UNITED STATES NUCLEAR REGULATORY COMMISSION

POLICY ISSUE
(Information)

May 28 1992

SECY-92-196

For: The Commissioners

From: James M. Taylor
Executive Director for Operations

Subject: DEVELOPMENT OF DESIGN ACCEPTANCE CRITERIA (DAC) FOR THE ADVANCED BOILING WATER REACTOR (ABWR)

Purpose: To inform the Commission of the status of the development of DAC for the GE Nuclear Energy (GE) ABWR.

Background: The staff has proposed the use of DAC as an approach to the design review and resulting design certification for the GE ABWR to resolve the difficulties being experienced in obtaining detailed design information for selected areas of the plant. The staff discussed this issue in SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews." The staff discussed two of these DAC areas in SECY-91-272, "Role of Personnel and Advanced Control Rooms in Future Nuclear Power Plants," and SECY-91-292, "Digital Computer Systems for Advanced Light Water Reactors."

Discussion: Design and engineering information for some areas of the design at a level of detail customarily reviewed by the staff in making a final safety determination is not available at this time from GE. GE has provided less detailed information in these areas because they are areas of rapidly changing technology or because they are areas for which GE does not have sufficient as-built or as-procured information to complete the final design. These areas include piping

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NOTE: TO BE MADE PUBLICLY AVAILABLE IN 3 WORKING DAYS FROM THE DATE OF THIS PAPER design, radiation shielding and airborne concentrations, control room design, and advanced instrumentation and controls. GE is working with the staff to develop DAC which when met would ensure the completed design and as-constructed plant conforms to the design certification. The use of DAC would enable the staff to make a final safety determination for the design, subject only to satisfactory design implementation and verification by the combined license applicant and holder, through appropriate inspections, tests, analyses, and acceptance criteria (ITAAC). Thus, the acceptance criteria for DAC are specified together with the related ITAAC, and both are part of the design certification.

GE has stated that it will submit the full set of DAC by May 31, 1992. The Nuclear Regulatory Commission's (NRC's) senior management has participated significantly in developing and reviewing the DAC, including participating in meetings with the Advisory Committee on Reactor Safeguards (ACRS). The staff has interacted extensively with GE to develop these DAC in conjunction with the ITAAC for the ABWR.

The DAC are a set of prescribed limits, parameters, procedures, and attributes, in conjunction with the available design information, which the NRC relies upon in a limited number of technical areas to make a final safety determination in support of the ABWR design certification. The acceptance criteria for the DAC areas are objective; that is, they are inspectable, testable, or subject to analysis using pre-approved methods. They will be incorporated into the design certification rule as appropriate for Tier 1 information. The standard safety analysis report (SSAR) will include, as appropriate, sample calculations or other supporting information to illustrate methods that are acceptable to the staff for meeting Tier 1 DAC commitments.

The DAC areas are generally comprised of three parts. These parts consist of the Tier 1 Design Description, the corresponding ITAAC, and the Tier 2 supporting information contained in the SSAR for the DAC area. This format is consistent with the format of other ITAAC for which DAC are not needed for the staff to reach a final safety decision. The staff will base its safety findings for the DAC areas on the Tier 2 information specified in the SSAR, the Tier 1 design information, including applicable methodologies, specified in the Tier 1 Design Description and the corresponding ITAAC. GE has stated that it will provide cross references showing where the acceptance criteria for the DAC areas apply to the systems.

Enclosure 1 is a draft of the staff's final safety evaluation report (SER) on the radiation protection and airborne concentration DAC area. The enclosure contains a Tier 1 Design Description, the corresponding ITAAC, and the corresponding Tier 2 SSAR information. GE cannot provide the complete design information in this DAC area before design certification because the radiation shielding design and the calculated airborne concentrations are dependent on as-built, as-procured information. The ITAAC in the enclosure address the verification of the plant radiation shielding design and the plant airborne concentrations of radioactive materials (e.g., the ventilation system and airborne monitoring system designs).

Enclosure 2 is a draft of the staff's final SER of the piping design DAC area. The Tier 1 design description, ITAAC, and corresponding SSAR are included. It should be noted that there are open items in the staff's SER and positions taken by the staff which are being addressed by GE. The staff expects additional GE submittals to close these open items, and in some cases GE will agree to staff positions. With respect to dynamic analysis of piping to assure pipe stress is within allowable values, the DAC is essentially complete. In this area, GE cannot have complete piping layout and final stress analyses before design certification because this information is dependent on as-built, as-procured data. Much of the staff's SER is based upon the staff's audit of the main steam, feedwater, and the safety-relief valve discharge piping analysis.

GE has stated that it will provide the DAC submittals for the control room design and instrumentation and controls systems as part of the third phase ITAAC submittal by May 31, 1992. In accordance with SECY-91-161, "Schedules for the Advanced Reactor Reviews and Regulatory Guidance Revisions," the staff is scheduled to issue a final safety evaluation report (FSER) in August 1992. As indicated in the Quarterly Status Report of Advanced Light Water Reactor Reviews (December 1991 - February 1992), dated April 9, 1992, the staff intends to supplement the FSER following its August 1992 issuance to address several issues including GE's May 31, 1992, ITAAC submittal. The August 1992 FSER issuance is, however, expected to provide the Commission and the ACRS with the bulk of the staff's safety findings including the evaluation of the DAC submitted by GE. The staff is continuing to interact with the ACRS to fully evaluate the use of DAC and will consider the ACRS comments on this and related ITAAC and DAC SECY papers in the development of the FSER.

Enclosures 1 and 2 are being provided for information on progress being made in the review and technical approach to DAC in the areas of radiation protection and piping design. The final SER on these areas will be part of the staff's SER on the ABWR for the final design approval (FDA). No decision is requested from the Commission on these DAC at this time.

Coordination: The staff is continuing to meet with the ACRS to discuss the development of the DAC areas.

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Enclosures:
1. Radiation Protection DAC Material
   Appendix A - Design Description and ITAAC
   Appendix B - Draft Safety Evaluation Report
   Appendix C - SSAR Material
2. Piping Design DAC Material
   Appendix A - Design Description and ITAAC
   Appendix B - Draft Safety Evaluation Report
   Appendix C - SSAR Material

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Enclosure 1

RADIATION PROTECTION DAC MATERIAL

- Appendix A - Design Description and ITAAC
- Appendix B - Draft Safety Evaluation Report
- Appendix C - SSAR Material

3.7 Radiation Protection

Design Description

The ABWR design provides radiation protection features that will keep exposures for both plant personnel and the general public well below allowable limits. These low exposure conditions are achieved by an integrated approach that recognizes the contribution of both shielding provisions and ventilation system designs that control airborne contaminants. Monitoring of radiation levels is an integral part of the plant radiation protection strategy.

The plant design provides radiation shielding for rooms, corridors and operating areas commensurate with their occupancy requirements and thus maintains radiation exposures to plant personnel as low as reasonably achievable. Maintenance of plant components is achieved without significant radiation exposure from adjacent plant systems or equipment by use of shielded cubicles, labyrinth access and provisions for temporary shielding. Under accident conditions, plant shielding designs permit operators to perform required safety functions in vital areas of the plant. In addition to protection of operating personnel, the plant design provides radiation shielding which maintains radiation exposure to the general public as low as is reasonably achievable.

Plant ventilation systems insure that concentrations of airborne radionuclides are maintained at levels consistent with personnel access requirements. In addition, airborne radioactivity monitoring is provided for those normally occupied areas of the plant in which there exists a significant potential for airborne contamination.

Inspection, Test, Analyses and Acceptance Criteria

Tables 3.7a and 3.7b provide a definition of the inspections, tests, and/or analyses together with associated acceptance criteria which will be undertaken for the ABWR plant shielding, ventilation and airborne monitoring equipment.

Table 3.7a: Plant Shielding Design

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The plant design shall provide radiation shielding for rooms, corridors and operating areas commensurate with their occupancy requirements to maintain radiation exposures to plant personnel as low as reasonably achievable.	1. An analysis of the expected radiation levels in each plant area will be performed to verify the adequacy of the shielding design. This analysis shall consider the following: a. Confirmatory calculations shall consider all significant radiation sources (greater than 5% contribution) for an area. Radiation source strength in plant systems and components will be determined based upon an assumed source term of 100,000 µCurie/second offgas release rate (after 30 minutes decay), a 200 µCurie/gram-steam N-16 source term at the vessel exit nozzle, and a core inventory commensurate with a 4005 MWT equilibrium core at 51.6 kwatt/liter. All source terms shall be adjusted for radiological decay and buildup of activated corrosion and wear products. b. Commonly accepted shielding codes, using nuclear properties derived from well known references (such as Vitamin C and ANSI/ANS-6.4) shall be used to model and evaluate plant radiation environments. 1) For non-complex geometries, point kernal shielding codes (such as QAD or GGG) shall be used. 2) For complex geometries, more sophisticated two or three dimensional transport codes (such as DORT or TORT) shall be used.	1. Maximum expected radiation levels are well within (25% or less) of the radiation zone designation, for each plant area, as indicated in Figures 3.7.a through 3.7.bb.

Table 3.7a: Plant Shielding Design (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	1. (Cont.) c. In any calculation, a safety factor shall be applied based upon benchmark comparisons of the code and data collected from known and measured environments.	
2. The plant design shall provide shielded cubicles, labyrinth access, and space for temporary shielding to allow for maintenance of plant components without significant radiation exposure from adjacent plant systems or equipment.	2. Using the methods identified in (1) above, radiation levels present in areas where maintenance is performed shall be evaluated for the contribution from adjacent high radiation areas and equipment.	2. Shielding design (with temporary shielding installed, where appropriate) is such that radiation from adjacent areas shall contribute no more than a small fraction (10% or less) of the radiation field intensity or less than 0.06mrem/hr whichever is larger, in plant areas where maintenance is performed.
3. The plant radiation shielding design shall permit operators to perform required safety functions in vital areas of the plant (including access and egress of these areas) under accident conditions.	3. An analysis of the expected high radiation levels in each area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident (vital area) shall be performed to verify the adequacy of the plant shielding design. This analysis shall use calculational methods consistent with (1.b) above and a radiation source term (adjusted for radioactive decay) based on the following: a. Liquid containing systems: 100% of the core equilibrium noble gas inventory, 50% of the core equilibrium halogen inventory and 1% of all others are assumed to be mixed in the reactor coolant and recirculation liquids recirculated by the residual heat removal system (RHR), the high	3. Under accident conditions, radiation shielding design allows access, occupancy and egress of vital areas such that personnel radiation exposures do not exceed 5 rem to the whole body, or its equivalent, for the duration of the accident (based on the required frequency of access to each vital area). For areas requiring continuous occupancy (such as the control room), local radiation hot spots shall not exceed 15 mrem/hr (averaged over 30 days).

Table 3.7a: Plant Shielding Design (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	3. (Cont.) pressure core flooder (HPCF), and the reactor core isolation cooling (RCIC) systems. b. Gas containing systems: 100% of the core equilibrium noble gas inventory and 25% of the core equilibrium halogen activity are assumed to be mixed in the containment atmosphere. For vapor containing systems (such as the main steam lines) these core inventory fractions are assumed to be contained in the reactor coolant vapor space.	
4. The plant design shall provide radiation shielding to maintain radiation exposure to the general public as low as is reasonably achievable.	4. Using the methods identified in (1) above, the radiation dose to the maximally exposed member of the general public from direct and scattered shall be determined.	4. The radiation dose to the maximally exposed member of the public is a small fraction (10% or less) of the dose limit to a member of the public listed in 40CFR190.

Table 3.7b: Ventilation And Airborne Monitoring

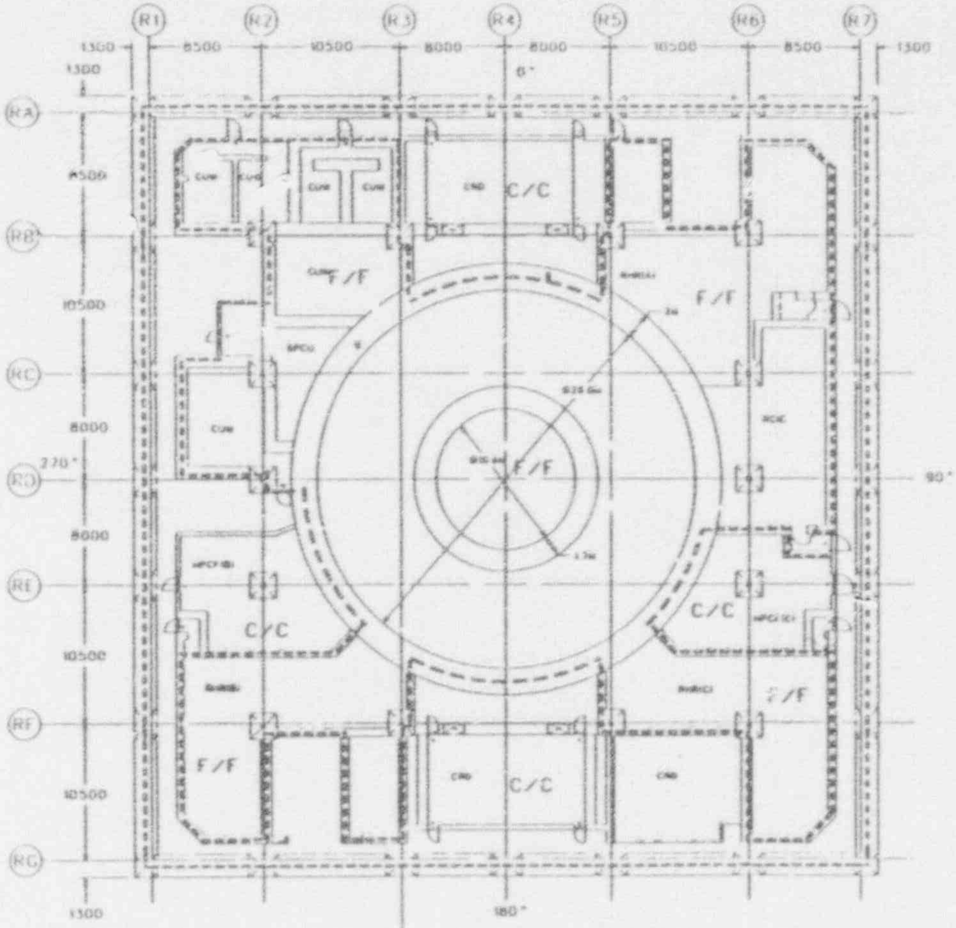
Inspections, Tests, Analyses and Acceptance Criteria

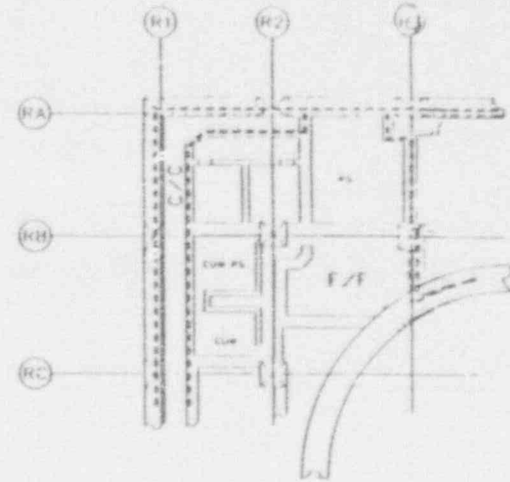
Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Plant design shall provide adequate containment of airborne radioactive materials and the ventilation system will ensure that concentrations of airborne radionuclides are maintained at levels consistent with personnel access requirements.	1. Expected concentrations of airborne radioactive material shall be calculated by nuclide for normal plant operations, anticipated operational occurrences for each equipment cubicle, corridor, and operating area requiring personnel access. Calculations shall consider: a. Design ventilation flow rates for each area, b. Typical leakage characteristics for equipment located in each area, and c. A radiation source term in each fluid system shall be determined based upon an assumed offgas rate of 100,000 Curie/second (30 minute decay) appropriately adjusted for radiological decay and buildup of activated corrosion and wear products.	1. Calculation of radioactive airborne concentration shall demonstrate that: a. For normally occupied rooms and areas of the plant (i.e. those areas requiring routine access to operate and maintain the plant) equilibrium concentrations of airborne nuclides will be a small fraction (10% or less) of the occupational concentration limits listed in 10 CFR 20 Appendix B. b. For rooms that require infrequent access (such as for non-routine equipment maintenance), the ventilation system shall be capable of reducing radioactive airborne concentrations to (and maintaining them at) the occupational concentration limits listed in 10CFR20 Appendix B during the periods that occupancy is required. c. For rooms that seldom require access (such as tank rooms), plant design shall provide sufficient containment and ventilation to ensure airborne contamination does not spread to other areas.

