N660A,B E51-N657A,B

ECCS Trip Function	Trip Unit MPL No.
19. Suppression chamber ambient temperature - high	E51-N666A,B,C,D
Suppression chamber differential temperature - high	E51-N664A,B,C,D E51-N665A,B,C,D E51-N663A,B,C,D
20. Emergency area cooler temperature - high	E51-N661A,B

4B.6.2 The bases for these proposed changes are as follows:

4B.6.2.1 Setpoint Bases for Trip Functions Assigned to RPS Cabinets

1. Main steamline flow - high (B21-N686A,B,C,D; B21-N687A,B,C,D; B21-N688A,B,C,D; B21-N689A,B,C,D)

   The analytical limit of 120 psid corresponds to 140 percent of rated flow. The 140 percent of rated flow is used as an input to the high-energy line break (HELB) calculations. The trip setpoint/allowable value of ≤ 125 psid was developed using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into account instrument drift.

2. Main steam line tunnel temperature - high (B21-N623A,B,C,D; B21-N624A,B,C,D; B21-N625A,B,C,D; B21-N626A,B,C,D)

   The value of 200°F was designed to detect a small steam line break. In addition, it provides early isolation of the main steam line to meet 10 CFR 100 requirements. The trip setpoint/allowable value of ≤ 194°F was developed using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into account instrument drift.

3. Reactor vessel steam dome pressure - high (B21-N678A,B,C,D)

   The analytical limit of 1071 psig is the value used in the plant transient analysis and is documented in the Plant Hatch-Unit 2 FSAR. The intent of this reactor scram function is to provide the nuclear system process barrier overpressure protection, because a rupture to the process barrier may result in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by collapsing voids, thus adding reactivity. A reactor scram will quickly reduce the neutron flux, counteracting the pressure increase. The trip setpoint/allowable value of ≤ 1054 psig was developed using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into consideration instrument drift.

4. Reactor vessel water level - level 3 (B21-N680A,B,C,D)

   Reactor vessel water level 3 is for isolation of primary containment, initiation of reactor scram, and for closure of RHR shutdown cooling isolation valves. The analytical limit of 7.5 in. was used to determine the trip setpoint and allowable value. A trip setpoint/allowable value of ≥ 8.5 in. was developed using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into consideration instrument drift.

5. Reactor vessel steam dome pressure - low (B31-N679A,D)

The trip function provides the low-pressure permissive signals to the RHR shutdown cooling mode loops. The analytical limit of 162 psig was determined based on plant-specific RHR system piping design and layout. The trip setpoint/allowable value of ≤ 145 psig was developed using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into account instrument drift.

6. Reactor vessel water level - level 1(B21-N681A,B,C,D)

The analytical limit for this application is -152.5 in. The trip setpoint/allowable value of ≥ -121.5 in. was developed using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into consideration instrument drift.

7. Drywell pressure - high (C71-N650A,B,C,D)

The value used as the analytical limit for this RPS scram function is 2.0 psig. The trip setpoint/allowable value of ≤ 1.85 psig was developed using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into consideration instrument drift.

8. RWCU room ambient temperature - high (G31-N662A,D,E,H,J,M)
RWCU area differential temperature - high (G31-N663A,D,E,H,J,M;
G31-N661A,D,E,H,J,M)

The leak detection system uses ambient and differential temperatures to detect small high-temperature leaks. The present design for this system uses a value of 130°F for the RWCU room outlet ambient temperature and 75°F for the RWCU area differential temperature.

The trip setpoint/allowable values of ≤ 124°F for the RWCU ambient temperature trip and ≤ 67°F for the RWCU differential temperature trip were developed using the criteria of Regulatory Guide 1.105.

4B.6.2.2 Setpoint Basis for Trip Functions Assigned to ECCS Cabinets

1. Reactor vessel water level - level 1 (B21-N691A,B,C,D)

The analytical limit of -152.5 in. is the value in the FSAR ECCS analysis for automatic depressurization system (ADS), low-pressure coolant injection (LPCI), and core spray initiation. The trip setpoint/allowable value of ≥ -121.5 in. was developed using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into consideration instrument drift.

2. Reactor vessel water level - level 8 (B21-N693B,D)

After HPCI has activated, this trip function prevents the water level in the reactor vessel from reaching the height of the main steam outlet. Its intended protective function is accomplished by tripping the HPCI steam turbine when the water level in the reactor vessel has reached the level 8 setting. The trip function is to protect the HPCI turbine system from potential damage. The analytical limit for this function is 59.5 in. The trip setpoint/allowable value of $\leq$ 56.5 in. was developed using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into consideration instrument drift.

3. Reactor shroud water level - level 0 (B21-N685A,B)

This trip function acts as a bypass interlock to prevent containment spray initiation during LPCI operation when the water level of the reactor shroud is low. Under post DBA conditions, water pumped from the suppression chamber through the RHR heat exchanger may be diverted for containment spray to remove the energy from the drywell. If the water level in the vessel is low, this function prevents the water from being diverted. An analytical value of -211 in. is used to develop the allowable value and trip setpoint. The trip setpoint/allowable value of $\geq$-207 in. was developed using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into consideration instrument drift.

4. Reactor vessel pressure - low (B21-N690A,B,C,D)

These instrument channels monitor the reactor vessel pressure to provide the permissive signal for core spray and the injection permissive signal for RHR (LPCI). The analytical limit for this pressure permissive function has a range from 400 to 500 psig. The criteria for this setpoint are that it should be low enough to assure no overpressurization of the low-pressure ECCS and yet high enough so as not to delay the flow injection of the low-pressure system. The trip setpoint/allowable value of $\geq$ 422 psig was developed using the criteria of Regulatory Guide 1.105 and an analytical limit of 400 psig. Since the important parameter is to ensure that RHR and core spray are operating as the pressure drops, a sign change of $\geq$ is required. The existing requirement that the setpoint be $\leq$500 psig is also retained in the Technical Specifications. The designated trip setpoint for the plant will take into consideration instrument drift. A new post-accident surveillance recorder (2B21-R623A,B) will take its signal from the B21-N690A,D transmitter.

5. Reactor vessel pressure - low (B21-N690E,F; B21-N641B,C)

An analytical limit of 300 psig is used in the Appendix K ECCS calculations. The intent of this trip function is to provide a permissive for closure of the recirculation discharge valves. The present Hatch setpoints of 309 psig and 314 psig are above the analytical limit of 300 psig. Plant Hatch, therefore, has operated within its design basis. However, a sign change is being proposed to correct the Technical Specifications deficiency. The trip setpoint/allowable value of $\geq$ 325 psig was developed using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into consideration instrument drift.

6. Reactor vessel water level - level 3 (B21-N695A,B)

Reactor vessel level 3 provides an ADS permissive. The analytical limit of 7.5 in. was used in the FSAR ECCS analysis. The trip setpoint/allowable value of $\geq$ 8.5 in. was developed from the analytical limit using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into consideration instrument drift.

7. Drywell pressure - high (E11-N694A,B,C,D)

The value used for the analytical limit for the ECCS function is 2.0 psig. The trip setpoint/allowable value of $\leq$ 1.85 psig was developed using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into consideration instrument drift.

8. RHR pump discharge pressure - high (E11-N655A,B,C,D; E11-N656A,B,C,D)

These instruments provide the RHR pump high-pressure permissive signal for ADS. The ADS interlock to sense if low-pressure ECCS pumps are running is not an direct input to the ECCS calculations; however, correct operation of the interlock is an analytical assumption. To avoid any false indication during an accident, the trip setpoint should be above the suction relief valve setpoint of approximately 100 psig. The range of the analytical limit is between 100 and 150 psig.

Selecting 100 psig at the lower bound for the analytical limit, the trip setpoint/allowable value of $\geq$ 105 psig was developed using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into consideration instrument drift.

9. Core spray pump discharge pressure - high (E21-N655A,B; E21-N652A,B)

The core spray pump discharge pressure - high trip is part of the ADS interlock function. This trip is not used as input to any transient or safety analysis; however, correct operation of the trip is assumed. To avoid a false indication during an accident, this ADS permissive function should be set above the suction relief valve setpoint. To assure this, the value from which to determine the trip setpoint and allowable value has been conservatively selected to be at $\geq$ 125 psig. The trip setpoint/allowable value of $\geq$ 130 psig was developed using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into consideration instrument drift.

10. HPCI steam supply pressure - low (E41-N658A,B,C,D)

This trip function is intended to prevent HPCI turbine stall and possible HPCI turbine damage. The signal of low HPCI steamline pressure will close the HPCI isolation valve and trip the HPCI turbine. Even though the HPCI design specification and Technical Specifications only require HPCI operability down to 150 psig, it is desirable to operate HPCI to as low a steam pressure as allowable. It has been determined that the use of an allowable value of $\geq$ 100 psig will prevent the possibility of damage to the HPCI equipment due to turbine stall. The designated trip setpoint for the plant will take into consideration instrument drift.

11. HPCI steamline differential pressure - high (E41-N657A,B; E41-N660A,B)

The purpose of this instrumentation is to detect HPCI steamline breaks and to isolate the HPCI system to confine the resulting radioactivity release and limit the reactor inventory loss. The HELB analysis assumes that the HPCI turbine trips and, the system isolates at 300 percent of rated flow. However, the HELB analysis is used for guillotine breaks which have flows several times higher than 300 percent of rated flow. A conservative analysis shows the leakage detection instrumentation isolates in the 400 percent of rated flow range with less inventory loss and less peak qualification parameters than the inventory loss and the qualification parameters calculated in the extreme HELB analysis. Since operability problems are a concern with setpoints derived from an analytical limit of 300 percent (using Regulatory Guide 1.105 methodology for this function), it is proposed that the present Plant Hatch setpoint be maintained. This setpoint was proven to be acceptable from operability considerations. Using this setpoint, an allowable value of 307 percent of rated flow was selected taking into account instrument drift.

12. HPCI turbine exhaust diaphragm pressure - high (E41-N655A,B,C,D)

The trip setpoint for this instrumentation shall be low enough such that when the rupture disc blows, the transmitter/trip unit will activate immediately, causing isolation of the HPCI system, yet high enough such that atmospheric variations will not cause unnecessary HPCI isolation. This isolation minimizes steam releases to the secondary containment. Since this trip function is not directly considered in any safety or transient analysis performed for the FSAR, no analytical limit is available.

An allowable value of $\leq$ 20 psig was determined to be adequate in preventing turbine stall. Also, since high-pressure conditions only occur when the diaphragm is ruptured, an abnormally high-pressure condition at the turbine exhaust is detected always by these pressure switches with an allowable value of 20 psig. The trip used for the instruments at the plant will take into consideration instrument drift using the criteria of Regulatory Guide 1.105.

13. Suppression chamber water level - high (E41-N662B,D)

The purpose of this trip function is to transfer the HPCI suction from the plant condensate storage tank to the suppression chamber when the water in the suppression chamber rises above a predetermined level.

Although no analytical limit has been developed for this function, the requirements of Mark 1 Long-Term Program have been considered. The trip setpoint was developed to minimize undesirable trips. The allowable value of ≤ 33.2 in. was determined by adding the design drift allowance to the trip setpoint.

14. Emergency area cooler temperature - high (E41-N670A,B)

The leak detection system uses ambient temperature elements to detect any small high-temperature leaks. This trip function provides a signal to trip the HPCI turbine and closes the HPCI isolation valves. The present design for this system uses a value of 175°F to isolate the HPCI equipment. The trip setpoint/ allowable value of ≤ 169°F was developed using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into consideration instrument drift.

15. HPCI pipe penetration room ambient temperature - high (E41-N671A,B)

The leak detection system uses ambient temperature elements to detect any small high-temperature leaks. This trip function provides a signal to trip the HPCI turbine and close the HPCI isolation valves. The present design for this system uses a value of 175°F to isolate HPCI equipment. The trip setpoint/ allowable value of ≤ 169°F was developed using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into consideration instrument drift. The identification of this trip function was modified to clarify the location of the instrument. However, the general area where the equipment is located will not change.

16. RCIC steam supply pressure - low (E51-N658A,B,C,D)

Per the Unit 1 Technical Specifications, the RCIC system is only required to operate down to 150 psig, but the system is capable of operating at lower pressures. The intent of this trip function is to prevent RCIC turbine stall at low reactor pressure. An allowable value of ≥ 60 psig was determined to be adequate in preventing turbine stall. The trip setpoint used for the instruments at the plant will take into consideration instrument drift using the criteria of Regulatory Guide 1.105.

17. RCIC turbine exhaust diaphragm pressure - high (E51-N685A,B,C,D)

The bases for the trip setpoint/allowable value of 20 psig are identical to that for the HPCI turbine exhaust diaphragm pressure - high trip provided in item 12.

18. RCIC steamline differential pressure - high (E51-N660A,B; E51-N657A,B)

For reasons similar to those for the HPCI steamline differential pressure - high trip (provided in item 11), it is proposed that the present Plant Hatch setpoint be maintained. This setpoint was proven to be acceptable from operability considerations. Using this setpoint, an allowable value of 312 percent of rated flow was calculated taking into account instrument drift.

19. Suppression chamber ambient temperature - high (E51-N666A,B,C,D)
Suppression chamber differential temperature - high (E51-N665A, B,C,D; E51-N663A,B,C,D; E51-N664A,B,C,D)

The leak detection system uses ambient and differential temperatures to detect small high-temperature leaks. The present design for this system utilizes a trip value of 175°F for the suppression chamber ambient temperature - high and 50°F for the suppression chamber differential temperature - high. The trip setpoint/allowable values of ≤ 169°F for the ambient temperature and ≤ 42°F for the differential temperature were developed using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into consideration instrument drift.

20. RCIC emergency area cooler temperature - high (E51-N661A,B)

The leak detection system uses ambient temperature elements to detect any small high-temperature leaks and provides a signal to trip the RCIC turbine and close the RCIC isolation valves. The present design uses a trip value of 175°F for providing the intended RCIC isolation function. The trip setpoint/allowable value of ≤ 169°F was developed using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into consideration instrument drift.

4B.6.3 Conclusion

For the trip functions identified in this section, the analytical limits that are presented are such that neither transient nor safety analysis results documented in the FSAR are adversely affected. The uncertainties associated with instrument accuracy, calibration, and drift are considered in the setpoint determination. Therefore, it was concluded that the proposed instrumentation setpoint changes do not reduce the margin of safety of the current Technical Specifications or change the FSAR setpoint bases and, therefore, constitute neither an 10 CFR 50.59 unreviewed safety question, nor a significant hazards consideration as described in 10 CFR 50.92. Appendix 1 (page A1-) provides the results of the significant hazards review.

4B.7 Reactor Vessel Water Level - High (Level 8) Trip Instrumentation Modifications

After HPCI and RCIC have activated, this trip function prevents the water level in the reactor vessel from reaching the height of the main steam outlet. Its intended protective function is accomplished by tripping the HPCI steam turbine enclosing the HPCI and RCIC steam supply valves when the water level in the reactor vessel reaches the level 8 setting. The trip function is to protect the HPCI and RCIC steam turbine system from potential damage.

This trip function is currently assigned to B21-N017A,B,C,D which controls the RPS reactor vessel water level 3 instrumentation. To separate the RPS and ECCS functions, the ATTS design assigns the reactor vessel water level 8 trip function to ECCS instrumentation. Since the functions of the level 8 trip remain the same, this modification constitutes neither an unreviewed safety question, nor a significant safety hazards consideration. Appendix 1 (page A1-8) provides the results of the significant hazards review.

The analytical limit for this function is 59.5 in. The trip setpoint/allowable value of $\leq$ 56.5 in. was developed using the criteria of Regulatory Guide 1.105. The designated trip setpoint for the plant will take into consideration setpoint drift.

4B.8 Elimination of the Reactor Pressure Permissive to the Bypass of the MSIV Closure Signal Due to Low Condenser Vacuum

It is proposed to delete the reactor steam dome pressure permissive which prevents the group 1 isolation valves signal from being bypassed on a low condenser vacuum isolation.

The manual bypass is provided to facilitate the following operations:

A. The bypass allows cold shutdown testing of the main steamline isolation logic and allows stroking the MSIVs open and closed for maintenance even though there is no condenser vacuum.

B. The bypass allows the MSIVs to be opened so seal steam and ejector steam can be available at the turbine and condenser, thereby allowing restart of the reactor from a hot pressurized condition. Attempting to establish condenser vacuum without seal steam from the hot condition by the mechanical vacuum pump may damage the turbine shaft seals.

Thus, the manual bypass of the MSIV closure is performed only when the reactor is not operating at full power. In addition, the manual bypass, which is annunciated in the control room, has the following three permissive conditions:

A. When the keylocked manual switches located on the back cabinets housing the MSIV logic are in the bypass position. One keylocked switch is in each isolation logic string.

B. When the turbine stop valves are less than 90 percent full open, the four independent contacts of the turbine stop valve position switch sensor relay of the RPS will trip.