Table 3.7b: Ventilation And Airborne Monitoring (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2. Airborne radioactivity monitoring shall be provided for those normally occupied areas of the plant in which there exists as significant potential for airborne contamination (greater than 0.1 per year)	2. An analysis shall be performed to identify the plant areas that require airborne radioactivity monitoring.	2. Airborne radioactivity monitoring system shall: a. Have the capability of detecting the time integrated change in concentrations of the most limiting particulate and iodine radionuclides in each area equivalent to the occupational concentration limits in 10CFR20, Appendix B for 10hours. b. Provide a calibrated response, representatvie of the concentrations within the area (i.E. air sampling monitors in ventilation exhaust streams shall collect and isokinetic sample). c. Provide local audible alarms (visual alarms in high noise areas) with variable alarm set points, and readout/ annunciation capability in the control room





FULL POWER/SHUTDOWN
RADIATION LEVELS IN
mRem/hour

A < 0.6
B < 1
C < 5
D < 25
E < 100
F > 100

FIGURE 3.7a REACTOR BUILDING
RADIATION ZONE MAP FOR
FULL POWER AND SHUTDOWN OPERATIONS
AT ELEVATION 8200 (B3F)

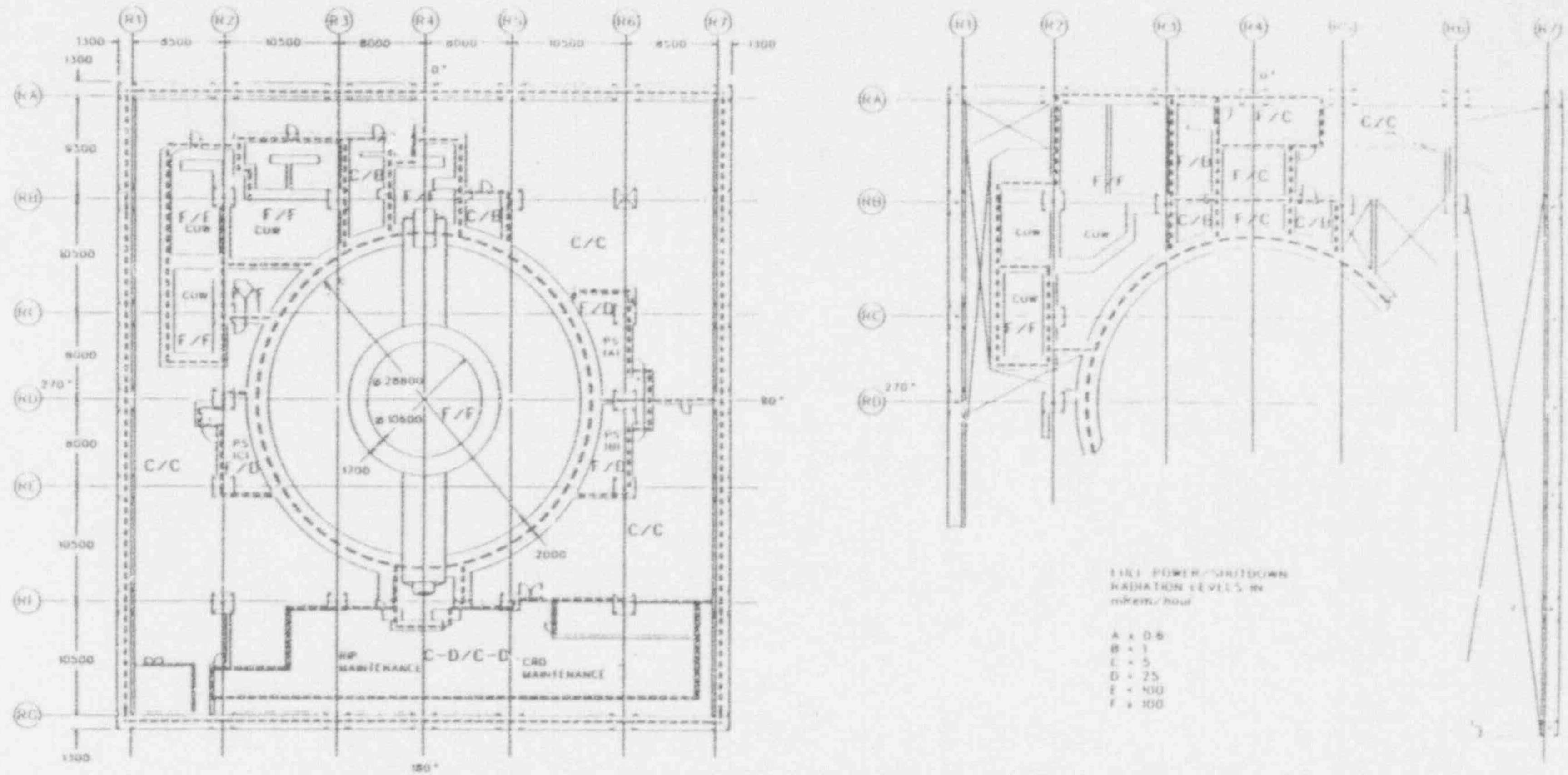
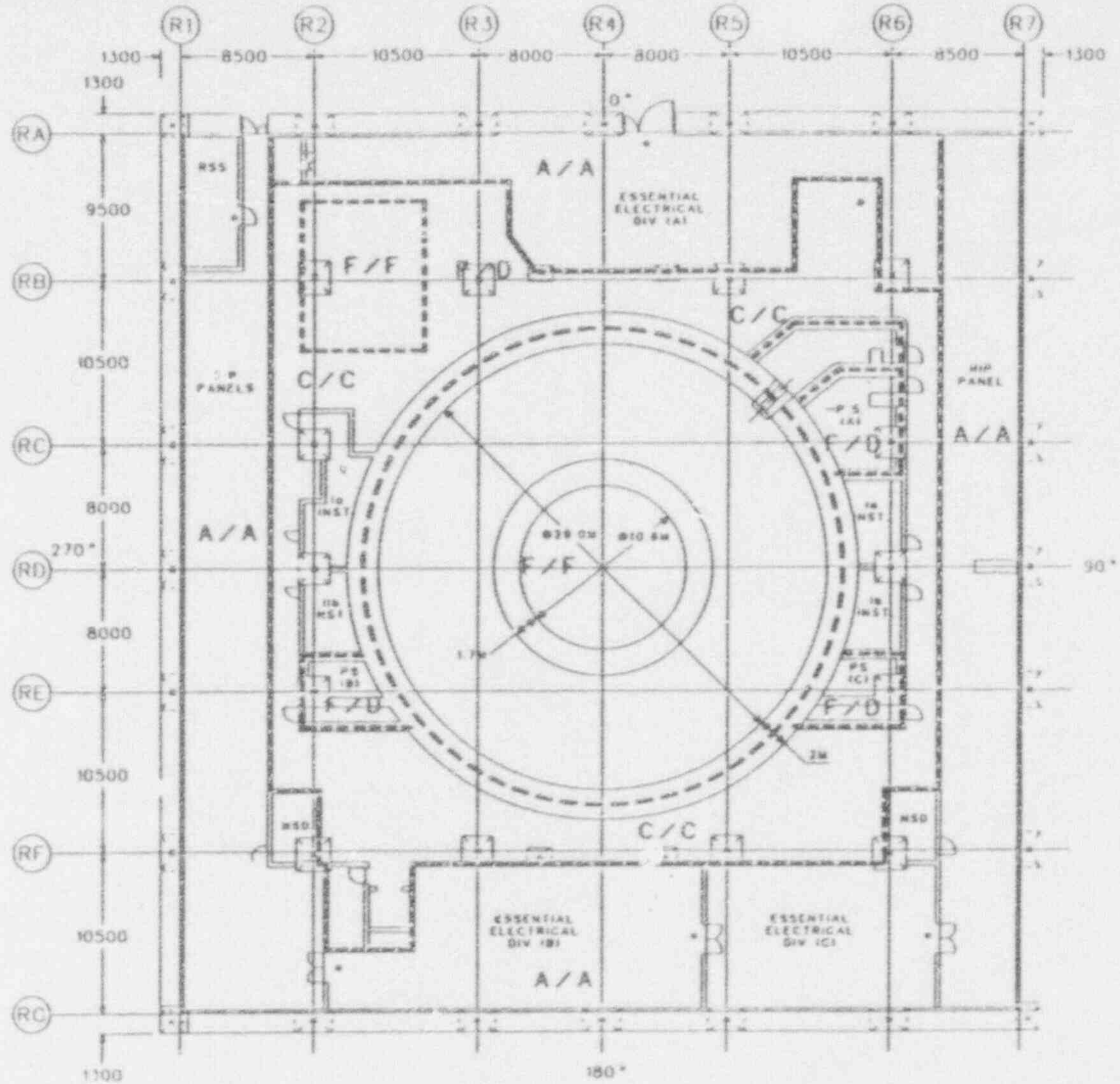


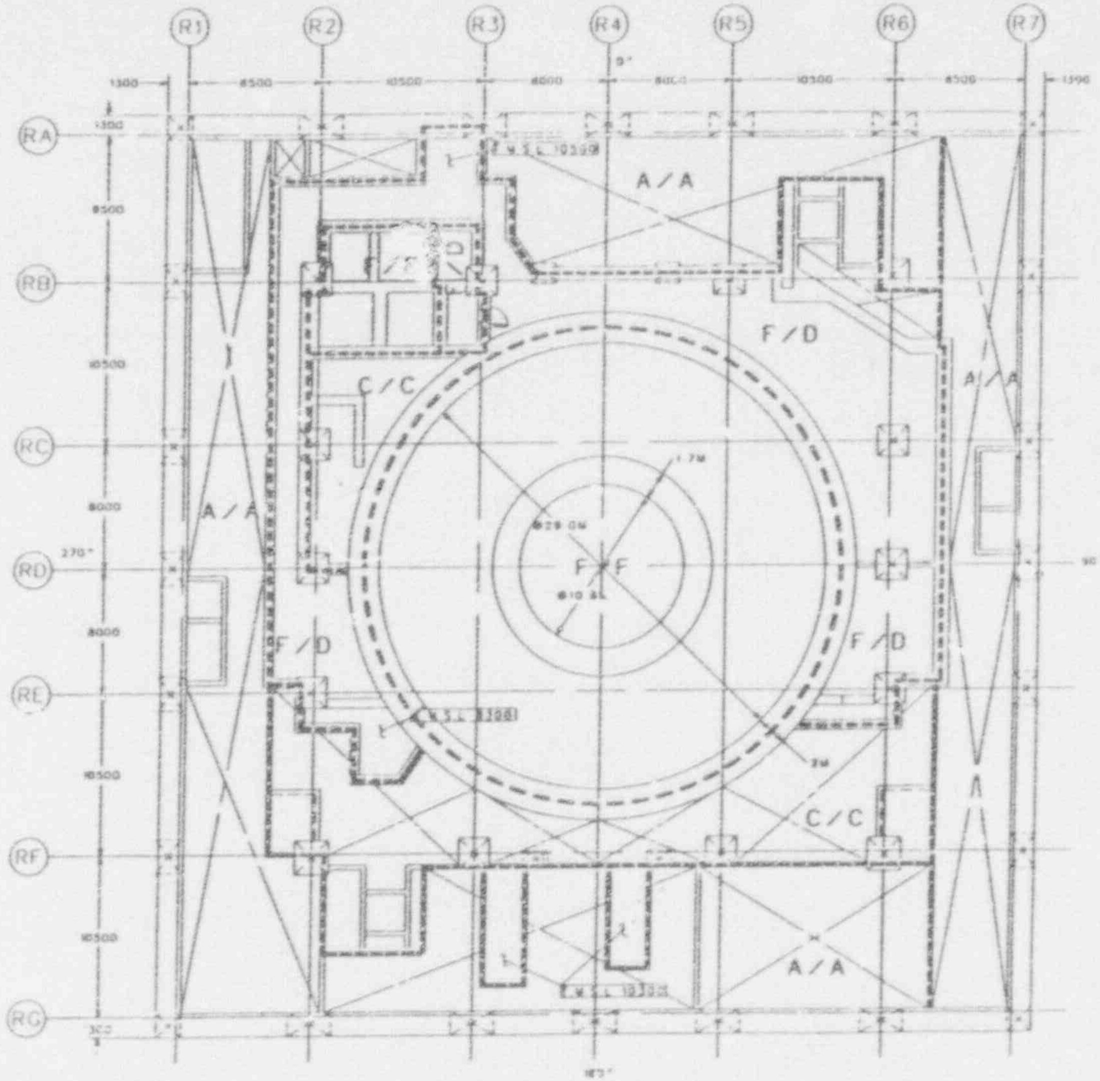
FIG. 3.7-6 REACTOR BUILDING
RADIATION ZONE MAP FOR
FULL POWER AND SHUTDOWN OPERATIONS
AT ELEVATION 1700 (62FT)



FULL POWER/SHUTDOWN
RADIATION LEVELS IN
mRem/hour

A ≤ 0.6
B < 1
C < 5
D < 25
E < 100
F > 100

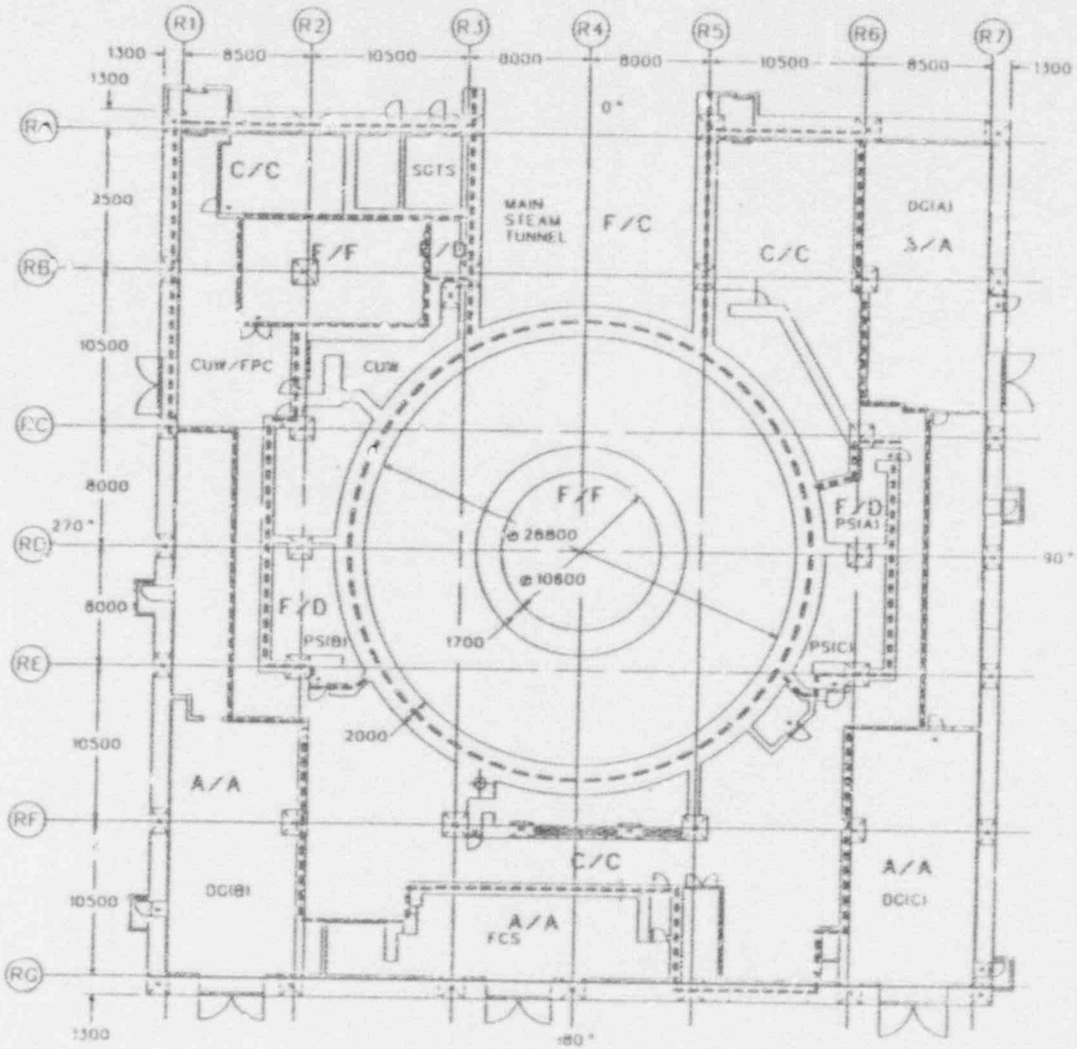
FIGURE 3.7c REACTOR BUILDING
RADIATION ZONE MAP FOR
FULL POWER, AND SHUTDOWN OPERATIONS
AT ELEVATION -4800 (B1F)



FULL POWER/SHUTDOWN
RADIATION LEVELS IN
mRem/hour

A ≤ 0.6
B < 1
C < 5
D < 25
E < 100
F ≥ 100

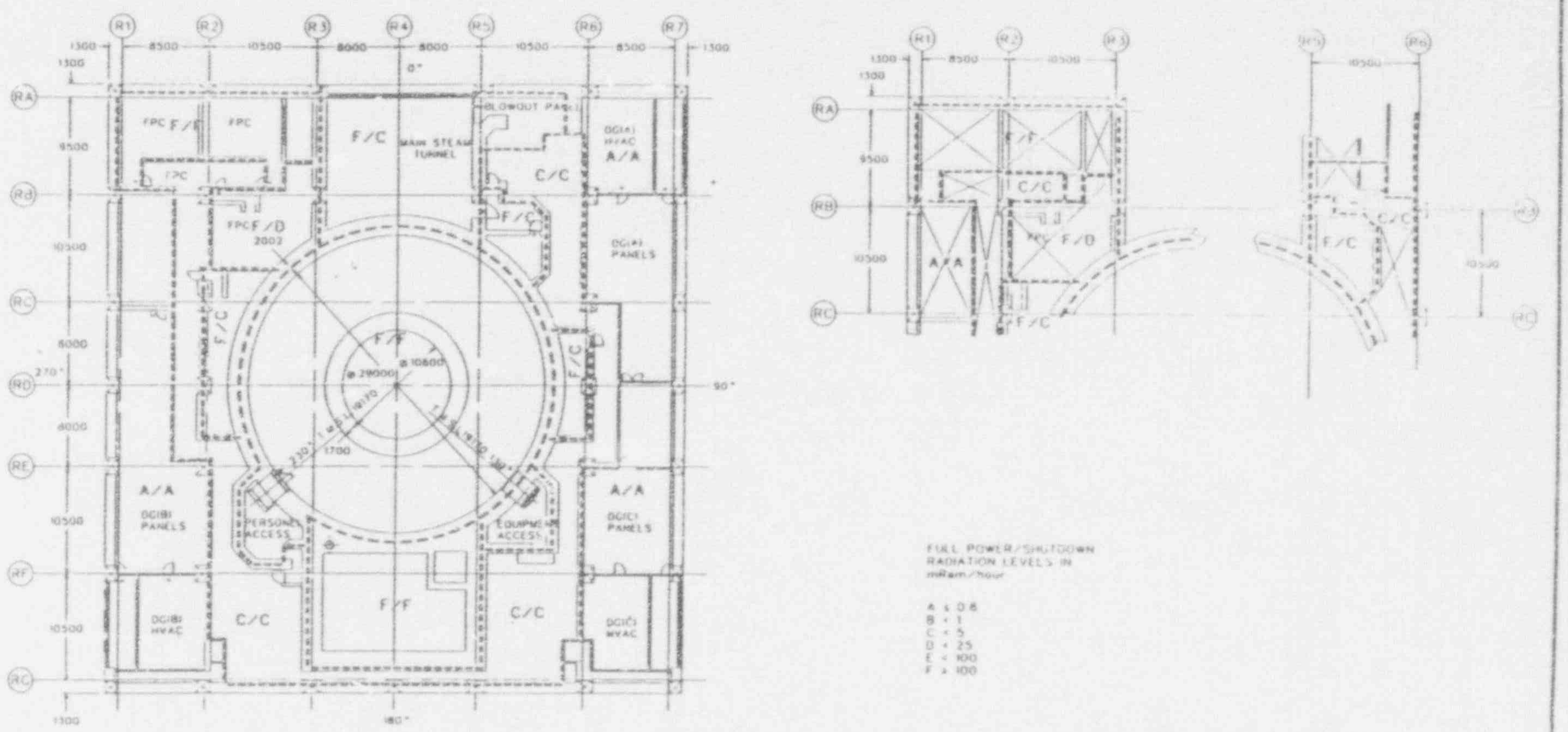
FIGURE 3.7d REACTOR BUILDING
RADIATION ZONE MAP FOR
FULL POWER AND SHUTDOWN OPERATIONS
AT ELEVATION -4500 (B1F)