C. When the reactor is below 1045 psig.

Of these three permissives on the manual bypass of the MSIV closure, only the reactor pressure permissive (item C above) does not have a safety function. This is also the only permissive that is proposed for deletion. With the septoint at 1045 psig, which is the same setpoint for the high reactor pressure scram, the manual bypass can be performed at any operating reactor pressure provided the other two permissives are cleared. When the manual bypass is activated, plant protection is provided by those two other permissives by the normal scram and isolation signals, e.g., turbine stop valve position, low reactor water level, high steam flow, high steam tunnel temperature, and turbine building temperature, and by the annunciators in the control room. Eliminating the reactor pressure permissive does not affect the existing plant protection in any way. Therefore, elimination of the reactor pressure permissive on the manual bypass of the MSIV closure (item C above) does not constitute an unreviewed safety question or a significant hazards consideration. Appendix 1 (page A1-9) provides the results of the significant hazards review.

V. SUMMARY

This submittal provides an overview of the evaluation and the justification for the incorporation of the proposed modifications into the Plant Hatch design. Proposed Technical Specifications revisions are supplied to cover these modifications. None of these modifications constitute an unreviewed safety question or a significant hazards consideration; and the plant safety margin has not been reduced.

VI. PROPOSED TECHNICAL SPECIFICATIONS REVISIONS

The HNP-2 Technical Specifications (Appendix A to Operating License NPF-5) are proposed for revision as presented in this section. Table 6.1 provides the instructions for incorporating the revision(s).

TABLE 6.1

INSTRUCTIONS FOR INCORPORATING TECHNICAL SPECIFICATIONS REVISIONS

If the Technical Specifications revisions are accepted as proposed, the HNP-2 Technical Specifications (Appendix A to Operating License NPF-5) should be incorporated as follows:

Item	Deletions (Page)	Insertions (Page)	Applicable Safety Evaluation Report (SER) Sections[a]
1	2-4	2-4	ATTS, 4B.6
2	3/4 3-2	3/4 3-2	ATTS
3	3/4 3-7	3/4 3-7	ATTS
4	3/4 3-11	3/4 3-11	ATTS, 4B.2, 4B.3
5	3/4 3-12	3/4 3-12	ATTS, 4B.2, 4B.3, 4B.4
6	3/4 3-13	3/4 3-13	Editorial, ATTS, 4B.4, 4B.5
7	3/4 3-14	3/4 3-14	Editorial, ATTS, 4B.3, 4B.4, 4B.5
8	3/4 3-15	3/4 3-15	Editorial, 4B.6, 4B.8
9	3/4 3-16	3/4 3-16	4B.3, 4B.6, 4B.1
10	3/4 3-17	3/4 3-17	Editorial, 4B.1, 4B.3, 4B.6
11	3/4 3-18	3/4 3-18	4B.6
12	3/4 3-19	3/4 3-19	4B.3
13	3/4 3-20	3/4 3-20	Editorial
14	3/4 3-21	3/4 3-21	ATTS, 4B.3, 4B.8
15	3/4 3-22	3/4 3-22	Editorial, ATTS, 4B.3
16	3/4 3-23	3/4 3-23	ATTS
17	3/4 3-26	3/4 3-26	ATTS, 4B.5
18	3/4 3-27	3/4 3-27	Editorial, ATTS, 4B.5
19	3/4 3-28	3/4 3-28	4B.6
20	3/4 3-29	3/4 3-29	4B.1, 4B.6
21	3/4 3-31	3/4 3-31	ATTS
22	3/4 3-32	3/4 3-32	ATTS
23	3/4 3-34	3/4 3-34	ATTS
24	3/4 3-35	3/4 3-35	4B.1
25	3/4 3-36	3/4 3-36	ATTS
26	3/4 3-54	3/4 3-54	ATTS
27	3/4 4-18	3/4 4-18	4B.6
28	3/4 6-41	3/4 6-41	4B.3
29	B 3/4 3-6	B 3/4 3-6	Editorial, ATTS, 4B.6

a. 1. ATTS refers to proposed revisions already presented in reference 1.
2. 4B.1 to 4B.8 refers to justifications presented in section 4B of this submittal.
3. Editorial refers to a typographical error. The proposed revision corrects the error; no additional justification is necessary.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High (2C51-K601 A,B,C,D,E,F,G,H)	≤ 120/125 divisions of full scale	≤ 120/125 divisions of full scale
2. Average Power Range Monitor: (2C51-K605 A,B,C,D,E,F)		
a. Neutron Flux-Upscale, 15%	≤ 15/125 divisions of full scale	≤ 20/125 divisions of full scale
b. Flow Referenced Simulated Thermal Power-Upscale	≤ (0.66 W + 51%), with a maximum ≤ 113.5% of RATED THERMAL POWER	≤ (0.66 W + 54%), with a maximum ≤ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux-Upscale, 118%	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High (2B21-N678 A,B,C,D)	≤ 1054 psig	≤ 1054 psig
4. Reactor Vessel Water Level - Low (Level 3) (2B21-N680 A,B,C,D)	≥ 8.5 inches above instrument zero*	≥ 8.5 inches above instrument zero*
5. Main Steam Line Isolation Valve - Closure (NA)	≤ 10% closed	≤ 10% closed
6. Main Steam Line Radiation - High (2D11-K603A,B,C,D)	≤ 3 x full power background	≤ 3 x full power background
7. Drywell Pressure - High (2C71-N650A,B,C,D)	≤ 1.85 psig	≤ 1.85 psig

*See Bases Figure B 3/4 3-1.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(a)	ACTION
1. Intermediate Range Monitors: (2C51-K601, A, B, C, D, E, F, G, H)			
a. Neutron Flux - High	2[(c)], 5[(b)]	3	1
	3, 4	2	2
b. Inoperative	2,5[(b)]	3	1
	3,4	2	2
2. Average Power Range Monitor: (2C51-K605 A, B, C, D, E, F)			
a. Neutron Flux - Upscale, 15%	2, 5	2	1
b. Flow Referenced Simulated Thermal Power - Upscale	1	2	3
c. Fixed Neutron Flux - Upscale, 118%	1	2	3
d. Inoperative	1, 2, 5	2	4
e. Downscale	1	2	3
f. LPRM	1, 2, 5	(d)	NA
3. Reactor Vessel Steam Dome Pressure - High (2B21-N678 A, B, C, D)	1, 2[(e)]	2(j, 2B21-N045 A, B, C, D)	5
4. Reactor Vessel Water Level - Low (Level 3) (2B21-N680 A, B, C, D)	1, 2	2(j, 2B21-N681 A, B, C, D)	5
5. Main Steam Line Isolation Valve - Closure (NA)	1[(f)]	4	3
6. Main Steam Line Radiation - High (2D11-K603 A, B, C, D)	1, 2[(e)]	2	6
7. Drywell Pressure - High (2C71-N650 A, B, C, D)	1, 2[(g)]	2	5

TABLE 4.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	D	S/U(b)(c)	R	2
	D	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor:				
a. Neutron Flux - Upscale, 15%	S	S/U(b)(c), W(d)	S/U(b), W(d)	2
	S	W	W	5
b. Flow Referenced Simulated Thermal Power - Upscale	S	S/U(b), W	W(e)(f), SA	1
c. Fixed Neutron Flux - Upscale, 118%	S	S/U(b), W	W(e), SA	1
d. Inoperative	NA	W	NA	1, 2, 5
e. Downscale	NA	W	NA	1
f. LPRM	D	NA	(g)	1, 2, 5
3. Reactor Vessel Steam Dome Pressure - High	S	M	R	1, 2
4. Reactor Vessel Water Level - Low (Level 3)	S	M	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	M	R(h)	1
6. Main Steam Line Radiation - High	D	W(i)	R(j)	1, 2
7. Drywell Pressure - High	S	M	R	1, 2
8. Scram Discharge Volume Water Level - High	NA	M	R(h)	1, 2, 5

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1. Low (Level 3) (2B21-N680 A, B, C, D)	2, 6, 10, 11, 12	2	1, 2, 3	20
2. Low-Low (Level 2) (2B21-N682 A, B, C, D)	5, #, *	2	1, 2, 3	20
3. Low-Low-Low (Level 1) (2B21-N681 A, B, C, D)	1	2	1, 2, 3	20
b. Drywell Pressure - High (2C71-N650 A, B, C, D)	2, 6, 7, 10, 12, #, *	2	1, 2, 3	20
c. Main Steam Line				
1. Radiation - High (2D11-K603 A, B, C, D)	1, 12, #, (d)	2	1, 2, 3	21
2. Pressure - Low (2B21-N015 A, B, C, D)	1	2	1	22
3. Flow - High (2B21-N686 A, B, C, D) (2B21-N687 A, B, C, D) (2B21-N688 A, B, C, D) (2B21-N689 A, B, C, D)	1, #	2/line	1, 2, 3	21
d. Main Steam Line Tunnel Temperature - High (2B21-N623 A, B, C, D) (2B21-N624 A, B, C, D) (2B21-N625 A, B, C, D) (2B21-N626 A, B, C, D)	1	2/line(e)	1, 2, 3	21
e. Condenser Vacuum - Low (2B21-N056 A, B, C, D)	1	2	1, 2,(f), 3(f)	23
f. Turbine Building Area Temperature - High (2U61-R001, 2U61-R002, 2U61-R003, 2U61-R004)	1	2(e)	1, 2, 3	21

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Exhaust Radiation - High (2D11-K609 A, B, C, D)	6, 10, 12, *	2	1,2,3,5 and**	24
b. Drywell Pressure - High (2C71-N650 A, B, C, D)	2, 6, 7, 10, 12, #, *	2	1, 2, 3	24
c. Reactor Vessel Water Level - Low Low (Level 2) (2B21-N682 A, B, C, D)	5, #, *	2	1, 2, 3	24
d. Refueling Floor Exhaust Radiation - High (2D11-K611 A, B, C, D)	6, 10, 12, #, *	2	1,2,3,5 and**	24
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High (2G31-N603 A, B)	5	1	1, 2, 3	25
b. Area Temperature - High (2G31-N662 A, D, E, H, J, M)	5	1	1, 2, 3	25
c. Area Ventilation Δ Temp. - High (2G31-N663 A, D, E, H, J, M; 2G31-N661 A, D, E, H, J, M; 2G31-N662 A, D, E, H, J, M)	5	1	1, 2, 3	25
d. SLCS Initiation (NA)	5(g)	NA	1, 2, 3	25
e. Reactor Vessel Water Level - Low Low (Level 2) (2B21-N682 A, B, C, D)	5, #, *	2	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Flow - High (2E41-N657 A,B)	3	1	1, 2, 3	26
b. HPCI Steam Supply Pressure - Low (2E41-N658 A,B,C,D)	3,8	2	1, 2, 3	26
c. HPCI Turbine Exhaust Diaphragm Pressure - High (2E41-N655 A,B,C,D)	3	2	1, 2, 3	26
d. HPCI Pipe Penetration Room Temperature - High (2E41-N671 A, B)	3	1	1, 2, 3	26
e. Suppression Pool Area Ambient Temperature-High (2E51-N666 C, D)	3	1	1, 2, 3	26
f. Suppression Pool Area Δ Temp.-High (2E51-N665 C, D; 2E51-N663 C, D; 2E51-N664 C, D)	3	1	1, 2, 3	26
g. Suppression Pool Area Temperature Timer Relays (2E41-M603 A, B)	3[(i)]	1	1, 2, 3	26
h. Emergency Area Cooler Temperature-High (2E41-N670 A, B)	3	1	1, 2, 3	26
i. Drywell Pressure-High (2E11-N694 C, D)	8	1	1, 2, 3	26
j. Logic Power Monitor (2E41-K1)	NA[(h)]	1	1, 2, 3	27

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow-High (2E51-N657 A,B)	4	1	1, 2, 3	26
b. RCIC Steam Supply Pressure - Low (2E51-N658 A, B, C, D)	4, 9	2	1, 2, 3	26
c. RCIC Turbine Exhaust Diaphragm Pressure - High (2E51-N685 A, B, C, D)	4	2	1, 2, 3	26
d. Emergency Area Cooler Temperature - High (2E51-N661 A, B)	4	1	1, 2, 3	26
e. Suppression Pool Area Ambient Temperature-High (2E51-N666 A, B)	4	1	1, 2, 3	26
f. Suppression Pool Area Δ T-High (2E51-N665 A, B; 2E51-N663 A,B; 2E51-N664 A,B)	4	1	1, 2, 3	26
g. Suppression Pool Area Temperature Timer Relays (2E51-M602 A, B)	4(i)	1	1, 2, 3	26
h. Drywell Pressure - High (2E11-N694 A, B)	9	1	1, 2, 3	26
i. Logic Power Monitor (2E51-K1)	NA(h)	1	1, 2, 3	27
6. SHUTDOWN COOLING SYSTEM ISOLATION				
a. Reactor Vessel Water Level-Low (Level 3)(2B21-N680 A, B, C, D)	6, 10, 11, 2 12	2	3, 4, 5	26
b. Reactor Steam Dome Pressure-High (2B31-N679 A, D)	11	1	1, 2, 3	28

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION 21 - Be in at least STARTUP with the main steam line isolation valves closed within 2 hours or be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION 22 - Be in at least STARTUP within 2 hours.

ACTION 23 - Be in at least STARTUP with the Group 1 isolation valves closed within 2 hours or in at least HOT SHUTDOWN within 6 hours.

ACTION 24 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

ACTION 25 - Isolate the reactor water cleanup system.

ACTION 26 - Close the affected system isolation valves and declare the affected system inoperable.

ACTION 27 - Verify power availability to the bus at least once per 12 hours or close the affected system isolation valves and declare the affected system inoperable.

ACTION 28 - Close the shutdown cooling supply and reactor vessel head spray isolation valves unless reactor steam dome pressure $\leq$ 145 psig.

NOTES

Actuates operation of the main control room environmental control system in the pressurization mode of operation.

* Actuates the standby gas treatment system.

** When handling irradiated fuel in the secondary containment.

a. See Specification 3.6.3, Table 3.6.3-1 for valves in each valve group.

b. A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

c. With a design providing only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken.

d. Trips the mechanical vacuum pumps.

e. A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.

f. May be bypassed with all turbine stop valves closed.

g. Closes only RWCU outlet isolation valve 2G31-F004.

h. Alarm only.

i. Adjustable up to 60 minutes.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1. Low (Level 3)	≥ 8.5 inches*	≥ 8.5 inches*
2. Low Low (Level 2)	≥ -55 inches*	≥ -55 inches*
3. Low Low Low (Level 1)	≥ -121.5 inches*	≥ -121.5 inches*
b. Drywell Pressure - High	≤ 1.85 psig	≤ 1.85 psig
c. Main Steam Line		
1. Radiation - High	≤ 3 x full power background	≤ 3 x full power
2. Pressure - Low	≥ 825 psig	≥ 825 psig
3. Flow - High	≤ 138% rated flow	≤ 138% rated flow
d. Main Steam Line Tunnel Temperature - High	≤ 194°F	≤ 194°F
e. Condenser Vacuum - Low	≥ 7" Hg vacuum	≥ 7" Hg vacuum
f. Turbine Building Area Temp.-High	≤ 200°F	≤ 200°F
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Exhaust Radiation - High	≤ 60 mr/hr	≤ 60 mr/hr
b. Drywell Pressure - High	≤ 1.85 psig	≤ 1.85 psig
c. Reactor Vessel Water Level - Low Low (Level 2)	≥ -55 inches*	≥ -55 inches*
d. Refueling Floor Exhaust Radiation - High	≤ 20 mr/hr	≤ 20 mr/hr

*See Bases Figure B 3/4 3-1.

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High	≤ 79 gpm	≤ 79 gpm
b. Area Temperature-High	≤ 124°F	≤ 124°F
c. Area Ventilation Δ Temperature - High	≤ 67°F	≤ 67°F
d. SLCS Initiation	NA	NA
e. Reactor Vessel Water Level-Low Low (Level 2)	≥ -55 inches*	≥ -55 inches*
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION		
a. HPCI Steam Line Flow-High	≤ 307% of rated flow	≤ 307% of rated flow
b. HPCI Steam Supply Pressure - Low	≥ 100 psig	≥ 100 psig
c. HPCI Turbine Exhaust Diaphragm Pressure-High	≤ 20 psig	≤ 20 psig
d. HPCI Pipe Penetration Room Temperature - High	≤ 169°F	≤ 169°F
e. Suppression Pool Area Ambient Temperature-High	≤ 169°F	≤ 169°F
f. Suppression Pool Area ΔT - High	≤ 42.5°F	≤ 42.5°F
g. Suppression Pool Area Temperature Timer Relays	NA	NA
h. Emergency Area Cooler Temperature - High	≤ 169°F	≤ 169°F
i. Drywell Pressure - High	≤ 1.85 psig	≤ 1.85 psig
j. Logic Power Bus Monitors	NA	NA

*See Bases Figure B 3/4 3-1.