FULL POWER/SHUTDOWN
RADIATION LEVELS IN
mRem/hour

A ≤ 0.5
B < 1
C < 5
D < 25
E < 100
F ≥ 100

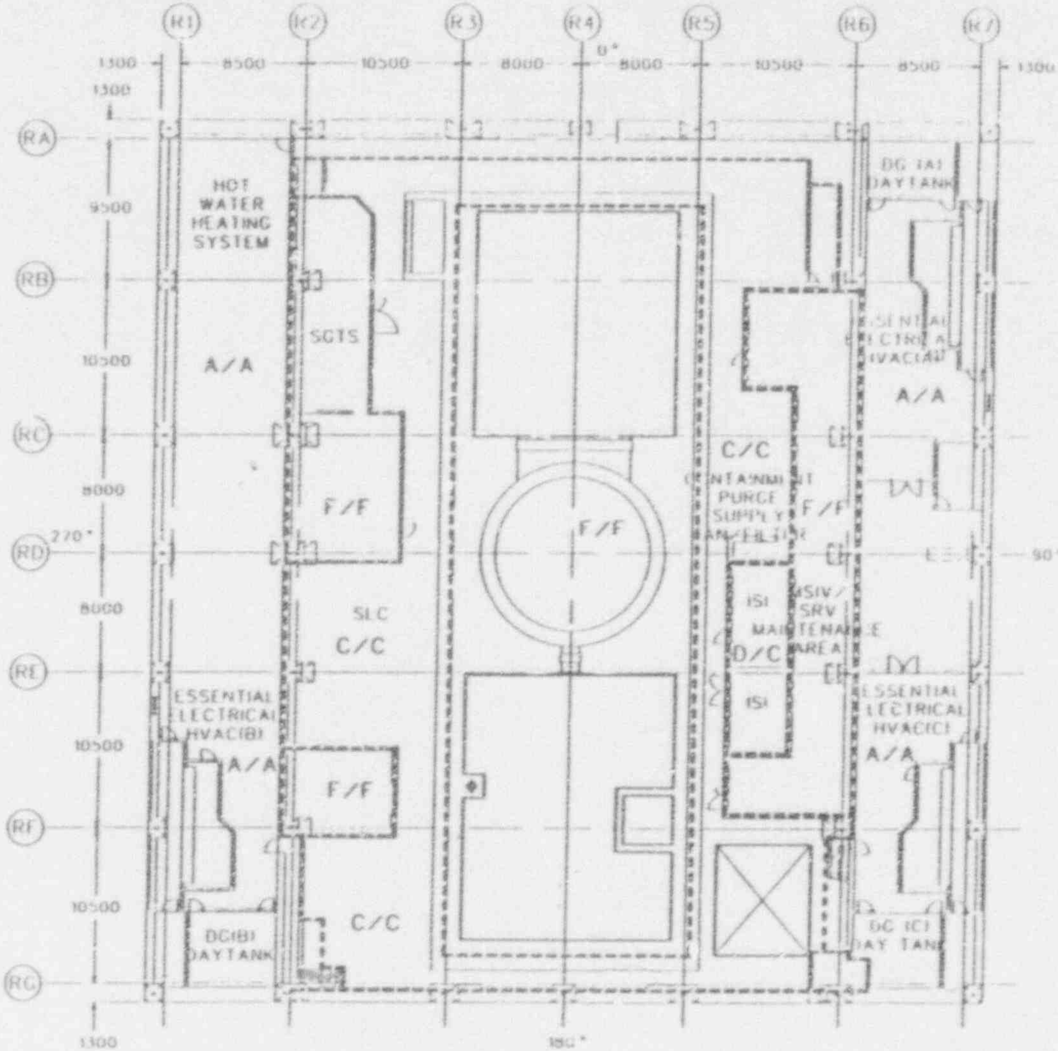
FIGURE 3.7e REACTOR BUILDING
RADIATION ZONE MAP FOR
FULL POWER AND SHUTDOWN OPERATIONS
AT ELEVATION 12300 (1F)



FULL POWER/SHUTDOWN
RADIATION LEVELS IN
mRem/hour

A ≤ 0.8
B < 1
C < 5
D < 25
E < 100
F ≥ 100

FIGURE 3.7I - - .TOR BUILDING
RADIATION ZONE MAP FOR
FULL POWER AND SHUTDOWN OPERATIONS
AT ELEVATION 18100 (2F)

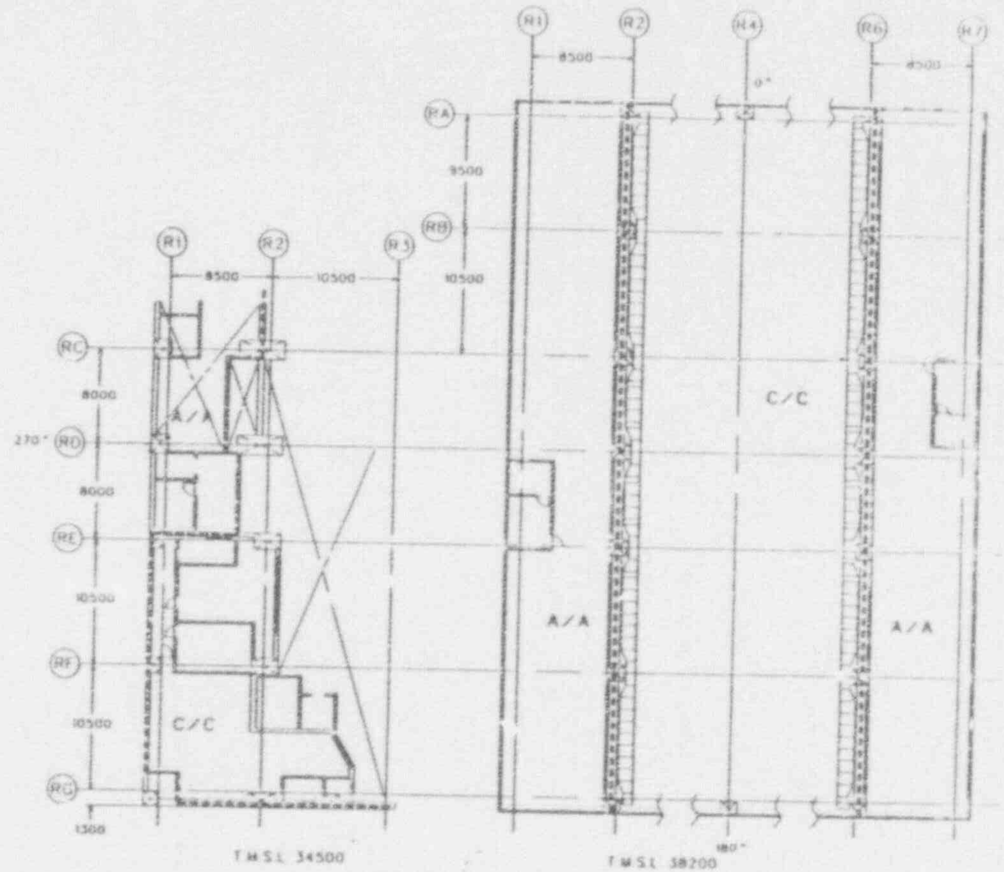


FULL POWER/SHUTDOWN
RADIATION LEVELS IN
mRem/hour

A ≤ 0.6
B < 1
C < 5
D < 25
E < 100
F ≥ 100

FIGURE 3.7-j REACTOR BUILDING
RADIATION ZONE MAP FOR
FULL POWER AND SHUTDOWN OPERATIONS
AT ELEVATION 23500 (B3F)

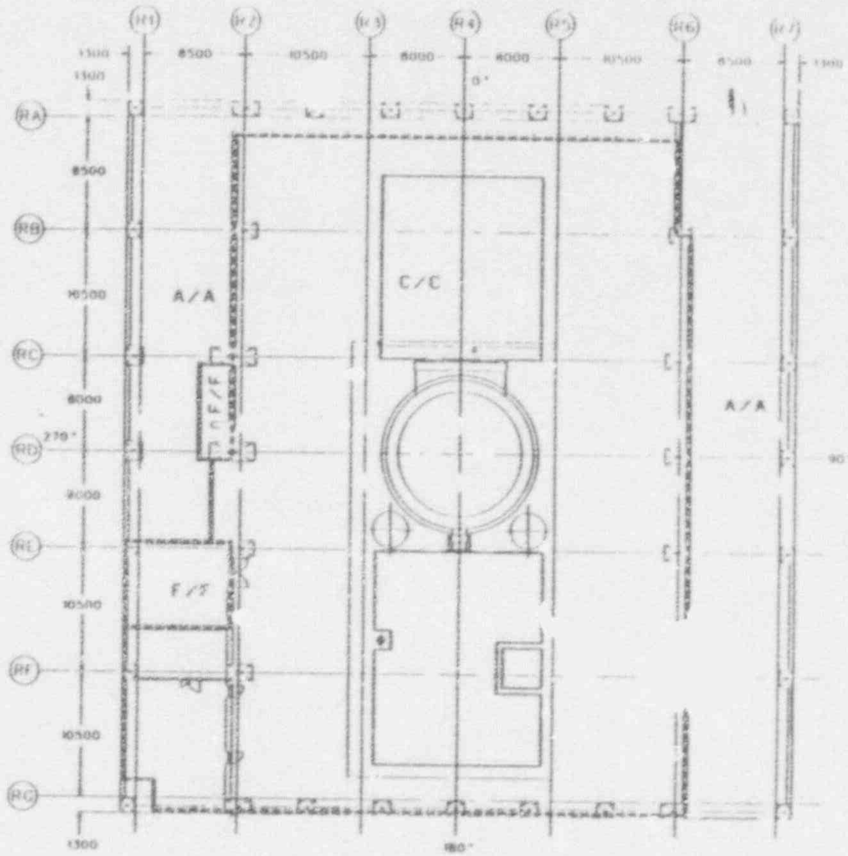
3.30.92



FULL POWER/SHUTDOWN
RADIATION LEVELS IN
mRem/hour

A ≤ 0.6
B < 1
C < 5
D < 25
E < 100
F ≥ 100

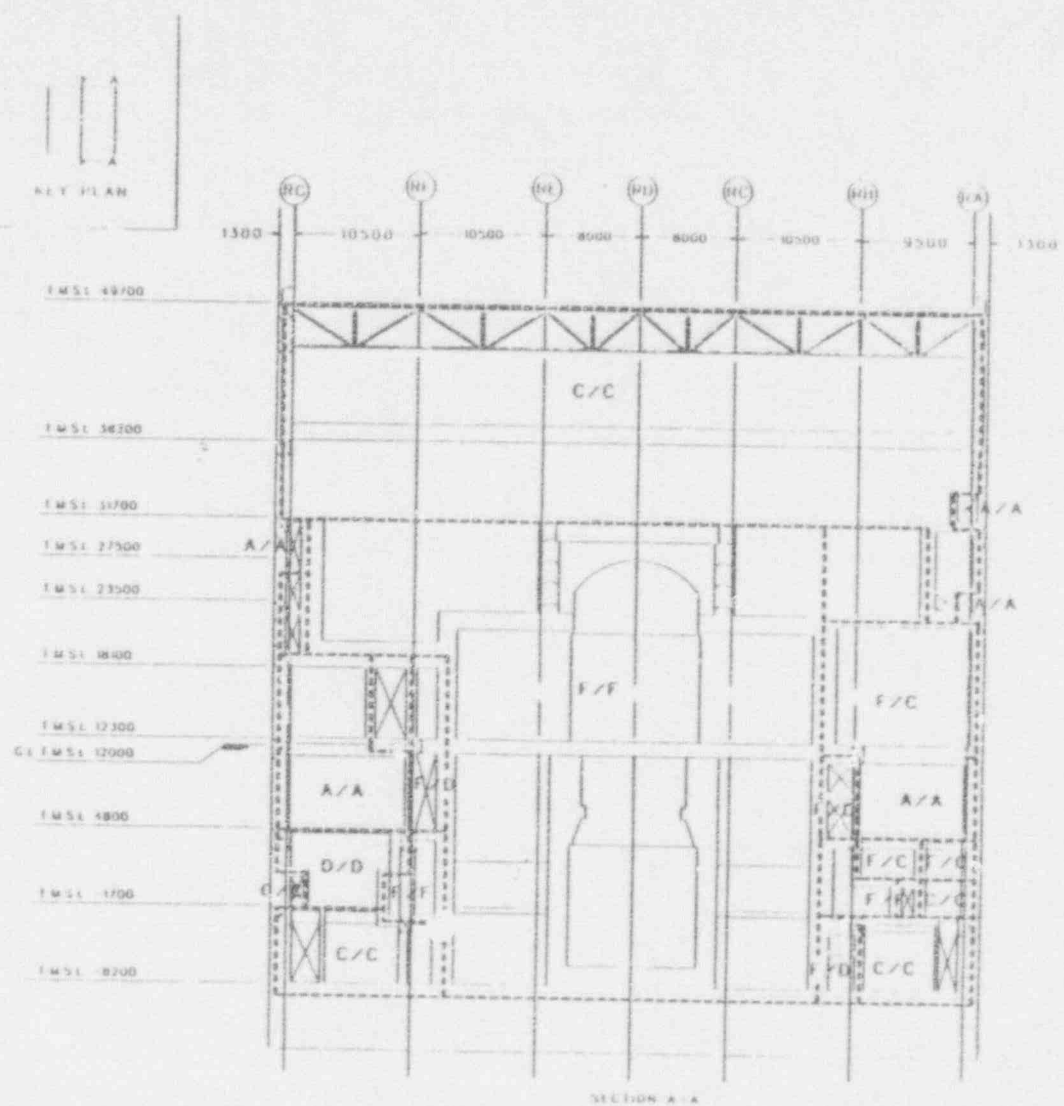
FIGURE 3.7h REACTOR BUILDING
RADIATION ZONE MAP FOR
FULL POWER AND SHUTDOWN OPERATIONS
AT ELEVATION 34500



FULL POWER/SHUTDOWN
RADIATION LEVELS IN
mRem/hour

A ≤ 0.6
B ≤ 1
C ≤ 5
D ≤ 25
E ≤ 100
F > 100

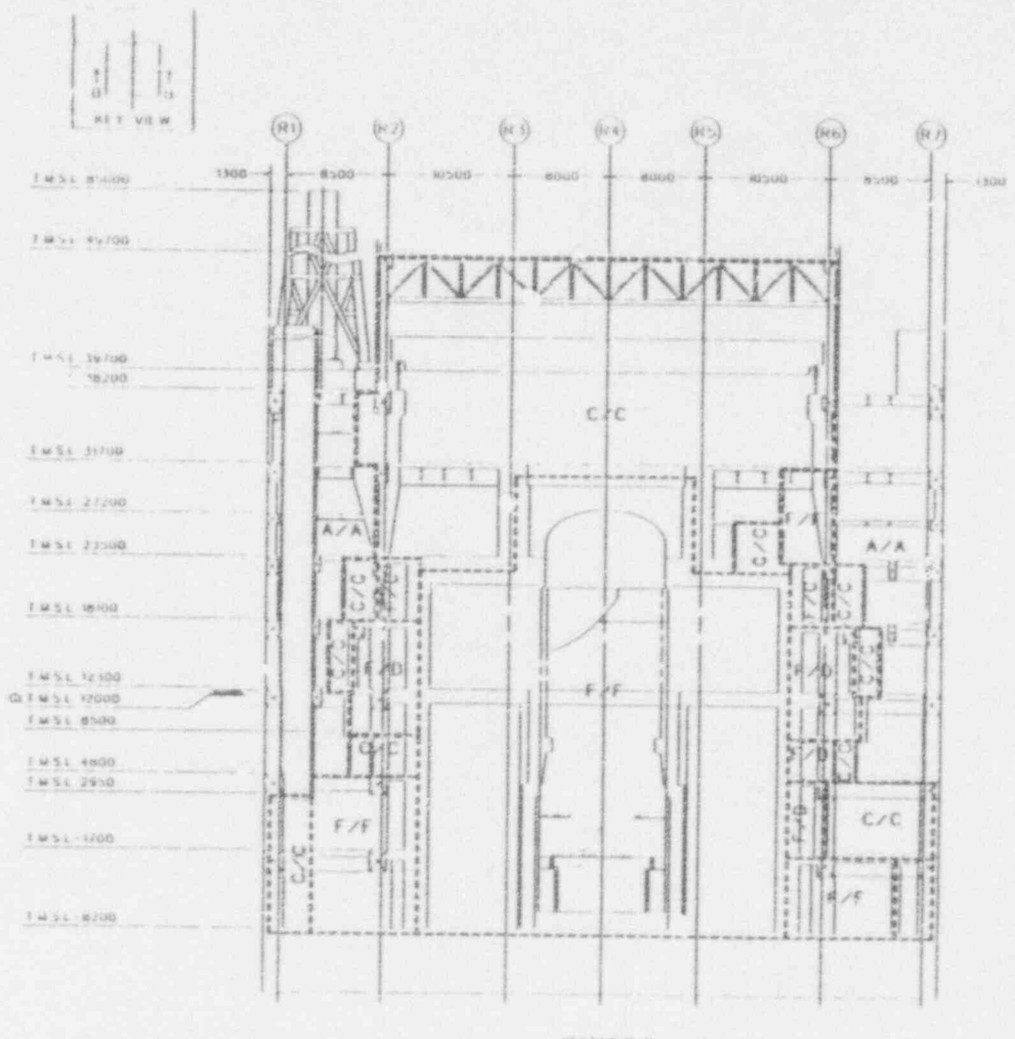
FIGURE 3.7-[illegible] REACTOR BUILDING
RADIATION ZONE MAP FOR
FULL POWER AND SHUTDOWN OPERATIONS
AT ELEVATION 31700 MM



FULL POWER/SHUTDOWN
RADIATION LEVELS IN
mRem/hour

A ≤ 0.6
B < 1
C < 5
D < 25
E < 100
F ≥ 100

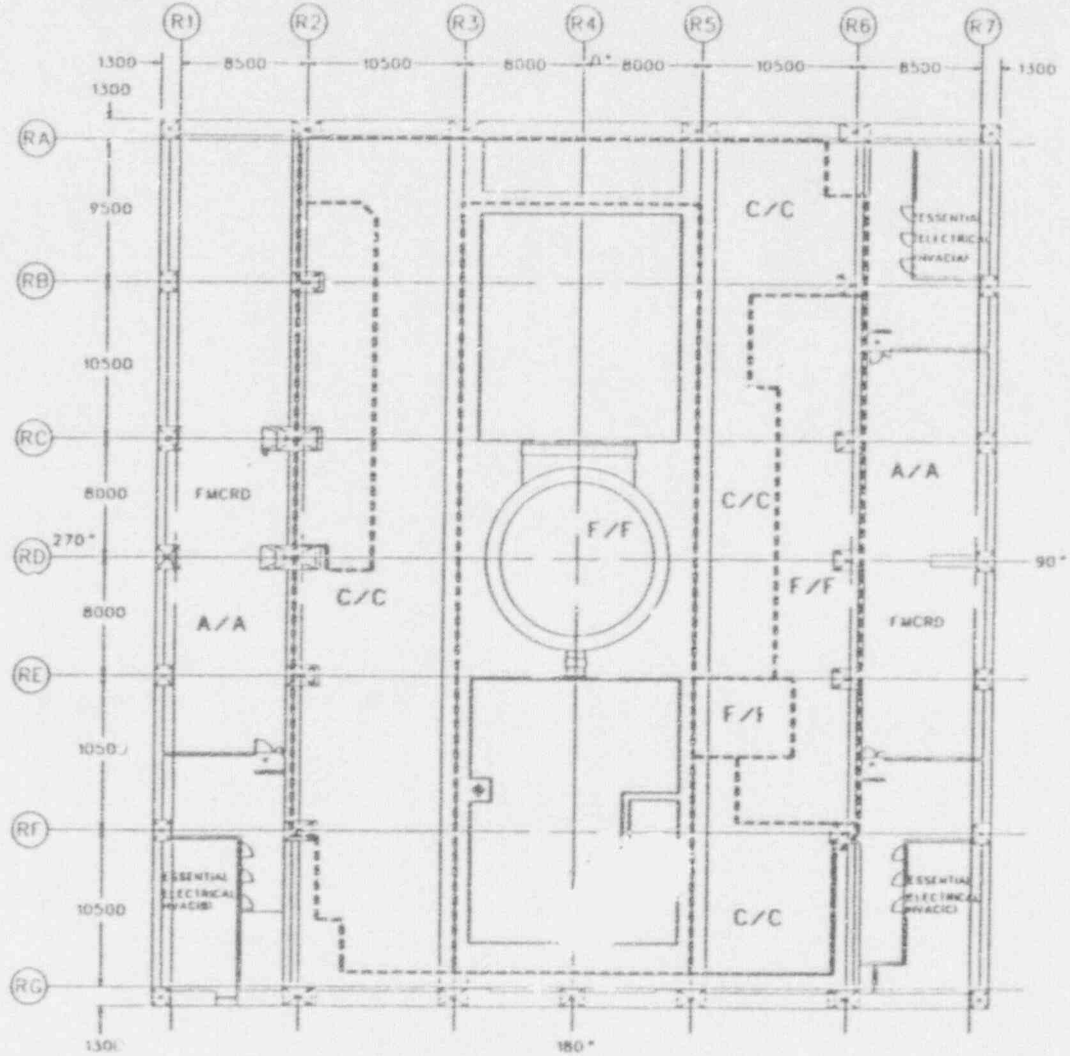
FIGURE 3.7j REACTOR BUILDING
RADIATION ZONE MAP FOR
FULL POWER AND SHUTDOWN OPERATIONS
CROSS SECTION VIEW A-A



FULL POWER/SHUTDOWN
RADIATION LEVELS IN
mRem/hour

A ≤ 0.6
B < 1
C < 5
D < 25
E < 100
F ≥ 100

FIGURE 3.7x REACTOR BUILDING
RADIATION ZONE MAP FOR
FULL POWER AND SHUTDOWN OPERATIONS
CROSS SECTION VIEW B-B



FULL POWER/SHUTDOWN
RADIATION LEVELS IN
mRem/hour

A ≤ 0.6
B < 1
C < 5
D < 25
E < 100
F ≥ 100

FIGURE 3.79 REACTOR BUILDING
RADIATION ZONE MAP FOR
FULL POWER AND SHUTDOWN OPERATIONS
AT ELEVATION 27200 (4F)

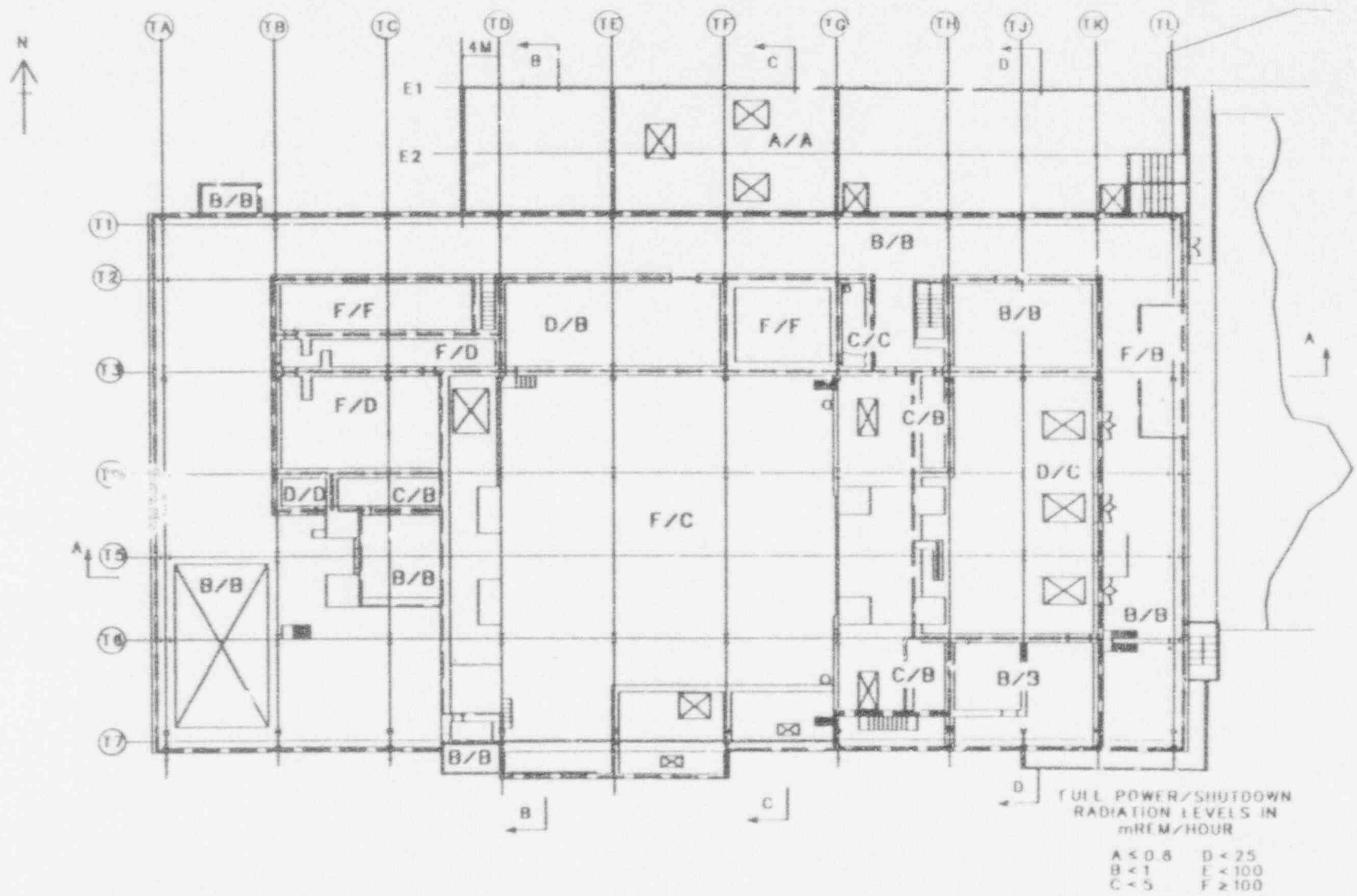


FIGURE 3.7m TURBINE BUILDING RADIATION ZONE AT ELEVATION TMSL 12.3M

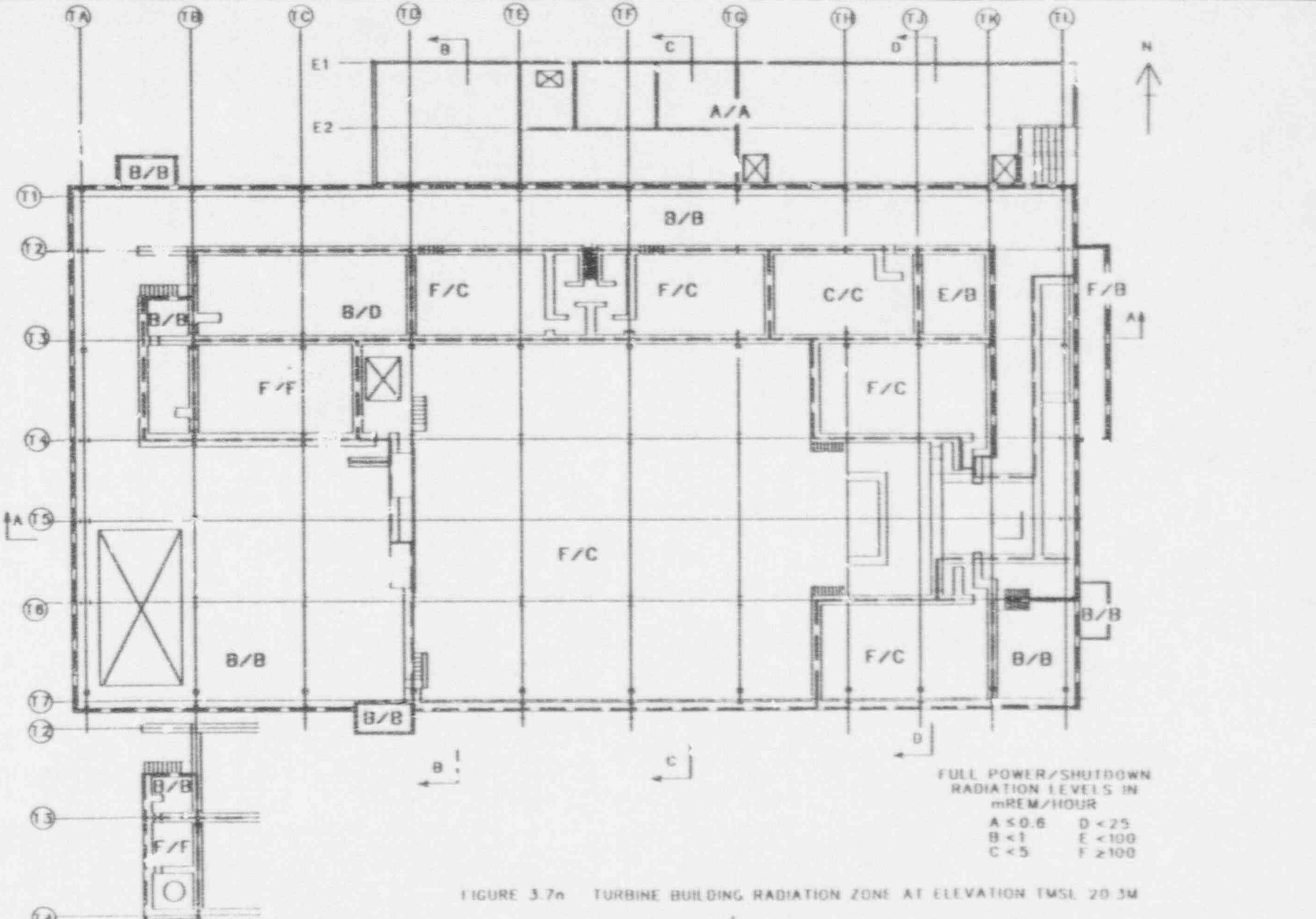
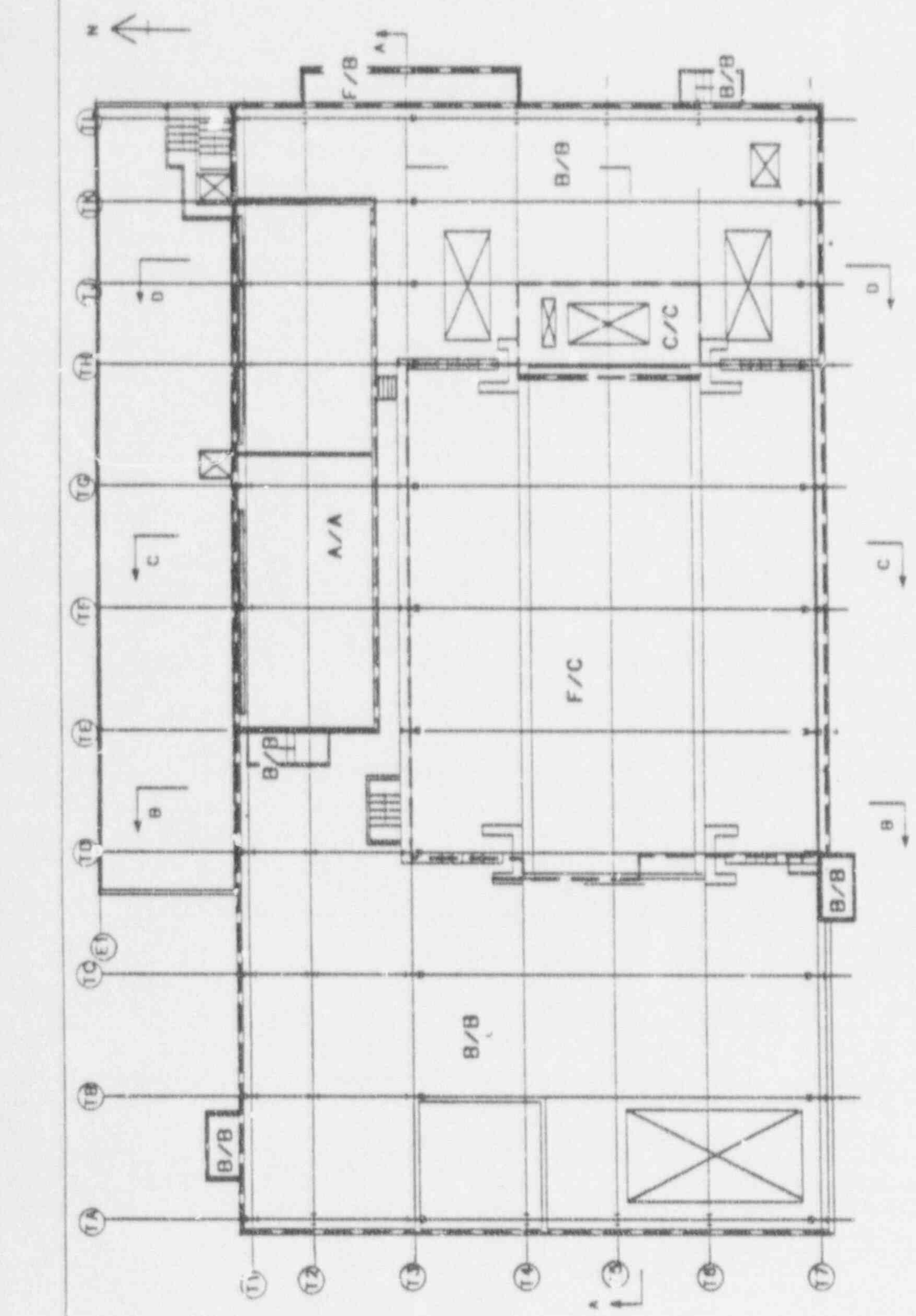