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Flow - High	≤ 312% of rated flow	≤ 312% of rated flow
b. RCIC Steam Supply Pressure - Low	≥ 60 psig	≥ 60 psig
c. RCIC Turbine Exhaust Diaphragm Pressure - High	≤ 20 psig	≤ 20 psig
d. Emergency Area Cooler Temperature-High	≤ 169°F	≤ 169°F
e. Suppression Pool Area Ambient Temperature High	≤ 169°F	≤ 169°F
f. Suppression Pool Area ΔT - High	≤ 42.5°F	≤ 42.5°F
g. Suppression Pool Area Temperature Timer Relays	NA	NA
h. Drywell Pressure - High	≤ 1.85 psig	≤ 1.85 psig
i. Logic Power Monitor	NA	NA
6. SHUTDOWN COOLING SYSTEM ISOLATION		
a. Reactor Vessel Water Level - Low (Level 3)	≥ 8.5 inches*	≥ 8.5 inches*
b. Reactor Steam Dome Pressure - High	≤ 145 psig	≤ 145 psig

*See Bases Figure B 3/4 3-1.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1. Low (Level 3)	$\leq$ 13*
2. Low Low (Level 2)	$\leq$ 13*
3. Low Low Low (Level 1), except MSIVs	$\leq$ 13*
b. Drywell Pressure - High	$\leq$ 13*
c. Main Steam Line	
1. Radiation - High***	$\leq$ 1.0**
2. Pressure - Low	$\leq$ 13*
3. Flow - High	$\leq$ 1.0**
4. Reactor Vessel Water Level - Low Low Low (Level 1)	$\leq$ 1.0**
d. Main Steam Line Tunnel Temperature - High	$\leq$ 13*
e. Condenser Vacuum - Low	NA
f. Turbine Building Area Temperature - High	NA
2. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Building Exhaust Radiation - High***	$\leq$ 13*
b. Drywell Pressure - High	$\leq$ 13*
c. Reactor Vessel Water Level - Low Low (Level 2)	$\leq$ 13*
d. Refueling Floor Exhaust Radiation - High***	$\leq$ 13*

*The isolation actuation instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Response time specified is diesel generator start delay time assumed in accident analysis.

**Isolation actuation instrumentation response time.

***Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

#Times to be added to valve movement times shown in Tables 3.6.3-1, 3.6.5.2-1 and 3.9.5.2-1 to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	$\leq$ 13*
b. Area Temperature - High	$\leq$ 13*
c. Area Ventilation Temperature ΔT - High	$\leq$ 13*
d. SLCS Initiation	NA
e. Reactor Vessel Water Level-Low Low (Level 2)	$\leq$ 13*
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION	
a. HPCI Steam Line Flow-High	$\leq$ 13*
b. HPCI Steam Supply Pressure - Low	$\leq$ 13*
c. HPCI Turbine Exhaust Diaphragm Pressure - High	NA
d. HPCI Pipe Penetration Room Temperature - High	NA
e. Suppression Pool Area Ambient Temp. - High	NA
f. Suppression Pool Area ΔT - High	NA
g. Suppression Pool Area Temp. Timer Relays	NA
h. Emergency Area Cooler Temperature - High	NA
i. Drywell Pressure - High	$\leq$ 13*
j. Logic Power Monitor	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Flow - High	NA
b. RCIC Steam Supply Pressure - Low	NA
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
d. Emergency Area Cooler Temperature - High	NA
e. Suppression Pool Area Ambient Temp. - High	NA
f. Suppression Pool Area ΔT - High	NA
g. Suppression Pool Area Temperature Timer Relays	NA
h. Drywell Pressure - High	$\leq$ 13*
i. Logic Power Monitor	NA
6. SHUTDOWN COOLING SYSTEM ISOLATION	
a. Reactor Vessel Water Level - Low (Level 3)	NA
b. Reactor Steam Dome Pressure - High	NA

TABLE 4.3.2-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1. Low (Level 3)	S	M	R	1, 2, 3
2. Low Low (Level 2)	S	M	R	1, 2, 3
3. Low Low Low (Level 1)	S	M	R	1, 2, 3
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Main Steam Line				
1. Radiation - High	D	W(a)	R	1, 2, 3
2. Pressure - Low	NA	M	Q	1
3. Flow - High	S	M	R	1, 2, 3
d. Main Steam Line Tunnel Temperature - High	S	M	R	1, 2, 3
e. Condenser Vacuum - Low	NA	M	Q	1, 2#, 3#
f. Turbine Building Area Temp. - High	NA	M	R	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Exhaust Radiation - High	D	M(a)	R	1, 2, 3, 5 and *
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Reactor Vessel Water Level - Low Low (Level 2)	S	M	R	1, 2, 3
d. Refueling Floor Exhaust Radiation - High	D	M(a)	Q	1, 2, 3, 5 and *

*When handling irradiated fuel in the secondary containment.
#May be bypassed with all turbine stop valves closed.
aInstrument alignment using a standard current source.

TABLE 4.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	D	M	R	1, 2, 3
b. Area Temperature - High	S	M	R	1, 2, 3
c. Area Ventilation Δ Temperature - High	S	M	R	1, 2, 3
d. SLCS Initiation	NA	R	NA	1, 2, 3
e. Reactor Vessel Water Level - Low Low (Level 2)	S	M	R	1, 2, 3
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Flow-High	S	M	R	1, 2, 3
b. HPCI Steam Supply Pressure-Low	S	M	R	1, 2, 3
c. HPCI Turbine Exhaust Diaphragm Pressure - High	S	M	R	1, 2, 3
d. HPCI Pipe Penetration Room Temperature - High	S	M	R	1, 2, 3
e. Suppression Pool Area Ambient Temp. - High	S	M	R	1, 2, 3
f. Suppression Pool Area ΔT - High	S	M	R	1, 2, 3
g. Suppression Pool Area Temp. Timer Relays	NA	SA	R	1, 2, 3
h. Emergency Area Cooler Temp. - High	S	M	R	1, 2, 3
i. Drywell Pressure - High	S	M	R	1, 2, 3
j. Logic Power Monitor	NA	R	NA	1, 2, 3

TABLE 4.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow-High	S	M	R	1, 2, 3
b. RCIC Steam Supply Pressure-Low	S	M	R	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure-High	S	M	R	1, 2, 3
d. Emergency Area Cooler Temperature - High	S	M	R	1, 2, 3
e. Suppression Pool Area Ambient Temperature-High	S	M	R	1, 2, 3
f. Suppression Pool Area ΔT - High	S	M	R	1, 2, 3
g. Suppression Pool Area Temp. Timer Relays	NA	SA	R	1, 2, 3
h. Drywell Pressure - High	S	M	R	1, 2, 3
i. Logic Power Monitor	NA	R	NA	1, 2, 3
6. SHUTDOWN COOLING SYSTEM ISOLATION				
a. Reactor Vessel Water Level - Low (Level 3)	S	M	R	3, 4, 5
b. Reactor Steam Dome Pressure - High	S	M	R	1, 2, 3

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS
1. CORE SPRAY SYSTEM		
a. Reactor Vessel Water Level - Low Low Low (Level 1) (2B21-N691A,B,C,D)	2	1,2,3,4,5
b. Drywell Pressure - High (2E11-N694 A,B,C,D)	2	1,2,3
c. Reactor Steam Dome Pressure - Low(Injection Permissive) (2B21-N690A,B,C,D)	2	1,2,3,4,5
d. Logic Power Monitor (2E21-K1A,B)	1/bus[(a)]	1,2,3,4,5
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM		
a. Drywell Pressure - High (2E11-N694A,B,C,D)	2	1,2,3
b. Reactor Vessel Water Level - Low Low Low (Level 1) (2B21-N691A,B,C,D)	2	1,2,3,4*,5*
c. Reactor Vessel Shroud Level (Level 0) - High (Drywell Spray Permissive) (2B21-N685A, B)	1	1,2,3,4*,5*
d. Reactor Steam Dome Pressure - Low (Injection Permissive) (2B21-N690A,B,C,D)	2	1,2,3,4*.5*
e. Reactor Steam Dome Pressure - Low (Recirc. Discharge Valve Permissive) (2B21-N641B,C and 2B21-N690E,F)	2	1,2,3,4*,5*
f. RHR Pump Start - Time Delay Relay	1/pump	1,2,3,4*,5*
1) Pump A (2E11-K70A, 2E11-K125B)		
2) Pump B (2E11-K70B, 2E11-K125A)		
3) Pump C (2E11-K75B)		
4) Pump D (2E11-K75A, 2E11-K126)		
g. Logic Power Monitor (2E11-K1A,B)	1/bus[(a)]	1,2,3,4*,5*

* Not applicable when two core spray system subsystems are OPERABLE per Specification 3.5.3.1.