FIGURE 3.7n TURBINE BUILDING RADIATION ZONE AT ELEVATION TMSL 20.3M



FULL POWER/SHUTDOWN RADIATION LEVELS IN mREM/HOUR

A ≤ 0.6	D < 25
B < 1	E < 100
C < 5	F ≥ 100

FIGURE 3.7o TURBINE BUILDING RADIATION ZONE AT ELEVATION TMSL 30.3M

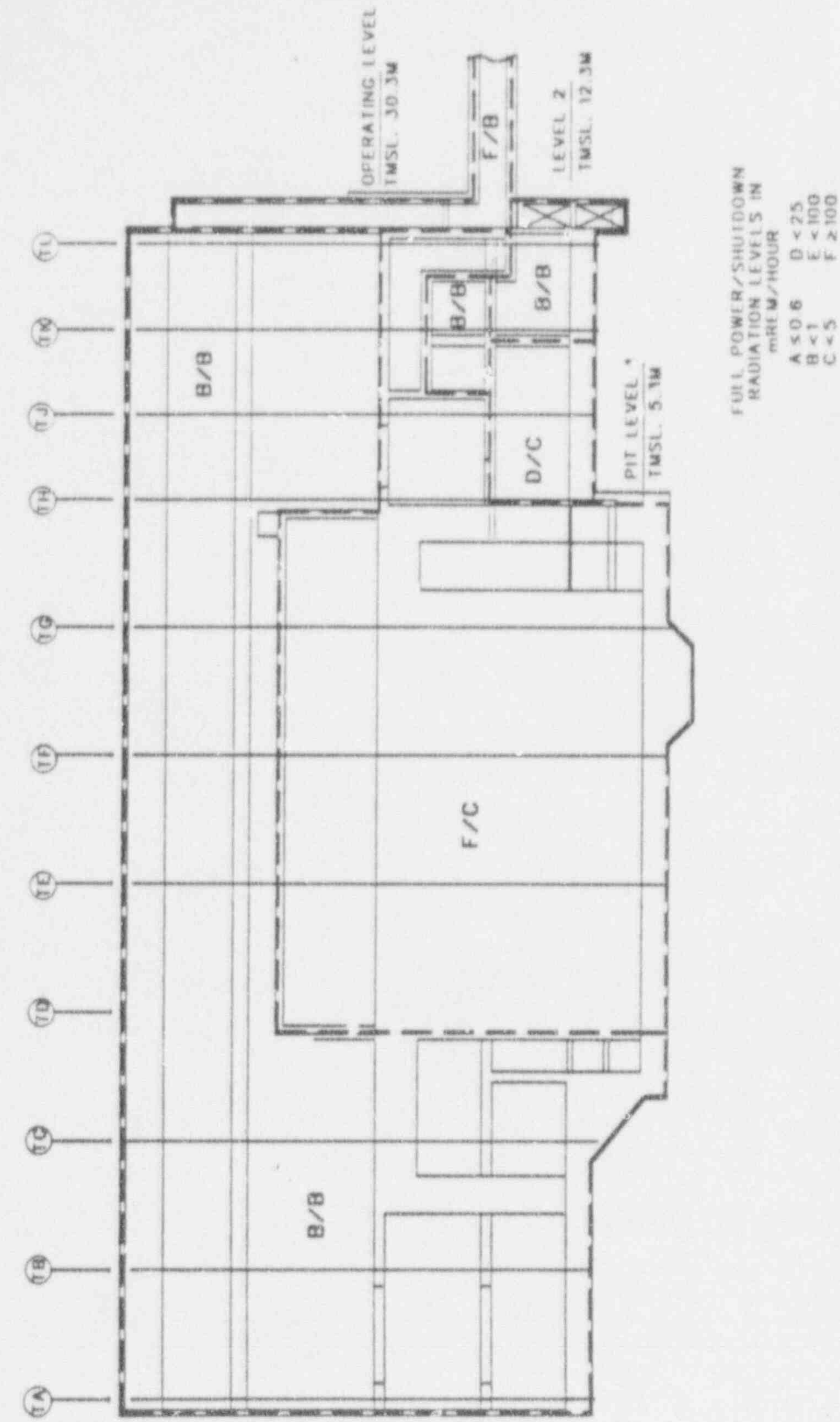


FIGURE 3.7p TURBINE BUILDING RADIATION ZONE AT NORMAL OPERATION LONGITUDINAL SECTION AA

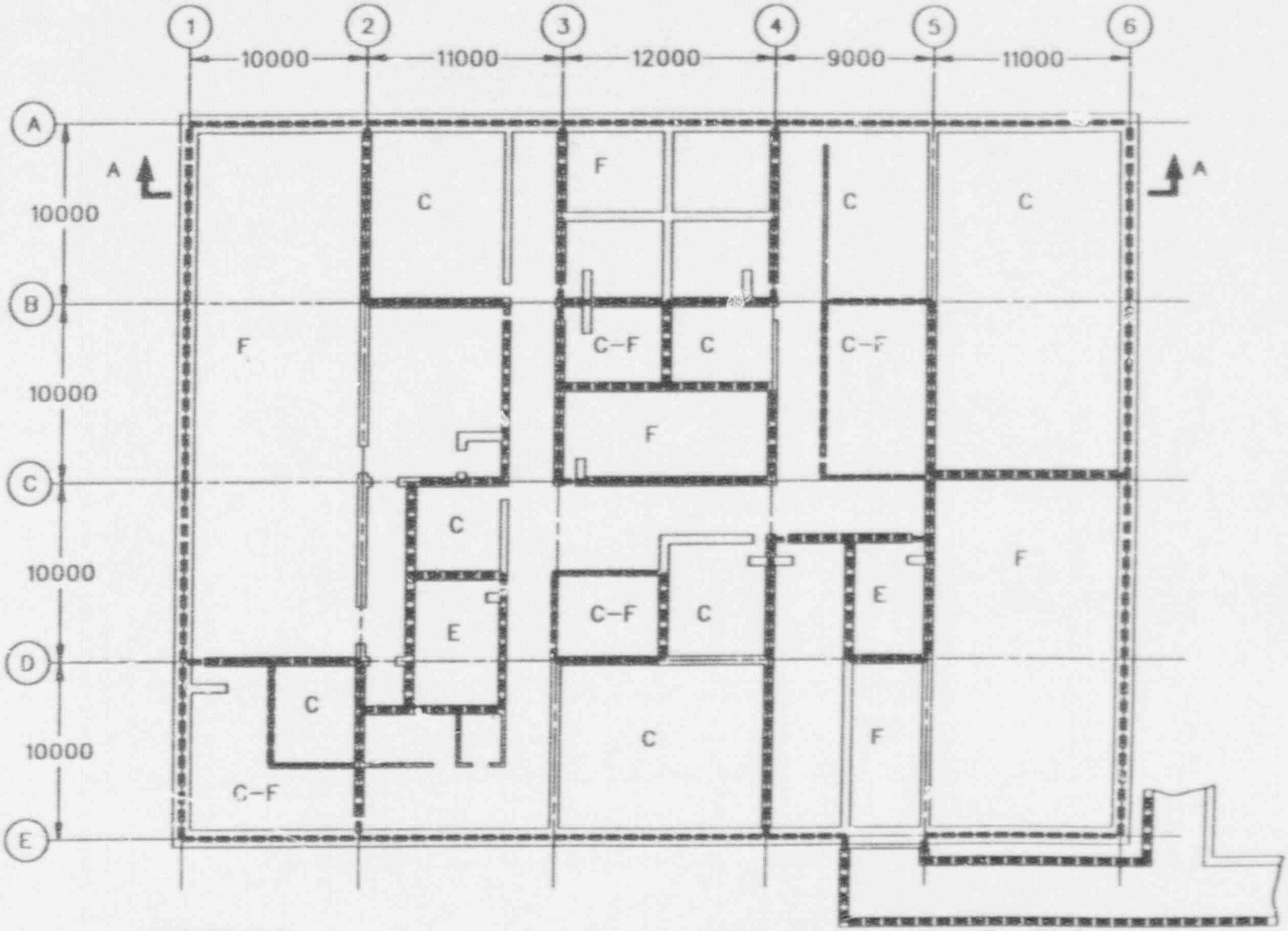


FIGURE 3.7q

RADWASTE BUILDING, RADIATION ZONE MAP,
NORMAL OPERATION AT ELEVATION (–) 6500mm

A ≤ 0.6mrem/hr D < 25.0mrem/hr
B < 1.0mrem/hr E < 100mrem/hr
C < 5.0mrem/hr F ≥ 100mrem/hr

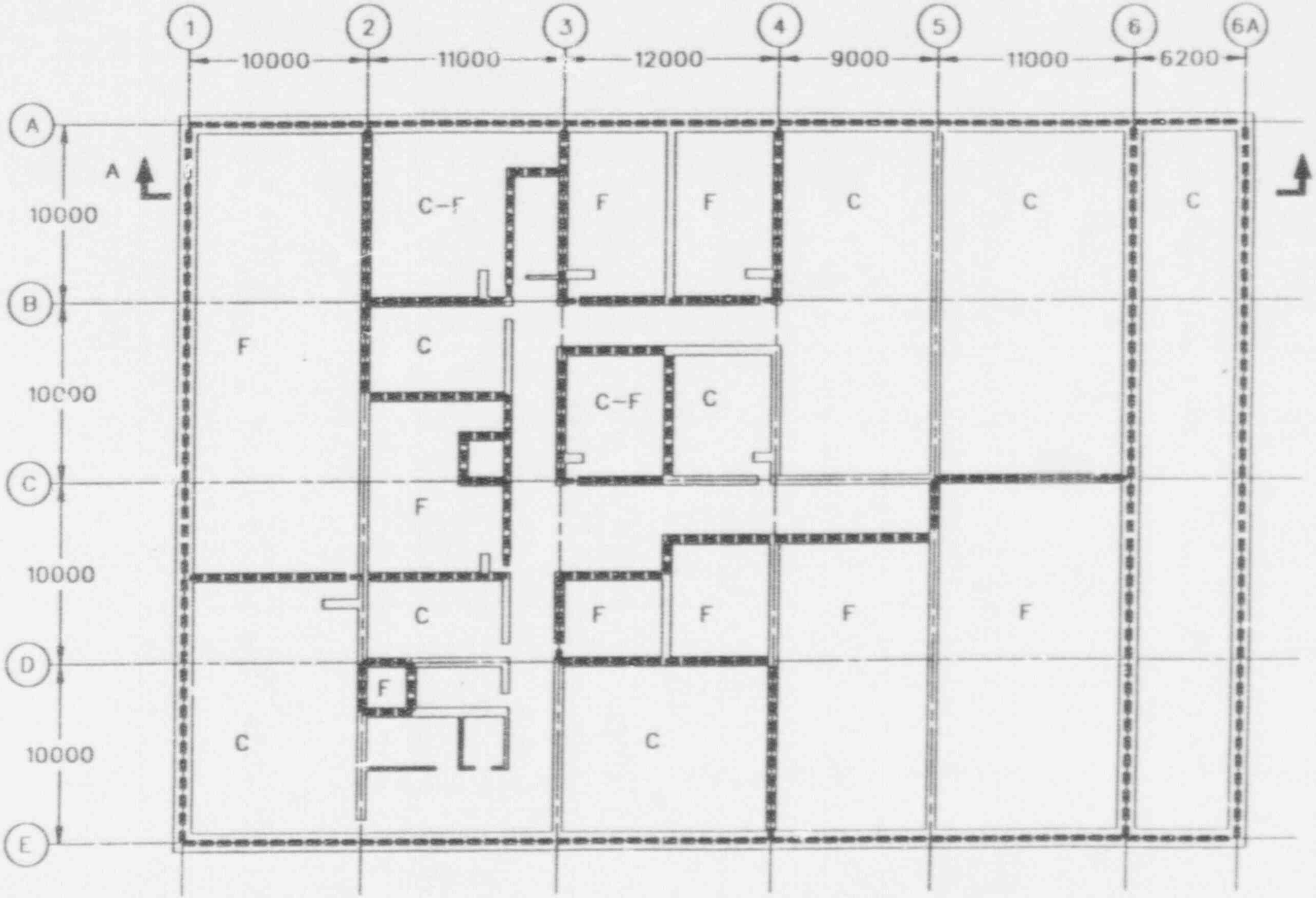


FIGURE 3.7r

RADWASTE BUILDING, RADIATION ZONE MAP, NORMAL OPERATION AT ELEVATION (-) 200mm

A ≤ 0.6mrem/hr
B < 1.0mrem/hr
C < 5.0mrem/hr
D < 25.0mrem/hr
E < 100mrem/hr
F ≥ 100mrem/hr

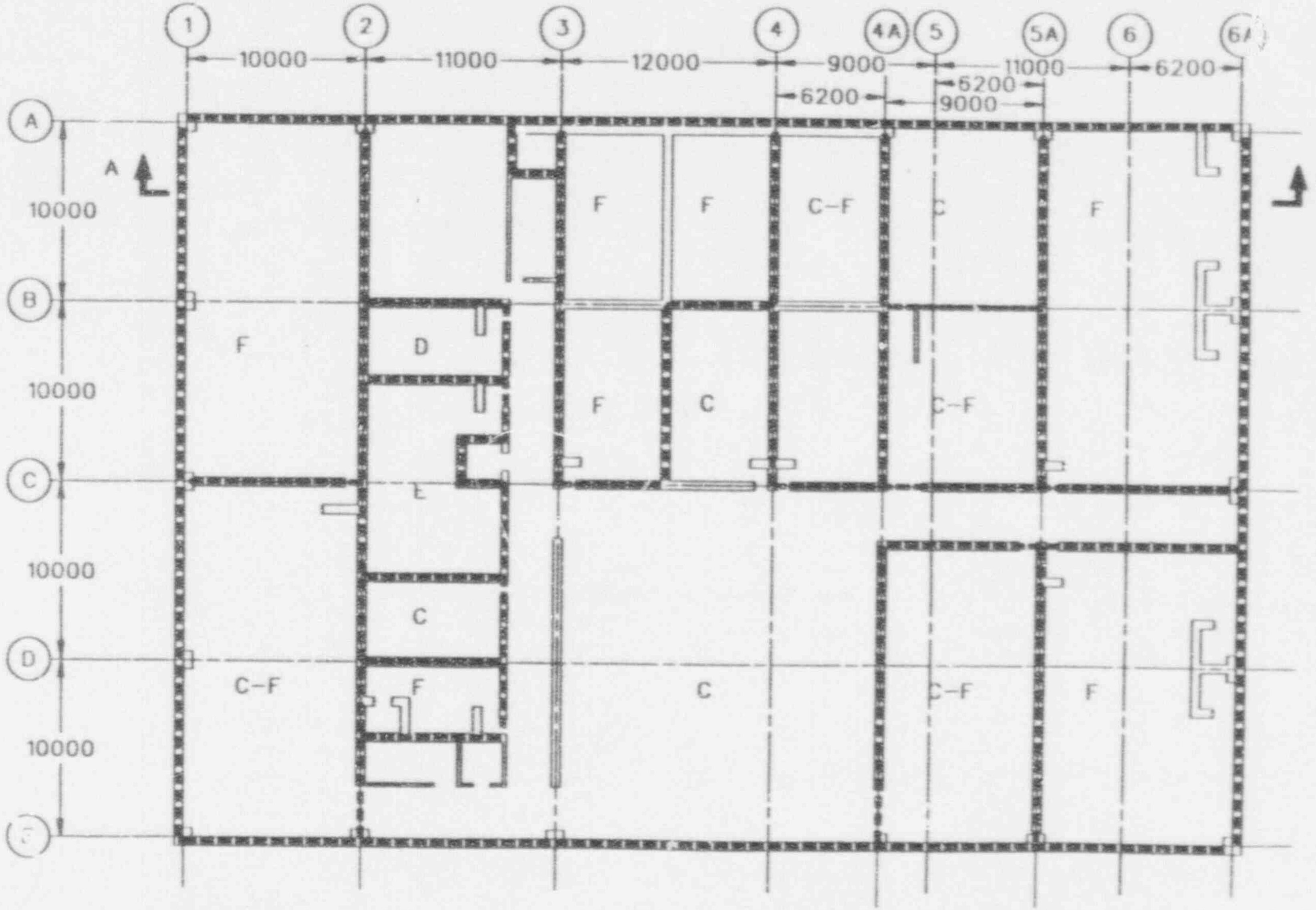


FIGURE 3.7s

RADWASTE BUILDING, RADIATION ZONE MAP,
NORMAL OPERATION AT ELEVATION 7300mm

A ≤ 0.6mrem/hr
B ≤ 1.0mrem/hr
C < 5.0mrem/hr
D < 25.0mrem/hr
E < 100mrem/hr
F ≥ 100mrem/hr

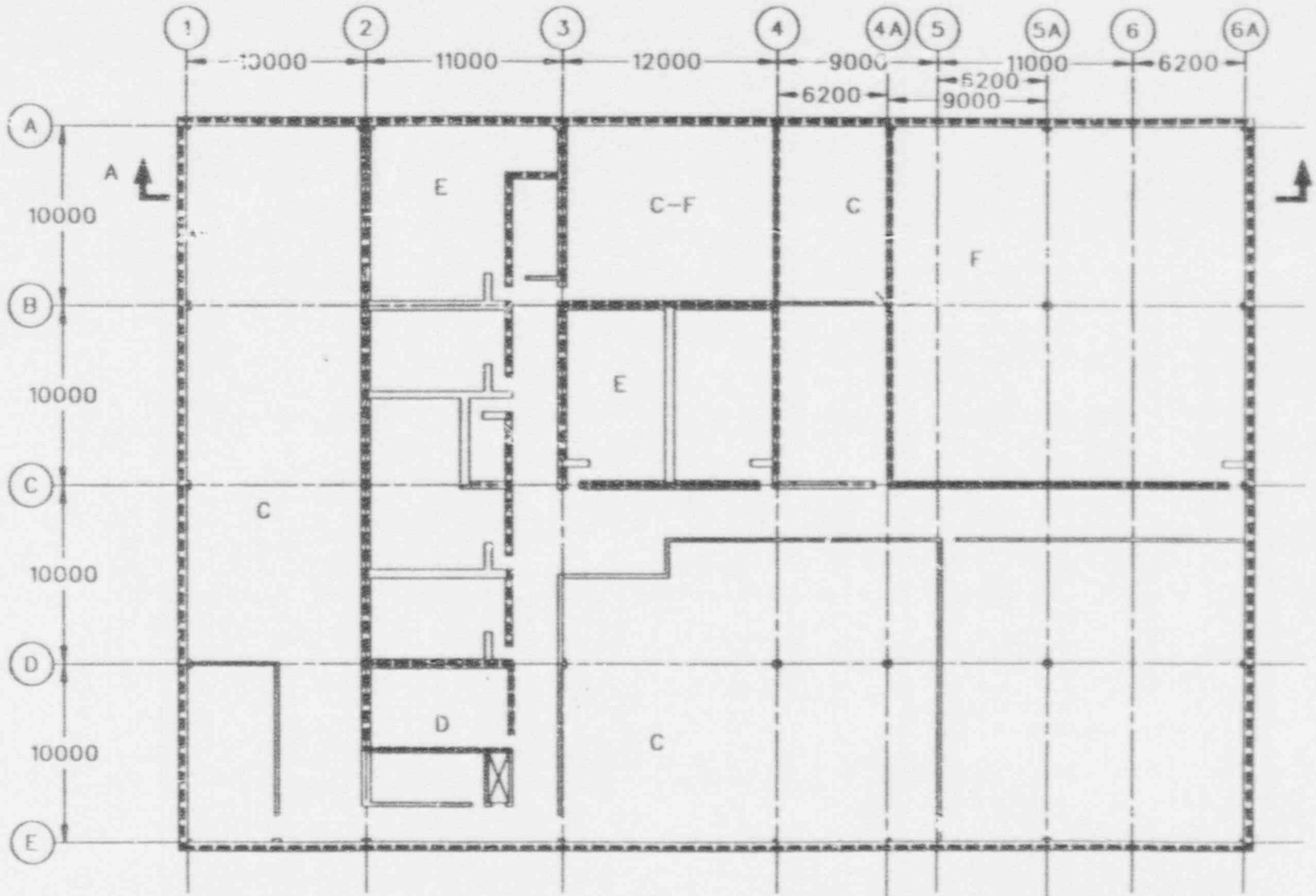


FIGURE 3.7t

RADWASTE BUILDING, RADIATION ZONE MAP,
NORMAL OPERATION AT ELEVATION 16000mm

A ≤ 0.6mrem/hr D < 25.0mrem/hr
B < 1.0mrem/hr E < 100mrem/hr
C < 5.0mrem/hr F ≥ 100mrem/hr