(a) Alarm only. When inoperable, verify power availability to the bus at least once per 12 hours or declare the system inoperable.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS#
3. HIGH PRESSURE COOLANT INJECTION SYSTEM		
a. Reactor Vessel Water Level - Low Low (Level 2) (2B21-N692 A,B,C,D)	2	1, 2, 3
b. Drywell Pressure - High (2E11-N694 A,B,C,D)	2	1, 2, 3
c. Condensate Storage Tank Level-Low (2E41-N002, 2E41-N003)	2(b)(c)	1, 2, 3
d. Suppression Chamber Water Level-High (2E41-N662B,D)	2(b)(c)	1, 2, 3
e. Logic Power Monitor (2E41-K1)	1(a)	1, 2, 3
f. Reactor Vessel Water Level-High (Level 8) (2B21-N693 B,D)	2	1, 2, 3
4. AUTOMATIC DEPRESSURIZATION SYSTEM		
a. Drywell Pressure - High (Permissive) (2E11-N694A,B,C,D)	2	1, 2, 3
b. Reactor Vessel Water Level - Low Low Low (Level 1) (2B21-N691 A,B,C,D)	2	1, 2, 3
c. ADS Timer (2B21-K5A,B)	1	1, 2, 3
d. Reactor Vessel Water Level-Low (Level 3) (Permissive) (2B21-N695A,B)	1	1, 2, 3
e. Core Spray Pump Discharge Pressure - High (Permissive) (2E21-N655A,B; 2E21-N652A,B)	2	1, 2, 3
f. RHR (LPCI MODE) Pump Discharge Pressure - High (Permissive) (2E11-N655A,B,C,D; 2E11-N656A,B,C,D)	2/loop	1, 2, 3
g. Control Power Monitor (2B21-K1A,B)	1/bus(a)	1, 2, 3
5. LOW LOW SET S/RV SYSTEM		
a. Reactor Steam Dome Pressure - High (Permissive) (2B21-N620A,B,C,D)	2	1, 2, 3

(a) Alarm only. When inoperable, verify power availability to the bus at least once per 12 hours or declare the system inoperable.

(b) Provides signal to HPCI pump suction valves only.

(c) When either channel of the automatic transfer logic is inoperable, align HPCI pump suction to the suppression pool.

\# HPCI and ADS are not required to be OPERABLE with reactor steam dome pressure $\leq$ 150 psig.

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. CORE SPRAY SYSTEM		
a. Reactor Vessel Water Level - Low Low Low (Level 1)	$\geq$ -121.5 inches*	$\geq$ -121.5 inches*
b. Drywell Pressure - High	$\leq$ 1.85 psig	$\leq$ 1.85 psig
c. Reactor Steam Dome Pressure - Low	$\geq$ 422 psig**	$\geq$ 422 psig**
d. Logic Power Monitor	NA	NA
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM		
a. Drywell Pressure - High	$\leq$ 1.85 psig	$\leq$ 1.85 psig
b. Reactor Vessel Water Level - Low Low Low (Level 1)	$\geq$ -121.5 inches*	$\geq$ -121.5 inches*
c. Reactor Vessel Shroud Level (Level 0) - High	$\geq$ -207 inches*	$\geq$ -207 inches*
d. Reactor Steam Dome Pressure-Low	$\geq$ 422 psig**	$\geq$ 422 psig**
e. Reactor Steam Dome Pressure-Low	$\geq$ 325 psig	$\geq$ 325 psig
f. RHR Pump Start - Time Delay Relay		
1) Pumps A, B and D	10 ± 1 seconds	10 ± 1 seconds
2) Pump C	0.5 ± 0.5 seconds	0.5 ± 0.5 seconds
g. Logic Power Monitor	NA	NA

*See Bases Figure B 3/4 3-1.
**This trip function shall be less than or equal to 500 psig.

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
3. HIGH PRESSURE COOLANT INJECTION SYSTEM		
a. Reactor Vessel Water Level - Low Low (Level 2)	$\geq$ -55 inches*	$\geq$ -55 inches*
b. Drywell Pressure-High	$\leq$ 1.85 psig	$\leq$ 1.85 psig
c. Condensate Storage Tank Level - Low	$\geq$ 0 inches**	$\geq$ 0 inches**
d. Suppression Chamber Water Level - High	$\leq$ 33.2 inches	$\leq$ 33.2 inches
e. Logic Power Monitor	NA	NA
f. Reactor Vessel Water Level-High (Level 8)*	$\leq$ 56.5 inches	$\leq$ 56.5 inches
4. AUTOMATIC DEPRESSURIZATION SYSTEM		
a. Drywell Pressure-High	$\leq$ 1.85 psig	$\leq$ 1.85 psig
b. Reactor Vessel Water Level - Low Low Low (Level 1)	$\geq$ -121.5 inches*	$\geq$ -121.5 inches*
c. ADS Timer	$\leq$ 120 seconds	$\leq$ 120 seconds
d. Reactor Vessel Water Level-Low (Level 3)	$\geq$ 8.5 inches*	$\geq$ 8.5 inches*
e. Core Spray Pump Discharge Pressure - High	$\geq$ 130 psig	$\geq$ 130 psig
f. RHR (LPCI MODE) Pump Discharge Pressure - High	$\geq$ 105 psig	$\geq$ 105 psig
g. Control Power Monitor	NA	NA
5. LOW LOW SET S/RV SYSTEM		
a. Reactor Steam Dome Pressure - High	$\leq$ 1054 psig	$\leq$ 1054 psig

* See Bases Figure B 3/4 3-1.
** Equivalent to 10,000 gallons of water in the CST.

TABLE 4.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. CORE SPRAY SYSTEM				
a. Reactor Vessel Water Level - Low Low Low (Level 1)	S	M	R	1, 2, 3, 4, 5
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Reactor Steam Dome Pressure - Low	S	M	R	1, 2, 3, 4, 5
d. Logic Power Monitor	NA	R	NA	1, 2, 3, 4, 5
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM				
a. Drywell Pressure - High	S	M	R	1, 2, 3
b. Reactor Vessel Water Level - Low Low Low (Level 1)	S	M	R	1, 2, 3, 4*, 5*
c. Reactor Vessel Shroud Level (Level 0) - High	S	M	R	1, 2, 3, 4*, 5*
d. Reactor Steam Dome Pressure - Low	S	M	R	1, 2, 3, 4*, 5*
e. Reactor Steam Dome Pressure - Low	S	M	R	1, 2, 3, 4*, 5*
f. RHR Pump Start-Time Delay Relay	NA	NA	R	1, 2, 3, 4*, 5*
g. Logic Power Monitor	NA	R	NA	1, 2, 3, 4*, 5*

*Not applicable when two core spray system subsystems are OPERABLE per Specification 3.5.3.1.

TABLE 4.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED#
3. HIGH PRESSURE COOLANT INJECTION SYSTEM				
a. Reactor Vessel Water Level - Low Low (Level 2)	S	M	R	1, 2, 3
b. Drywell Pressure-High	S	M	R	1, 2, 3
c. Condensate Storage Tank Level - Low	NA	M	Q	1, 2, 3
d. Suppression Chamber Water Level - High	S	M	R	1, 2, 3
e. Logic Power Monitor	NA	R	NA	1, 2, 3
f. Reactor Vessel Water Level-High (Level 8)	S	M	R	1, 2, 3
4. AUTOMATIC DEPRESSURIZATION SYSTEM				
a. Drywell Pressure-High	S	M	R	1, 2, 3
b. Reactor Vessel Water Level - Low Low Low (Level 1)	S	M	R	1, 2, 3
c. ADS Timer	NA	NA	R	1, 2, 3
d. Reactor Vessel Water Level - Low (Level 3)	S	M	R	1, 2, 3
e. Core Spray Pump Discharge Pressure - High	S	M	R	1, 2, 3
f. RHR (LPCI MODE) Pump Discharge Pressure - High	S	M	R	1, 2, 3
g. Control Power Monitor	NA	R	NA	1, 2, 3
5. LOW LOW SET S/RV SYSTEM				
a. Reactor Steam Dome Pressure - High	S	M	R	1, 2, 3

\# HPCI and ADS are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

TABLE 3.3.4-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	MINIMUM NUMBER OF OPERABLE CHANNELS PER TRIP SYSTEM
a. Reactor Vessel Water Level - Low Low (Level 2) (2B21-N692 A, B, C, D)	2

TABLE 3.3.4-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level - Low Low (Level 2)	$\geq$ -55 inches*	$\geq$ -55 inches*

* See Bases Figure B 3/4 3-1.

TABLE 4.3.4-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level - Low Low (Level 2)	S	M	R

TABLE 3.3.6.4-1

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE
1. Reactor Vessel Pressure (2B21-R623 A, B)	2
2. Reactor Vessel Shroud Water Level (2B21-R610, 2B21-R615)	2
3. Suppression Chamber Water Level (2T48-R622 A, B)	2
4. Suppression Chamber Water Temperature (2T47-R626, 2T47-R627)	2
5. Suppression Chamber Pressure (2T48-R608, 2T48-R609)	2
6. Drywell Pressure (2T48-R608, 2T48-R609)	2
7. Drywell Temperature (2T47-R626, 2T47-R627)	2
8. Post-LOCA Gamma Radiation (2D11-K622 A, B, C, D)	2
9. Drywell H_2 -O_2 Analyzer (2P33-R601 A, B)	2
10.a)Safety/Relief Valve Position Primary Indicator (2B21-N301 A-H and K-M)	*
b)Safety/Relief Valve Position Secondary Indicator (2B21-N004 A-H and K-M)	*

*If either the primary or secondary indication is inoperable, the torus temperature will be monitored at least once per shift to observe any unexplained temperature increases which might be indicative of an open SRV. With both the primary and secondary monitoring channels of an SRV inoperable, either verify that the S/RV is closed through monitoring the backup low low set logic position indicators (2B21-N302 A-H and K-M) at least once per shift or restore sufficient inoperable channels such that no more than one SRV has both primary and secondary channels inoperable within 7 days or be in at least hot shutdown within the next 12 hours.

REACTOR COOLANT SYSTEM

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1054 psig.

APPLICABILITY: CONDITION 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1054 psig, reduce the pressure to less than 1054 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than 1054 psig at least once per 12 hours.

* Not applicable during anticipated transients.

3. Verifying a system flow rate of 4000 +0, -1000 cfm during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches Water Gauge while operating the filter train at a flow rate of 4000 +0, -1000 cfm.

2. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

a. Drywell pressure-high,

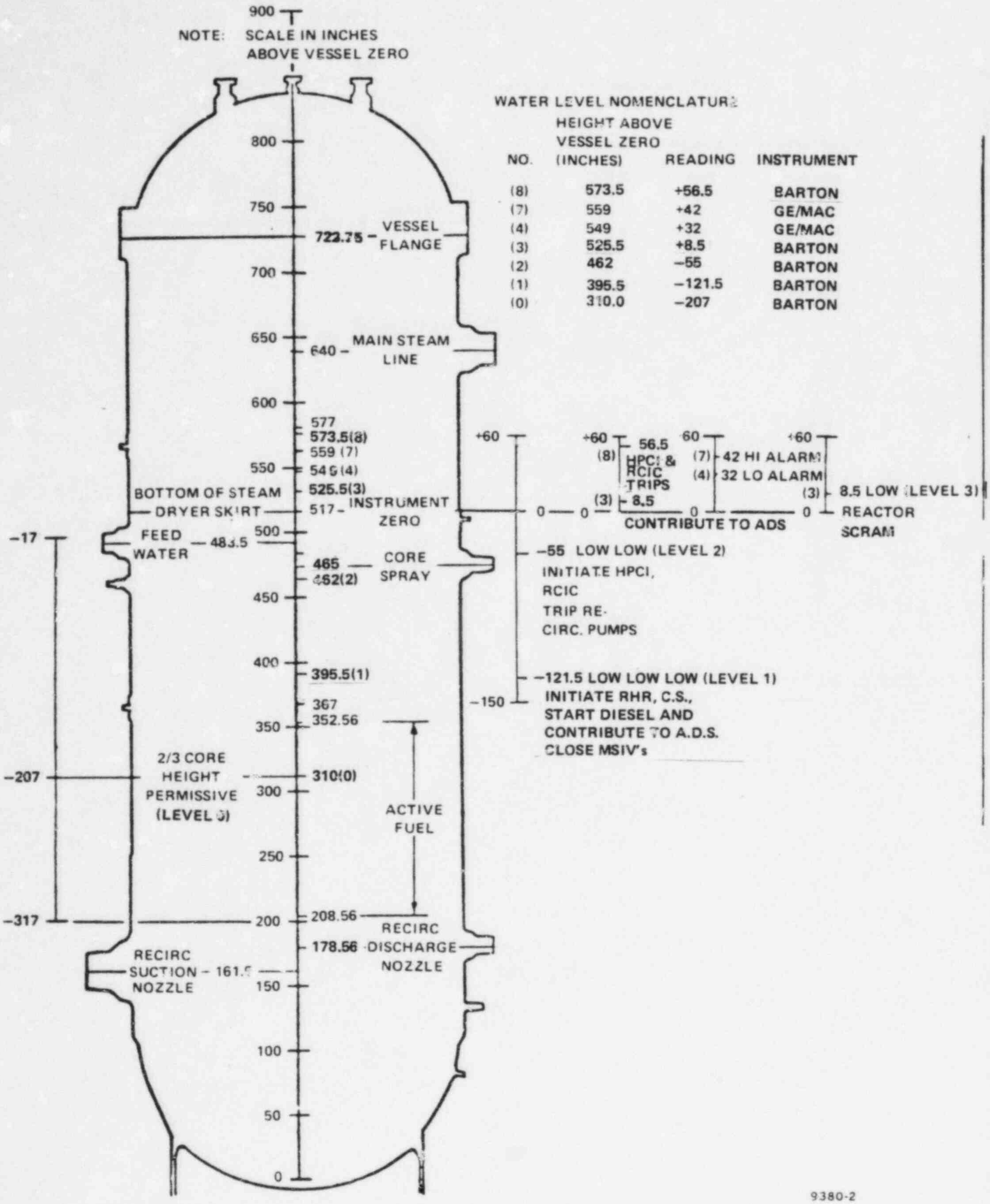
b. High radiation on the;

1) Refueling floor,

2) Reactor building.

c. Reactor Vessel Water Level-Low Low (Level 2).

3. Verifying that the heaters dissipate 18.5 + 1.5 KW when tested in accordance with ANSI N510-1975.



NO.	HEIGHT ABOVE VESSEL ZERO (INCHES)	READING	INSTRUMENT
(8)	573.5	+56.5	BARTON
(7)	559	+42	GE/MAC
(4)	549	+32	GE/MAC
(3)	525.5	+8.5	BARTON
(2)	462	−55	BARTON
(1)	395.5	−121.5	BARTON
(0)	310.0	−207	BARTON

BASES FIGURE B 3/4 3-1
REACTOR VESSEL WATER LEVELS

APPENDIX 1

SIGNIFICANT HAZARDS REVIEW

10 CFR 50.92 Evaluation for the Proposed Changes to the Technical Specifications as a Result of the Installation of the Analog Transmitter Trip System for Edwin I. Hatch Nuclear Plant-Unit 2

Georgia Power Company (GPC) has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed Technical Specifications changes due to the installation of the analog transmitter trip system (ATTS). ATTS replaces the pressure, level, and temperature switches in the reactor protection system and emergency core cooling system (ECCS) with analog sensor/trip unit combinations. The system is designed to improve sensor intelligence and reliability, while providing continued monitoring of critical parameters and performing the basic logic function. Since the ATTS instrumentation is superior to the mechanical switches currently used at Plant Hatch and demonstrates less drift, certain surveillance requirements may be extended without significantly increasing the probability or consequences of an accident previously evaluated. GPC proposes to change the surveillance requirements for the ATTS instrumentation to once per shift for channel checks, once per month for channel functional tests, and once per operating cycle for channel calibrations. These proposed surveillance requirements were previously approved on a generic basis for ATTS equipment by the Nuclear Regulatory Commission (NRC) in NEDO-21617-A. Additional changes to the nomenclature used in the Technical Specifications are included for clarification and consistency with this proposed change. GPC has reviewed these proposed changes and considers them not to involve a significant hazards consideration, because they will not significantly increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different accident from any accident previously evaluated, or involve a significant reduction in a margin of safety.

The results of this change are clearly within design and all acceptance criteria; therefore, this change is consistent with Item (vi) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983, issue of the Federal Register. The changes in nomenclature are administrative in nature and consistent with Item (vi) of the above referenced page of the Federal Register.

10 CFR 50.92 Evaluation for the Proposed Changes to the Technical Specifications Reactor Vessel Water Level - Low Low (Level 2) Trip Setpoint for Edwin I. Hatch Nuclear Plant-Unit 2[a]

Georgia Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed reactor vessel water level 2 trip setpoint Technical Specifications change. This proposed change lowers the level 2 trip setpoint/allowable value from -38 in. to -55 in., increasing plant operability by decreasing the number of trips due to normal operational transients. Thus, this change will not involve a significant hazards consideration, because it will not significantly increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety.

The proposed reactor vessel water level 2 trip setpoint/allowable value revision is well within design and all other acceptance criteria. Consequently, this change is consistent with Item (vi) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983, issue of the Federal Register and will not result in a significant hazards consideration.

a. See subsection 4B.1 (page 4-31) for discussion of proposed changes.