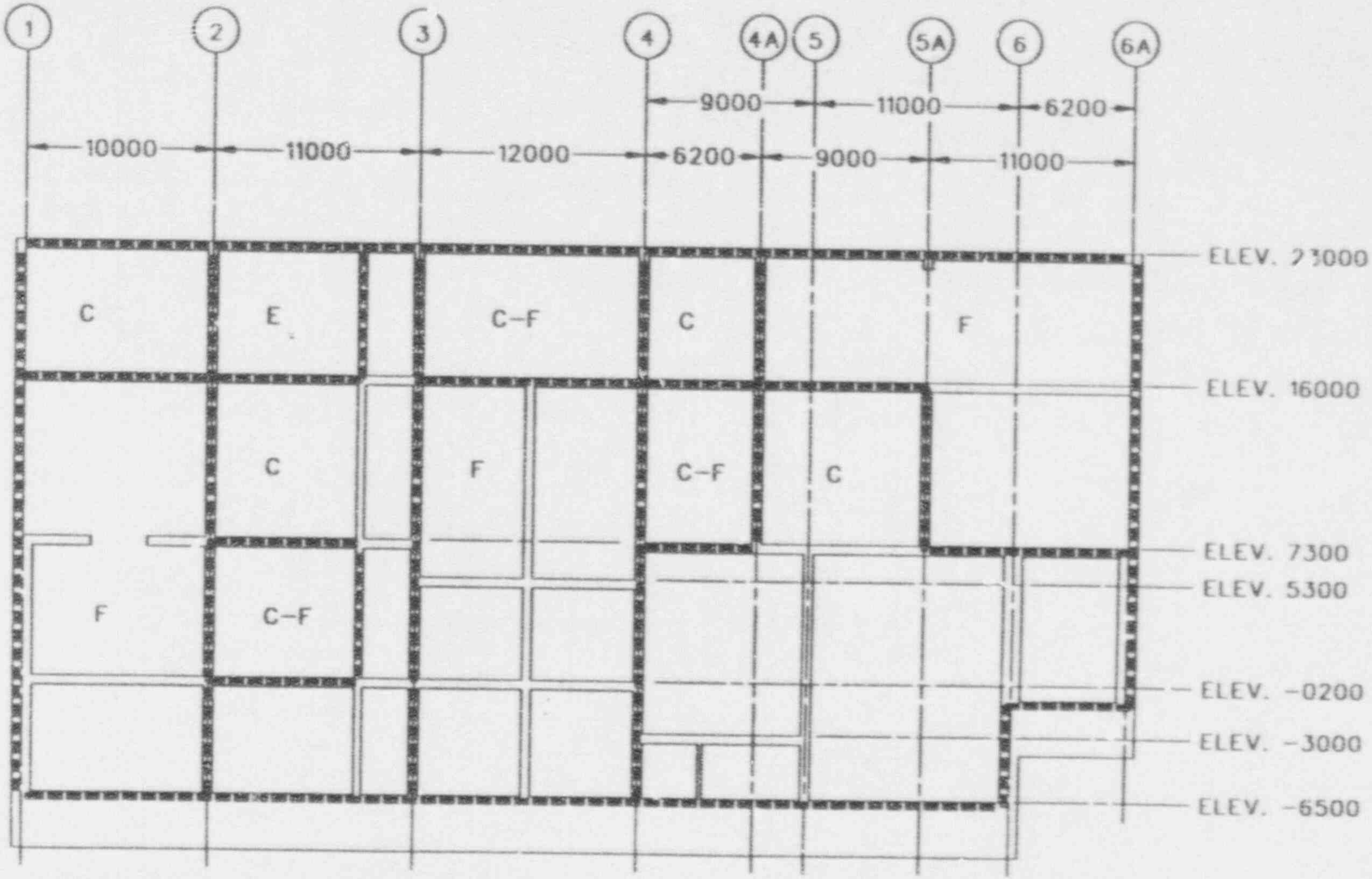
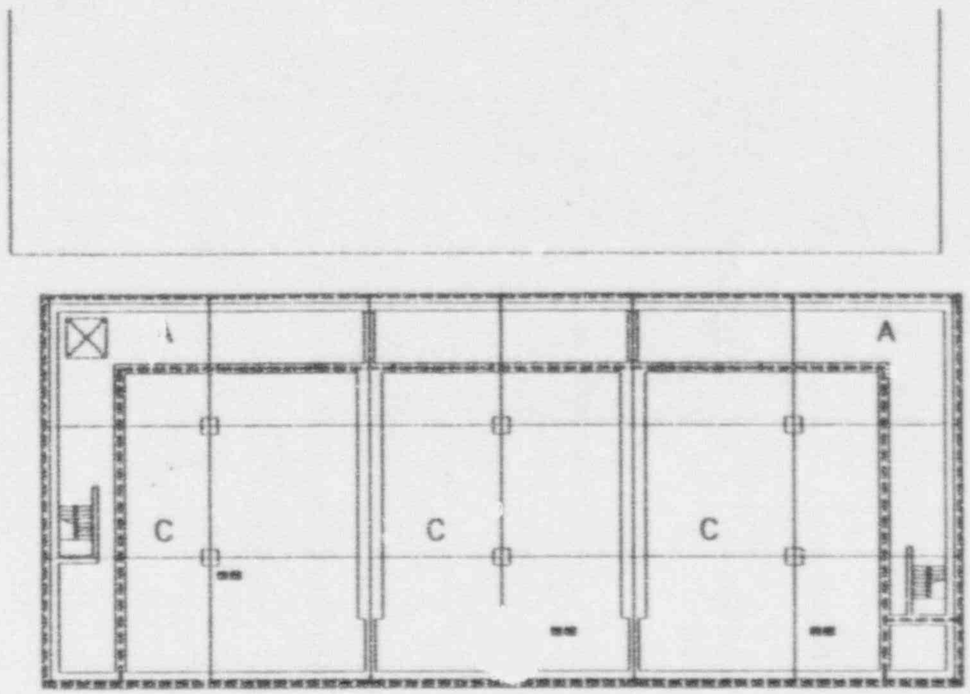


FIGURE 3.7u

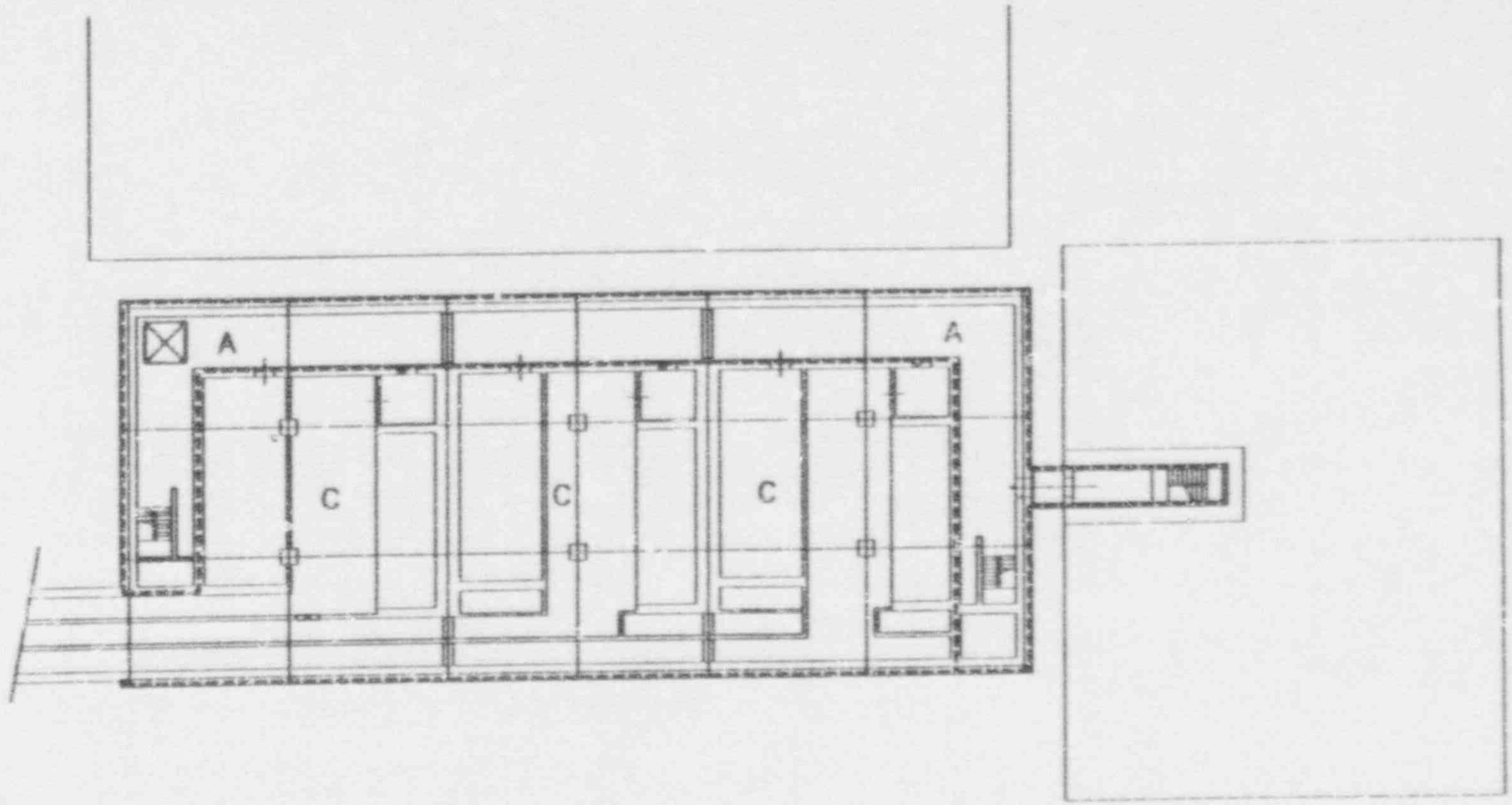
RADWASTE BUILDING, RADIATION ZONE MAP, NORMAL OPERATION AT CROSS SECTION A–A

A ≤ 0.6mrem/hr D < 25.0mrem/hr
B < 1.0mrem/hr E < 100mrem/hr
C < 5.0mrem/hr F ≥ 100mrem/hr



A ≤ 0.8 mREM/HOUR
B < 1.0 mREM/HOUR
C < 5.0 mREM/HOUR
D < 25.0 mREM/HOUR
E < 100.0 mREM/HOUR
F ≥ 100.0 mREM/HOUR

FIGURE 3.7v CONTROL BUILDING RADIATION ZONE, NORMAL OPERATION AT FLOOR LEVEL TMSL 18200MM



A ≤ 0.6 mREM/HOUR
B < 1.0 mREM/HOUR
C < 5.0 mREM/HOUR
D < 25.0 mREM/HOUR
E < 100.0 mREM/HOUR
F ≥ 100.0 mREM/HOUR

FIGURE 3.7w CONTROL BUILDING RADIATION ZONE, NORMAL OPERATION AT FLOOR LEVEL TMSL (-12150MM

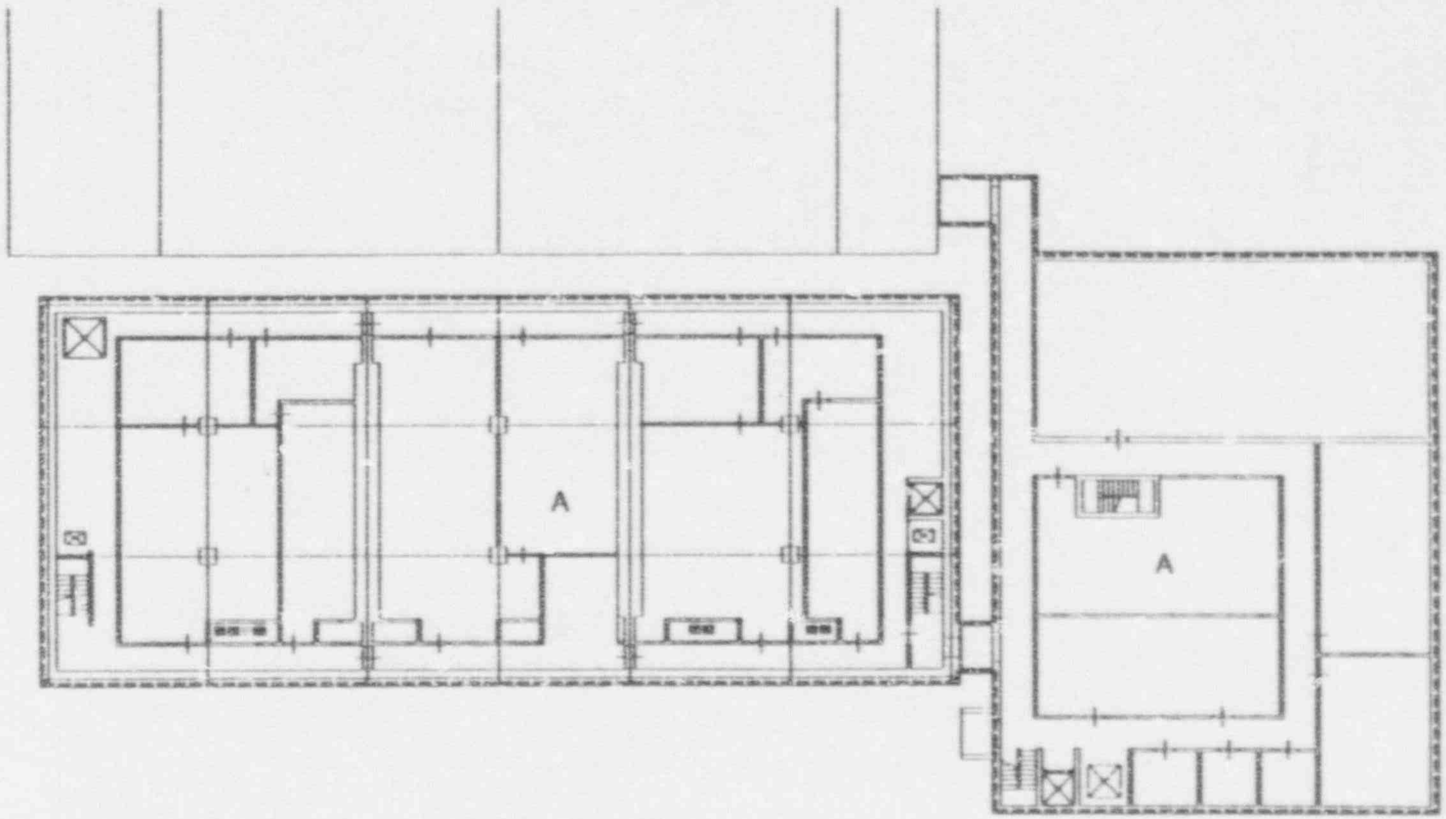
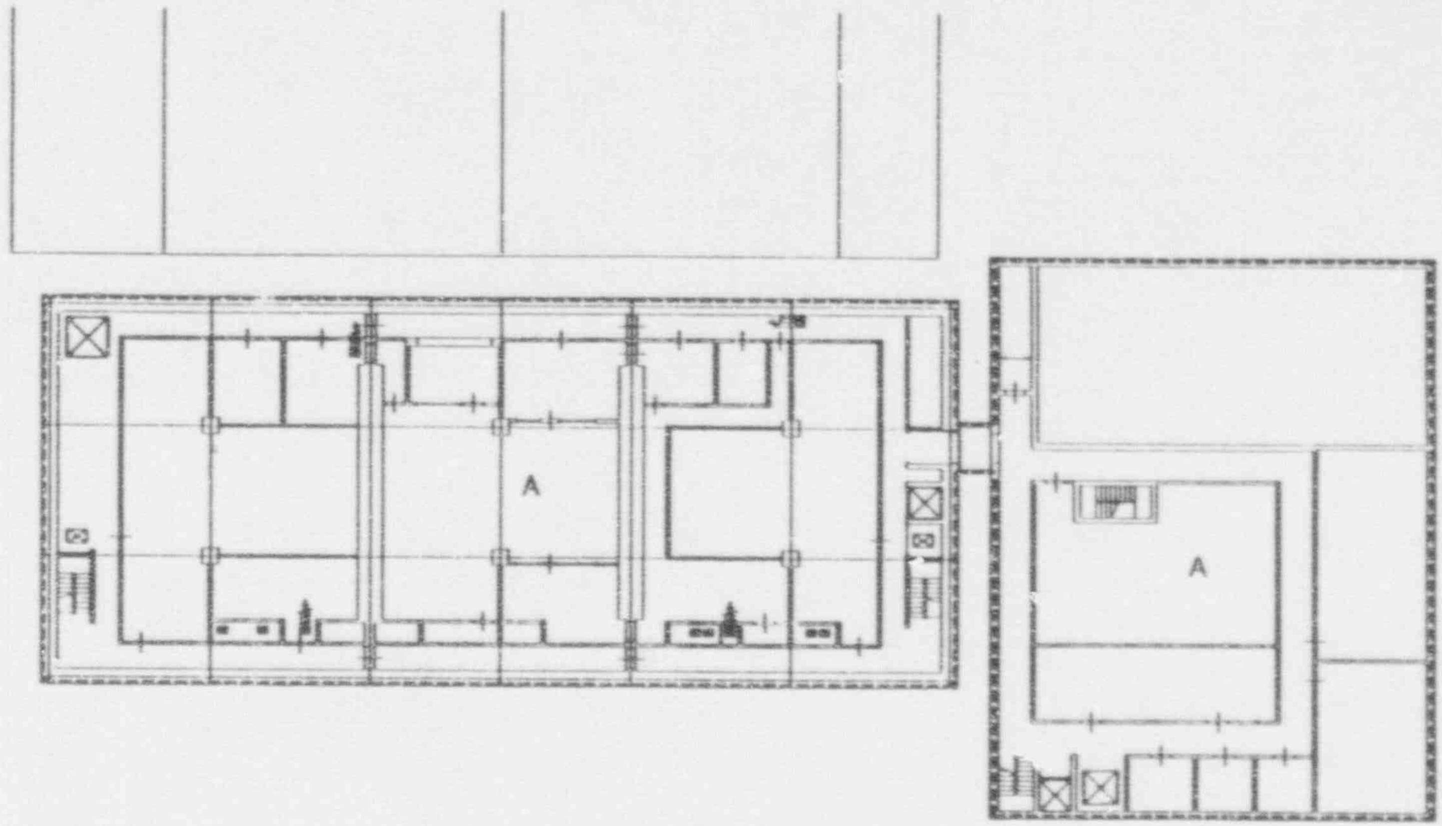
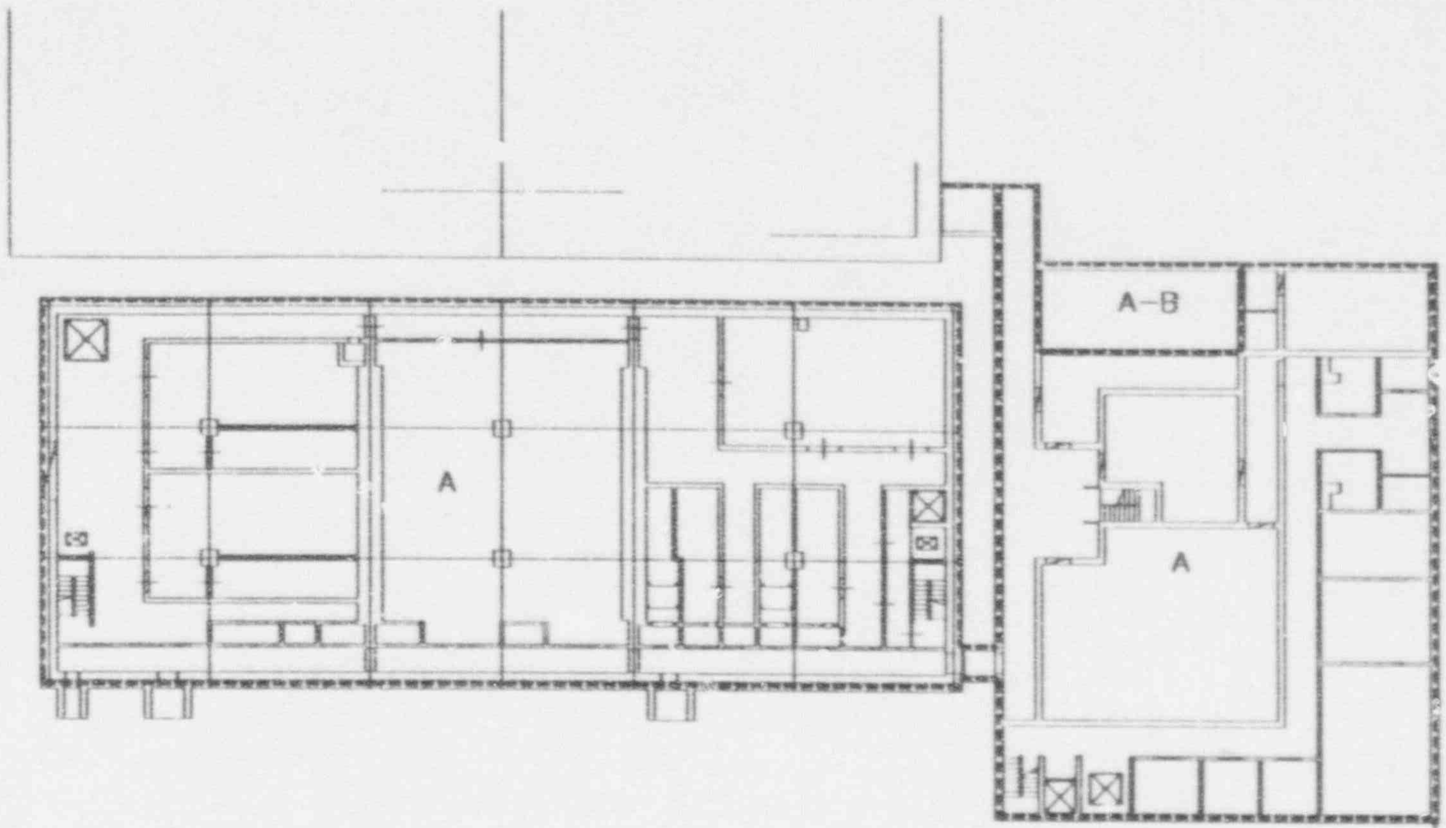