10 CFR 50.92 Evaluation for the Proposed Changes to the Technical Specifications due to the Deletion of the High Drywell Pressure Signal for Residual Heat Removal (Shutdown Cooling Mode), Reactor Pressure Vessel (RPV) Head Spray Valves, and Reactor Water Cleanup Isolation for Edwin I. Hatch Nuclear Plant-Unit 2[a]

Georgia Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed changes to the Technical Specifications due to the deletion of the high drywell pressure signal for residual heat removal (RHR) (shutdown cooling mode), RPV head spray valves, and RWCU isolation. The purpose of this change is to stop small steam leaks in the drywell from preventing operation of the RHR and RWCU systems during the shutdown cooling mode, thereby prohibiting an acceptable normal shutdown procedure. GPC has reviewed this change and considers it not to involve a significant hazards consideration, because it will not significantly increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety.

Although this change removes a trip signal, it was shown that the use of high drywell pressure as an isolation signal has negligible effect in preventing coolant losses due to an RHR or RWCU pipe break. Therefore, this change does not affect the Appendix K calculation results, and the requirements of 10 CFR 100 are still met. This change is consistent with Item (vi) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Considerations" listed on page 14870 of the April 6, 1983, issue of the Federal Register and will not result in a significant hazards consideration. Thus, the results of this change are clearly within all acceptance criteria.

a. See subsection 4B.2 (page 4-4) for discussion of proposed changes.

10 CFR 50.92 Evaluation for the Proposed Changes to the Technical Specifications due to the Lowered Water Level Trip Setpoint for Isolation of Reactor Water Cleanup and Secondary Containment, and Starting of Standby Gas Treatment System for Edwin I. Hatch Nuclear Plant-Unit 2[a]

Georgia Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed changes to the Technical Specifications due to the lowered water level trip setpoint for isolation of RWCU and secondary containment, and starting of the standby gas treatment system (SGTS). This change proposes to lower the water level trip setpoint for isolation of RWCU and secondary containment, and startup of the standby gas treatment system (SGTS) from level 3 to level 2. A reactor scram from normal power (<50-percent rated) usually results in a reactor vessel water level transient due to a void collapse that causes RWCU isolation at level 3. This usually results in the dropping of the cleanup filter cake, added radwaste processing, and loss of ability to remove water from the reactor vessel immediately after scram. These problems may be avoided by lowering RWCU isolation to level 2. Lowering the SGTS actuation and secondary containment isolation from level 3 to level 2 reduces the potential for spurious isolations. GPC has reviewed these changes and does not consider them to involve a significant hazards consideration, because they will not significantly increase the probability or consequences of an accident from any accident previously evaluated, create the possibility of a new or different accident from any accident previously evaluated, or involve a significant reduction in a margin of safety.

Since the above changes reduce operational problems associated with RWCU and secondary containment isolation and because the ECCS analysis design basis already assumes that the SGTS initiates at level 2, these changes are well within the design and all other acceptance criteria. Consequently, this change is consistent with Item (vi) of the "Example of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983, issue of the Federal Register and will not result in a significant hazards consideration.

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a. See subsection 4B.3 (page 4-5) for discussion of proposed changes.

10 CFR 50.92 Evaluation for the Proposed Changes to the Technical Specifications due to the Deletion of Ambient Temperature Loops in the Leak Detection System Modification for Edwin I. Hatch Nuclear Plant-Unit 2[a]

Georgia Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed change to the Technical Specifications due to the deletion of ambient temperature loops from the leak detection system. As part of the ATTS modification, the change proposes to use the hot leg of the differential temperature sensor for high ambient trip rather than using an independent trip element and trip device. This arrangement may cause slight changes in the sensitivity of the leak detection system, depending upon the heating, ventilation, and air-conditioning design, but it will not defeat the intended function of the system. The proposed Technical Specifications revisions will reference the trip unit loop from which the ambient temperature trip is taken in place of the existing ambient temperature trip instrument. GPC has reviewed this change and considers it not to involve a significant hazards consideration, because it will not significantly increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety.

The proposed revision to the leak detection system is well within design and all other acceptance criteria. Since the current single-failure criteria will be maintained by this new logic, this change is consistent with Item (vi) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983, issue of the Federal Register and will not result in a significant hazards consideration.

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a. See subsection 4B.4 (page 4-6) for discussion of proposed changes.

10 CFR 50.92 Evaluation for the Proposed Changes to the Technical Specifications due to the Deletion of Drywell Pressure Sensors E11-N011A, B, C, D for Edwin I. Hatch Nuclear Plant-Unit 2[a]

Georgia Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed change to the Technical Specifications due to the deletion of drywell pressure sensors E11-N011A, B, C, D. This change proposes to make the drywell pressure sensor configuration consistent with water levels 2 and 3 sensors. Drywell pressure sensors E11 N010A, B, C, D may be used to provide signals for all four systems of the ECCS and still maintain single-failure criteria. The Technical Specifications revision involves changing the instrument numbers from E11-N010A, B, C, D to E11-N694A, B, C, D. Thus, this change involves no reduction in safety. GPC has reviewed this proposed change and considers it not to involve a significant hazards consideration, because it will not significantly increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different accident from any accident previously evaluated, or involve a significant reduction in a margin of safety.

Although sensors E11-N011A, B, C, D are being deleted, this change involves no reduction in safety and is within all design and acceptance criteria. Therefore, GPC considers this to be consistent with Item (vi) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983, issue of the Federal Register and will not result in a significant hazards consideration.

a. See subsection 4B.5 (page 4-9) for discussion of proposed changes.

10 CFR 50.92 Evaluation for the Proposed Trip Setpoint/Allowable Value Modifications to the Technical Specifications for Edwin I. Hatch Nuclear Plant-Unit 2[a]

Georgia Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed miscellaneous trip setpoint/allowable value modifications to the Technical Specifications. The purpose of this change is to update the Technical Specifications trip setpoints. Since the time that the original setpoints were calculated, a better methodology has been developed. This proposed change uses Regulatory Guide 1.105 methodology in updating the setpoints for the instruments being replaced with the new ATTS units. This change replaces the trip setpoints listed in the Technical Specifications which are the original analytical limits with the newly evaluated allowable values determined through Regulatory Guide 1.105 methodology. GPC has reviewed this change and considers it not to involve a significant hazards consideration, because it will not significantly increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different accident from any accident previously evaluated, or involve a significant reduction in a margin of safety.

Since the new setpoints were determined using the criteria of Regulatory Guide 1.105; and the new ATTS instruments are more accurate and reliable, the setpoint changes are clearly within all design and acceptance criteria. Consequently, GPC considers the setpoint changes to be consistent with both Item (vi), as discussed above, and Item (vii) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983, issue of the Federal Register and will not result in a significant hazards consideration.

a. See subsection 4B.6 (page 4-10) for discussion of proposed changes.

10 CFR 50.92 Evaluation for the Proposed Changes to the Technical Specifications due to the Reactor Vessel Water Level - High (Level 8) Trip Instrumentation Modification for Edwin I. Hatch Nuclear Plant-Unit 2[a]

Georgia Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed changes to the Technical Specifications due to the reactor vessel water level - high (level 8) trip instrumentation. This change proposes to replace the current instruments with new ATTS units. Since the ATTS instruments are superior to the mechanical switches currently used at Plant Hatch and demonstrate less drift, the setpoint/allowable value will be lowered from 58 in. to 56.5 in., using the criteria of Regulatory Guide 1.105. GPC has reviewed this change and considers it not to involve a significant hazards consideration, because it will not significantly increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety.

Since the new ATTS instrumentation is superior and demonstrates less drift and the new setpoints were calculated using the criteria of Regulatory Guide 1.105, the lowering of the reactor vessel water level 8 trip setpoint/allowable value is well within the design and all other acceptance criteria. Consequently, this change is consistent with Item (vi) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983, issue of the Federal Register and will not result in a significant hazards consideration.

a. See subsection 4B.7 (page 4-20) for discussion of proposed changes.

10 CFR 50.92 Evaluation for the Proposed Change to the Technical Specifications as a Result of the Elimination of the Reactor Pressure Permissive to the Bypass of the MSIV Closure Signal Due to Low Condenser Vacuum for Edwin I. Hatch Nuclear Plant-Unit 2[a]

Georgia Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed change to the Technical Specifications as a result of the elimination of the reactor pressure permissive to the bypass of the main steam isolation valve (MSIV) closure signal due to low condenser vacuum. The change proposes to delete the reactor steam dome pressure permissive which allows the group 1 isolation valves to be bypassed on a low condenser vacuum isolation signal. With the permissive deleted, the operator may open the valves from a hot pressurized condition before clearing a scram. GPC has reviewed this change and considers it not to involve a significant hazards consideration, because it will not significantly increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different accident from any accident previously evaluated, or involve a significant reduction in a margin of safety.

Consequently, this change is consistent with Item (vi) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983, issue of the Federal Register and will not result in a significant hazards consideration.

By eliminating this permissive, the plant protection features and, therefore, plant safety are not compromised in any manner.

a. See subsection 4B.8 (page 4-21) for discussion of proposed changes.