FIGURE 3.7x CONTROL BUILDING RADIATION ZONE, NORMAL OPERATION AT FLOOR LEVEL TMSL 3500MM



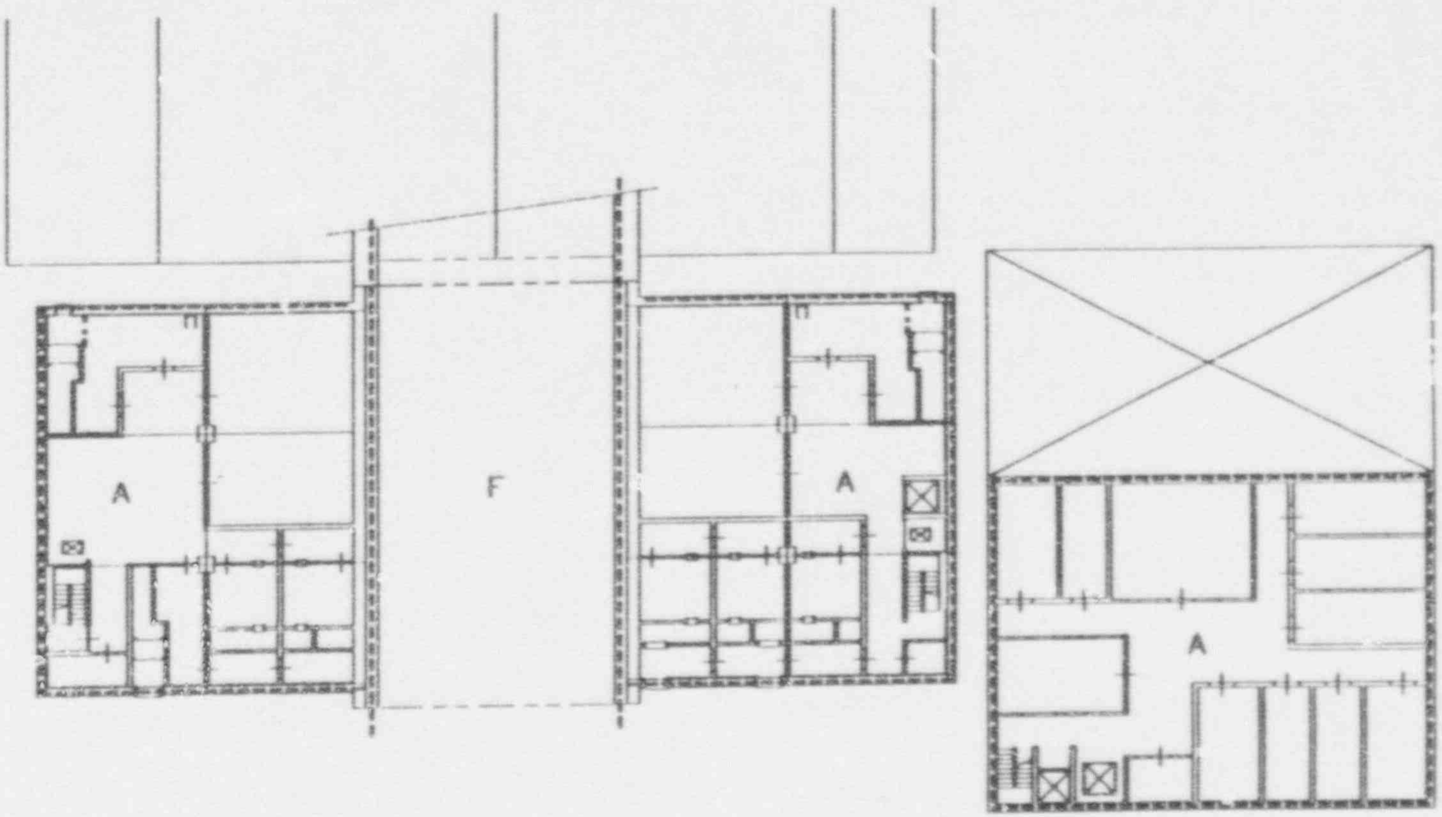
A ≤ 0.6 mREM/HOUR
B < 1.0 mREM/HOUR
C < 5.0 mREM/HOUR
D < 25.0 mREM/HOUR
E < 100.0 mREM/HOUR
F ≥ 100.0 mREM/HOUR

FIGURE 3.7y CONTROL BUILDING RADIATION ZONE, NORMAL OPERATION AT FLOOR LEVEL TMSL 7900MM



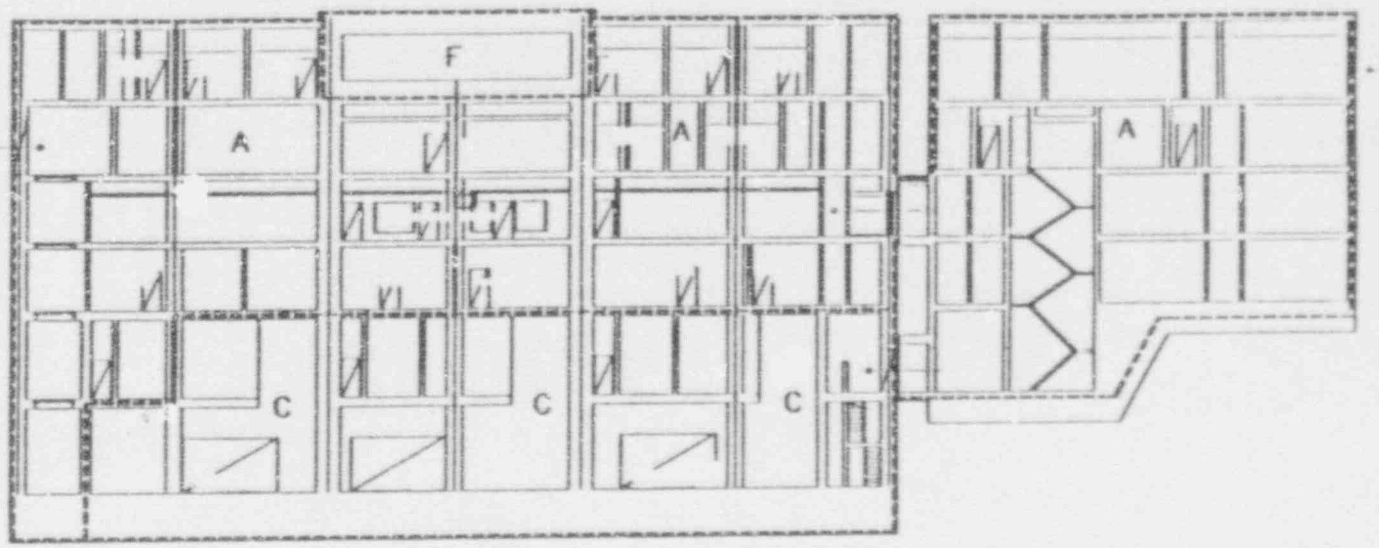
A ≤ 0.6 mREM/HOUR
B < 1.0 mREM/HOUR
C < 5.0 mREM/HOUR
D < 25.0 mREM/HOUR
E < 100.0 mREM/HOUR
F ≥ 100.0 mREM/HOUR

FIGURE 3.7z CONTROL BUILDING RADIATION ZONE, NORMAL OPERATION AT FLOOR LEVEL TMSL 12,300MM



A ≤ 0.6 mREM/HOUR
B < 1.0 mREM/HOUR
C < 5.0 mREM/HOUR
D < 25.0 mREM/HOUR
E < 100.0 mREM/HOUR
F ≥ 100.0 mREM/HOUR

FIGURE 3.7aa CONTROL BUILDING RADIATION ZONE, NORMAL OPERATION AT FLOOR LEVEL TMSL 17,150MM



A ≤ 0.6 mREM/HOUR
B < 1.0 mREM/HOUR
C < 5.0 mREM/HOUR
D < 25.0 mREM/HOUR
E < 100.0 mREM/HOUR
F ≥ 100.0 mREM/HOUR

FIGURE 3.7bb CONTROL BUILDING RADIATION ZONE, NORMAL OPERATION, SIDE VIEW

12.2 Radiation Sources

The staff has audited the contained sources and airborne radioactive material source terms provided in Section 12.2 and Chapter 11 of the ABWR SSAR for completeness against the guidelines in Regulatory Guide (RG) 1.70, and against the criteria set forth in Section 12.2 of NUREG-0800 (SRP). The contained source terms are used as the basis for designing radiation protection features (including radiation shielding calculations) and for personnel dose assessment. Airborne radioactive source terms are used in the design of ventilation systems and personnel dose assessment. The staff review consisted of ensuring that GE had either committed to following the criteria of RGs and staff positions contained in Section 12.2 of NUREG-0800 or provided acceptable alternatives. In addition, the staff selectively compared source terms for specific systems against those used for plants of similar design. The staff's review indicates that source term descriptions in the SSAR are not of sufficient detail to allow the staff to perform confirmatory calculations of the shielding effectiveness nor confirm that the plant ventilation and airborne radioactive material monitoring systems are adequate to protect personnel. Therefore, the SSAR does not meet the criteria of RG 1.70 and NUREG-0800.

At the current stage in the ABWR design, GE is not able to describe many in-plant radiation sources to the level of detail specified in RG 1.70 and NUREG-0800. As an alternative, GE has provided design acceptance criteria (DAC) to ensure that these confirmatory calculations are performed prior to plant start up. The source term information contained in the SSAR and DAC is consistent with the acceptance criteria in NUREG-0800. Therefore, the staff concludes that the information provided by the applicant with respect to radiation sources is acceptable (pending final acceptance of the DAC as discussed in section 12.2.2) and meets the requirements of 10CFR Part 20, 10CFR Part 50 subsection 50.34(f) and GDC 61. Details of the review follow.

12.2.1 Contained Sources and Airborne Radioactive Material Sources

GE's description of radioactive sources in the ABWR are provided in Chapters 11 and 12 of the SSAR. Section 11.1 provides information on the radioactive source terms in reactor water and steam. Section 12.2 provides descriptions of plant components that become significant sources of radiation during plant operations, including shutdown. Sources of airborne radioactive material are discussed in Section 12.2.2 of the SSAR.

During power operations, the greatest potential for personnel radiation dose is inside the primary containment (drywell) due to

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nitrogen-16, noble gases, reactor neutrons and prompt gamma. Nitrogen-16 is also a significant source of radiation from the steam and condensate systems outside of the drywell during power operations. In other areas outside of the drywell, and inside the drywell after shutdown, the primary sources of personnel radiation exposure are the fission products from fuel clad defects, and the activation products that are transported to, and deposited in, plant systems and components. The estimates of fission and activation products concentrations in the ABWR systems containing reactor water are based on ANSI/ANS-18.1, adjusted using the assumptions in RG 1.112. Allowances are included for the buildup of activated corrosion and wear products based on operating experience of reactors of similar design. Neutron and prompt gamma source terms are based on reactor core physics calculations. The accident source terms are based on NUREG-0737 NRC "Clarification of TMI Action Plan Requirements".

The DSER identified several deficiencies in the SSAR description of the contained radioactive source terms for ABWR. These deficiencies were the omission of sources inside the drywell and in the turbine building, missing description of sources in post-accident vital areas, and insufficient source characterization. GE has amended section 12.3.5 of the SSAR to indicate that the post-accident sources of concern in plant vital areas are limited to gamma radiation shine from the reactor building and the radioactive material contained in the post-accident coolant and effluent monitoring systems. GE has also amended the tables of source terms in section 12.2 to include sources inside the drywell and turbine building. GE has provided tables of nominal source strengths based on expected system configuration and approximate component geometry. As discussed in section 12.2.2 of this FSER, the actual source strengths for contaminated reactor system components is dependent on as-built specifications. Therefore, the source terms used in confirmatory shielding calculations will be determined as part of the shielding DAC. These SSAR changes are acceptable to the staff.

Almost all of the airborne radioactivity within the plant is due to equipment leakage. As discussed below, the leakage of contaminated fluids from system components can not be quantified at this stage in the ABWR design. GE has proposed a DAC to determine the airborne source terms in each room and operating area of the plant.

12.2.3 ITAAC

The level of design detail in the SSAR does not provide system layouts within their rooms or cubicles. Information concerning the type and size of components in these systems is also not provided. Without this "as-built" or "as-procured" information, source term parameters needed to calculate radiation shielding for these systems or the concentrations of airborne radioactive material in the rooms, cannot be provided as specified in the SRP. As an alternative, GE has provided design acceptance criteria (DAC) that require the COL holder to determine these source term parameters during the ITAAC phase of plant construction. These DAC are discussed in section 12.3.5 of this FSER. The criteria in each DAC describe the bases for the source term, consistent with the acceptance criteria in NUREG-0800, used in its analysis. Compliance with these DAC, supplemented with the information in sections 11.1, 12.2, and 12.3 of the SSAR, will demonstrate that the ABWR design can meet the requirements of 10CFR Part 20, as they relate to the evaluation of radiation sources,and the related provisions of 10CFR Part 50, GDC 61, as supplemented by the guidance of RG 1.112, NUREG-0737 and ANSI/ANS 18.1. Therefore , pending a review of the final DAC, the staff finds this acceptable.

12.2.3 Interfaces

Section 12.2.3 of the SSAR identifies two issues as plant interfaces. It is the staff's position that these issues are incorporated in the DAC discussed in section 12.2.2.above. During a conference call, following the February 27,1992 meeting on plant interfaces, GE agreed to amend the SSAR and appropriately characterize these issues. This is a confirmatory issue pending a review of the amended SSAR.

12.3 Radiation Protection Design

The staff has audited the facility design features, shielding, ventilation, and radiation and airborne monitoring instrumentation contained in the ABWR SSAR for completeness against the guidelines in RG 1.70 and against the criteria set forth in NUREG-0800, Section 12.3. The staff review consisted of ensuring that GE had either committed to following the criteria of RGs and staff positions referenced in Section 12.3 of NUREG-0800, or provided acceptable alternatives. In addition, the staff selectively reviewed GE's SSAR against the acceptance criteria of the SRP using the review procedures in NUREG-0800.

The staff concludes that GE has not demonstrated that the ABWR design can meet the relevant requirements of 10CFR Part 20, 10 CFR Part 50, and General Design Criteria 19 and 61 in all areas of the plant. Details of the staff's review follow.

12.3.1 Facility Design Features

GE has provided evidence that radiation dose accumulating tasks (maintenance, refueling, radioactive material handling, in-service inspection, decommissioning, and accident recovery) have been considered in the plant design. Several features, as discussed above (see Section 12.1.2), have been included in the design to help maintain doses ALARA. These features will facilitate access

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to work areas, reduce or allow the reduction of source intensity, reduce the occupancy requirements in high radiation fields, and provide for portable shielding, remote-operation and instrumentation for radioactive systems. These ABWR features are consistent with the guidance of RG 8.8 (Rev. 3) and NUREG-0800 and are acceptable to the staff.

GE has provided drawings of the plant layout which indicate radiation zones used in the plant design. The six radiation zones provide a basis for classifying occupancy and access restrictions for various areas within the plant during normal operations and accident conditions. Maximum design dose rates are established for each zone and used as the basis for shielding of the respective zones. This method of plant zoning is consistent with the guidance in RG 1.70 and NUREG-0800 and is acceptable to the staff.

The DSER identified several deficiencies related to the Chapter 12 figures 12.3-1 through 12.3-73 that depict plant radiation zones (during normal operations, normal shutdown and accident conditions) and area radiation monitor locations. GE has amended the SSAR to provide more legible figures for the reactor, control and radwaste buildings. These updated figures also indicate the normal controlled and uncontrolled access routes to the plant as well as the access/egress route to plant vital areas under accident conditions. On April 13, and May 1, 1992, GE provided draft revised copies of the reactor and turbine building figures. The revised figures resolved the inconsistencies between the turbine building figures noted in the DSER. This is a confirmatory item pending a review of an corresponding amendment to the SSAR.

Several features are included in the ABWR design to minimize the buildup of activated corrosion and wear products, a major contribution to occupational doses. These features include a reduction in cobalt bearing components used in reactor systems (activated cobalt is a major contributor to plant radiation levels) and pre-filming of reactor systems prior to plant operation, to minimize activated material deposition on system interior surfaces. Main condenser tubes and tube-sheets will be made of titanium alloys to minimize the introduction of foreign material into the reactor system (which become activated and/or promote corrosion) resulting from condenser tube leakage. Other features such as the use of seamless piping, the use of straight through valve design wherever possible, the use of butt-welded piping connections, and the use of back-flushing connections on instrument lines, minimize build-up of radioactivity in plant piping systems.

The DSER contained an open item concerning the provision in the ABWR design to facilitate chemical decontamination of heat exchangers in systems that carry radioactive water. On April 9, 1992, GE provided a draft SSAR amendment that indicates that separate connections are provided on the reactor water cleanup non-regenerative and regenerative heat exchangers. Heat exchangers in the RHR system and the RIP cooling heat exchangers are provided with fittings by which they can be flushed with clean water.

GE's corrosion product control features are consistent with the guidance in RG 8.8 (Rev. 3) and NUREG-0800 and are acceptable to the staff. This is a confirmatory item pending a review of the SSAR amendment consistent with the April 9, 1992 memorandum.

The ABWR is designed such that operation will not require an application for alternate high radiation area controls (per 10 CFR 20.203(c)(5)), as experienced with current operating BWRs. The design provides that all high radiation areas (greater than 100 mrem/hr) are maintained locked to control unauthorized access and no credit is taken for the relief provided in Section 12.6 of the BWR Standard Technical Specifications (i.e., locked area at 1000 mrem/hr). This design position is acceptable to the staff.

12.3.2 Shielding

The objective of the plant's radiation shielding is to provide protection against radiation exposure for personnel, both inside and outside the plant, during normal operation, including abnormal operational occurrences (AOOs), and during reactor accidents. All radioactive sources are provided with shielding based on access and exposure level requirements consistent with the designed radiation zoning. Concrete used for radiation shielding meets the NRC guidance provided in RG 1.69. Shielding calculations were performed by GE with the QAD-F, GGG and DOT.4 computer codes. These are commonly accepted shielding calculational codes and are therefore acceptable to the staff.

The thickness of specific radiation shields have not been provided by GE, in accordance with the guidance of RG 1.70 and the acceptance criteria of NUREG-0800. GE's position is that since the system layouts and the physical dimensions of the radioactive system components are not known, the shielding requirements for these systems cannot be provided at this stage of the ABWR design. Therefore the staff cannot perform confirmatory calculations of shielding effectiveness. As an alternative method, GE has provided DAC to verify the adequacy of the ABWR shielding design. The staff's review of these DAC is discussed in section 12.3.5.1 below. This alternative is acceptable to the staff. This is a confirmatory issue pending the review and acceptance of the final DAC.

An open item identified in the DSER is the adequacy of the shielding in the upper drywell. The biological shield surrounding the reactor vessel (depicted in Figures 12.3-23 and 24) does not cover a significant portion of the top of the reactor vessel. As noted in Section 12.1, a fuel handling mishap resulting in dropping a SFA across the reactor flange is a significant radiological hazard in BWRs. In addition to the radiological hazard presented

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by this AOO, it appears that raising an irradiated fuel bundle in proximity of the vessel wall could result in significant radiation dose rates in the upper drywell. On July 29, 1991, GE provided details of a proposed design change to the shielding in the upper drywell. This design change would raise the biological shield, to within four inches of the upper drywell ceiling. The staff's evaluation of this proposal indicated that the revised design would provide sufficient shielding during the normal withdrawal of SFAs form the reactor. However, a dropped SFA resting across the reactor flange would still produce significant radiation streaming into the upper drywell. Personnel in the upper drywell during this AOO could still receive lethal radiation doses before they could escape. The staff concludes that the ABWR design as described in the SSER ,as revised by the July 29,1991 memorandum, is inadequate to ensure radiation protection during this event, and is not acceptable. During a management meeting held on March 25-26, 1992 in San Jose, GE committed to revise the upper drywell shielding to resolve this issue.

This remains an open item pending a review of the revised design.

The DSER also identified an open item concerning the shielding of the TIP system. As discussed in 12.1.2 above TIP drive and storage are located in separate shielded rooms. However, the conduit that guides the TIP from the reactor to its storage, is virtually unshielded. This conduit shares the primary containment penetration with the lower drywell personnel access. Personnel located at the lower drywell access hatch, or in the access tunnel, would be exposed to the activated TIP and drive cable as they are retracted from the reactor core. On March 26,1992, GE provided a draft SSAR amendment that discusses the radiation design features associated with the TIP system. This amendment notes that the lower drywell access is located in a separate shielded room that can be locked to prevent access to these areas while the TIP is being withdrawn from the core. In addition, flashing alarms at the door to this room and at the lower drywell access hatch are provided to warn personnel when power is applied to the TIP drives. Also, the TIP system operates such that TIP withdraw is in the high speed mode which will minimize the transit time of the activated components through the unshielded portions of the system. These features ensure that the personnel radiation exposures resulting from the operation of the TIP system can be maintained ALARA, and are acceptable to the staff. This is a confirmatory item pending a review of the SSAR amendment consistent with the March 26, 1992 memorandum.

12.3.3 Ventilation

The ABWR ventilation systems are designed to protect personnel and equipment from extreme environmental conditions and ensure that plant personnel are not inadvertently exposed to airborne contaminants exceeding the concentration limits given in 10 CFR 20. Design features intended to maintain personnel exposures ALARA

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include:

1) Airflow between areas potentially having airborne contamination is always from the area of lower potential contamination to the area of higher potential contamination;

2) The appropriate use of negative or positive pressure in areas to prevent exfiltration or infiltration of possible airborne contamination respectively;

3) A dual fresh air intake system for the control room ventilation designed such that at least one intake is free of contamination following a LOCA accident.

These design features are in accordance with the guidelines of RG 8.8 (Rev. 3) and are acceptable to the staff. However, as noted in Section 12.2, the expected leakage of radioactive fluids from plant systems cannot be determined at t' s stage of the ABWR design. Without this source term, GE is .t able to provide the concentrations of airborne contamination in cubicles, rooms, and corridors as specified in the SRP. Therefore, the staff cannot verify that the plant ventilation system design meets the criteria in the SRP. As an alternative, GE has provided DAC (on April 6, 1992) that requires the COL holder to calculate the expected concentrations of airborne radionuclides, for each normally occupied plant area (i.e., areas that are accessed for maintenance and operation of the plant). Section 12.3.5.2 contains the staff's evaluation of this DAC. In addition, on May 1,1992, GE provided a draft proposed appendix to SSAR Ch. 12 that describes the calculational methods and assumptions used to determine these airborne concentrations of radionuclides. These calculational methods and assumptions are consistent with the provisions of NUREG-0800. Therefore, the staff finds them acceptable.

12.3.4 Area Radiation and Airborne Radioactive Monitoring Instrumentation

The DSER identified an open item concerning the description of the ABWR area radiation monitoring system. GE has revised the SSAR with the following information. The ABWR Area Radiation Monitoring system consists of twenty five gamma sensitive detectors and their associated digital monitors. The detectors are provided in key locations of the plant and have operating ranges (sensitivity) commensurate with the expected radiation levels in the areas. These area monitors are powered from the non-1E vital 120 Vac bus. The monitors have adjustable alarm settings (both up-scale and down-scale), with local audible alarms to warn personnel of abnormal conditions such as higher than normal radiation levels or detector failure. Although the RHR equipment areas are not vital areas, as specified in NUREG-0737, high range monitors that meet the criteria of RG 1.97 are provided in the unlikely event that personnel have to enter these areas under accident condi ions. In addition, four

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high range gamma sensitive ion chambers are provided in the primary containment that can measure up to $3.6X10^{8}$ C/Kg/sec. (10^{7} R/hr.) for assessing the magnitude of the release of radioactive material from the core during an accident. The staff concludes that the area radiation monitoring system meets the applicable criteria in RG 8.8, RG 1.97 and the provisions in item II.F.1.3 of NUREG-0737 and is therefore acceptable. COL applicants will be required to address the operational considerations, such as monitor alarm set points, listed in RG 1.70 section 12.3.4.

The DSER notes that the ABWR design does not provide criticality accident monitors that meet the requirements of 10CFR70.24 as provided in the SRP. In response to the staff's request, GE has amended the SSAR to indicate that these monitors are not necessary since ABWR is designed to ensure subcritical conditions during fuel handling and storage. Several operating BWRs have cited similar design features, and a commitment to certain fuel handling and storage procedures, as a basis for requesting a license condition exempting them from the 10CFR70.24 requirement. The requirements of 10CFR Part 70 are outside the scope of this review. COL applicants will be required to provide information showing that their plant meets the requirements of 10CFR70.24 or request an exemption .

Monitoring of airborne radioactive materials in nuclear power plants typically provided by fixed continuous air monitors, that sample the ventilation air exhausted from plant areas having the highest potential for radioactivity release, supplemented with movable continuous air monitors that are positioned in plant areas that have a potential for airborne radioactivity release during certain operating modes (i.e., area where a radioactive system is opened during maintenance). GE has not provided a description of the airborne monitoring for the ABWR design. As discussed in section 12.3.3 above, the expected concentrations of airborne radionuclides can not be determined due to the current level of the ABWR design. As an alternative, GE has provided DAC that would require the COL holder to verify that airborne monitors provided in the final ABWR design meet the criteria of the SRP.The staffs review of these DAC is in section 12.3.5 below.

12.3.5 ITAAC

The staff's review has identified three areas where the level of detail in the SSAR does not allow the staff to concluded that the ABWR design meets the acceptance criteria in Ch.12 of the SRP. These areas are the adequacy of the plant radiation shielding, the adequacy of the plant ventilation system and the adequacy of the plant airborne radioactivity monitoring system. As an alternative, GE has provided design acceptance criteria (DAC) that require shielding analysis and airborne radionuclide concentration calculations be performed by the COL holder during the ITAAC stage of plant construction and verify that the final ABWR design is acceptable. Details of the staff's review of these DAC follow.

12.3.5.1 Plant Shielding DAC

Chapter 12 of the SSAR contains layout drawings of the plant that indicate the designed maximum radiation levels (or zones) for each room, equipment cubicle, and operating space during normal power operations, shutdown operations and accident conditions. As discussed in section 12.2 above, the piping layout and component selection have not been set for the ABWR systems. Parameters (such as source strength and geometry) needed to verify the adequacy of the radiation shields around these systems, as specified in the SRP, are not available. In addition, nitrogen-16 gammas from the plant can be a significant contributor to off-site dose rates. The adequacy of the plant shielding needed to comply with the public dose limits in 40 CFR Part 190 can not be verified since site specific parameters (such as distance to the site boundary) are unknown.

The staff has reviewed the Plant Shielding DAC provided in table 3.7.a of the "Tier 1 Design Certification Material for the GE ABWR Design Stage 2 Submittal",submitted April 6,1992. This DAC requires analysis be performed to verify the adequacy of 1) the shielding around rooms and spaces during normal operations and shutdown conditions, 2) the shielding, and temporary shield space, provided between plant systems during maintenance activities, 3) the shielding provided around vital plant areas during accident conditions and 4) the plant shielding needed to limit public dose.the staff's review indicates that the analysis assumptions, methods and acceptance criteria in this DAC are consistent with the criteria in the SRP The staff concludes that compliance with these DAC, as supplemented by the information contained in section 12.3.2 of the SSAR, meets the relevant requirements of 10CFR Part 20, 10CFR Part 50, GDC 19 and 61, as supplemented by the guidance of RG 8.8 and NUREG-0737, and is therefore acceptable.

12.3.5.2 Ventilation and Airborne Monitoring DAC

Due to the level of detail in the current ABWR design, the expected airborne concentrations in rooms and operating areas within the plant cannot be provided as specified in RG 1.70. Therefore, the staff has not been able to verify that the plant ventilation nor airborne radionuclide monitoring meet the criteria of the SRP.

The staff reviewed the Ventilation and Airborne Monitoring DAC provided in table 3.7.b of the "Tier 1 Design Certification Material for the GE ABWR Design Stage 2 Submittal",submitted April 6,1992. This DAC requires the COL holder to calculate the expected concentrations of airborne radioactivity in each equipment cubicle, corridor, and operating area that require personnel access. In addition this DAC requires that an analysis be preformed to identify those areas of the plant that require continuous monitoring of airborne radioactive materials. The staff's review indicates that the assumptions and acceptance criteria in these DAC are consistent with the criteria in NUREG-0800. The staff concludes that compliance with these DAC, as supplemented by the information in sections 12.3.3, 12.3.4 and Appendix 12A of the SSAR, meets the requirements in 10CFR Part 20, 10CFR Part 50, and GDC 61, with respect to the control of to airborne radioactive materials, the provisions for maintaining personnel exposure to airborne radionuclides ALARA in RG 8.8, and the requirements in 10CFR Part 20, 10CFR Part 50, and GDC 64 related to in-plant monitoring during normal operations. Therefore, the staff finds them acceptable.

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SECTION 12.3

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ILLUSTRATIONS (continued)

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ILLUSTRATIONS (continued)

SECTION 12.3

ILLUSTRATIONS (continued)

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 Facility Design Features

The ABWR Standard Plant is designed to meet the intent of Regulatory Guide 8.8 (i.e., to keep radiation exposures to plant personnel as low as reasonably achievable (ALARA)). This section describes the component and system designs in addition to the equipment layout employed to maintain radiation exposures ALARA. Consideration of individual systems is provided to illustrate the application of these principles.

Material application for primary coolant piping, tubing, vessel internal surfaces, and other components in contact with the primary coolant is discussed in the following pages. Typical nickel and cobalt contents of the principal materials applied are given in Table 12.3-2.

Carbon steel is used in a large portion of the system piping and equipment outside of the nuclear steam supply system. Carbon steel is typically low in nickel content and contains a very small amount of cobalt impurity.

Stainless steel is used in portions of the system such as the reactor internal components and heat exchanger tubes where high corrosion resistance is required. The nickel content of the stainless steels is in the 9 to 10.5% range and is controlled in accordance with applicable ASME material specifications. Cobalt content is controlled to less than 0.05% in the XM-19 alloy used in the control rod drives.

A previous review of materials certifications indicated an average cobalt content of only 0.15% in austenitic stainless steels.

Ni-Cr-Fe alloys such as Inconel 600 and Inconel X750, which have high nickel content, are used in some reactor vessel internal components. These materials are used in applications for which there are special requirements to be satisfied (such as possessing specific thermal expansion characteristics along with adequate corrosion resistance) and for which no suitable alternative low-nickel material is available. Cobalt content in the Inconel X750 used in the fuel assemblies is limited to 0.05%.

Stellite is used for hard facing of components which must be extremely wear resistant. Use of high cobalt alloys such as Stellite is restricted to those applications where no satisfactory alternative material is available. An alternative material (Colmonoy) has been used for some hard facings in the core area.

12.3.1.1 Equipment Design for Maintaining Exposure ALARA

This subsection describes specific components as well as system design features that aid in maintaining the exposure of plant personnel during system operation and maintenance ALARA. Equipment layout to provide ALARA exposures of plant personnel are discussed in Subsection 12.3.1.2.

(1) Pumps

Pumps located in radiation areas are designed to minimize the time required for maintenance. Quick change cartridge-type seals on pumps, and pumps with back pullout features that permit removal of the pump impeller or mechanical seals without disassembly of attached piping, are employed to minimize exposure time during pump maintenance. The configuration of piping about pumps is designed to provide sufficient space for efficient pump maintenance. Provisions are made for flushing and in certain cases chemically cleaning pumps prior to maintenance. Pump casing drains provide a means for draining pumps to the sumps prior to disassembly, thus reducing the exposure of personnel and decreasing the potential for contamination. Where two or more pumps conveying highly radioactive fluids are required for operational reasons to be located adjacent to each other, shielding is provided between the pumps to maintain exposure levels

ALARA. An example of this situation is the RWCS circulation pumps. Pumps adjacent to other highly radioactive equipment are also shielded to reduce the maintenance exposure, for example, in the radwaste system.

Whenever possible, operation of the pumps and associated valving for radioactive systems is accomplished remotely. Pump control instrumentation is located outside high radiation areas, and motor- or pneumatic-operated valves and valve extension stems are employed to allow operation from outside these areas.

(2) Instrumentation

Instruments are located in low radiation areas such as shielded valve galleries, corridors, or control rooms, whenever possible. Shielded valve galleries provided for this purpose include those for the RWCS, FPCC, and radwaste (cleanup phase separator, spent resin tank, and waste evaporator) systems. Instruments required to be located in high radiation areas due to operations requirements are designed such that removal of these instruments to low radiation areas for maintenance is possible. Sensing lines are routed from taps on the primary system in order to avoid placing the transmitters or readout devices in high radiation areas. For example, reactor water level as well as recirculation system pressure sensing instruments are located outside the drywell.

Liquid service equipment for systems containing radioactive fluids are provided with vent and backflush provisions. Instrument lines, except those for the reactor vessel, are designed with provisions for backflushing and maintaining a clean fill in the sensing lines. The reactor vessel sensing lines may be flushed with condensate following reactor blowdown.

(3) Heat Exchangers

Heat exchangers are constructed of stainless steel or Cu/Ni tubes to minimize the possibility of failure and reduce maintenance requirements. The heat exchanger design allows for the complete drainage of fluids from the exchanger, avoiding pooling effects that could lead to radioactive crud deposition. Connections are available for condensate or demineralized water flushing of the heat exchangers. For the reactor water clean up (CUW) system, separate connections are provided to chemically decontaminate both the heat exchangers (both regenerative and non-regenerative) and the pumps. The other main heat exchangers (RHR and RIP) are provided connections by which the exchangers can be flushed with clean water. The last main heat exchanger, the fuel pool heat exchanger, is downstream of the filter demineralizer and is therefore not subjected to flows containing significant amounts of fission or activation products. In all cases, the pumps directly involved with the heat exchangers are also inline for decontamination with the exchangers.. Instrumentation and valves are remotely operable to the maximum extent possible in the shielded heat exchanger cubicles, to reduce the need for entering these high radiation areas.

(4) Valves

Valve packing and gasket material are selected on a conservative basis, accounting for environmental conditions such as temperature, pressure, and radiation tolerance requirements to provide a long operating life. Valves have back seats to minimize the leakage through the packing. Straight-through valve configurations were selected where practical, over those which exhibit flow discontinuities or internal crevices to minimize crud trapping. Teflon gaskets are not used.

Wherever possible, valves in systems containing radioactive fluids are separated from those for "clean" services to reduce the radiation exposure from adjacent valves and piping during maintenance.

Pneumatic or mechanically operated valves are employed in high radiation areas, whenever practical, to minimize the need for entering these areas. For certain situations, manually operated valves are required, and in such cases extension valve stems are provided which are operated from a shielded area. Flushing and drain provisions are employed in radioactive systems to reduce exposure to personnel during maintenance.

For areas in which especially high radiation levels are encountered, valving is reduced to the maximum extent possible with the bulk of the valve and piping located in an adjacent valve gallery where the radiation levels are lower.

(5) Piping

Piping was selected to provide a service life equivalent to the design life of the plant, with consideration given to corrosion allowances and environmental conditions. Piping for service in radioactive systems such as the RWC system have butt-welded connections, rather than socket welds, to reduce crud traps. Distinction is made between piping conveying radioactive and nonradioactive fluids, and separate routing is provided whenever possible. Piping conveying highly radioactive fluids is usually routed through shielded pipe chases and shielded cubicles. However, when these options are not feasible, the radioactive piping is embedded in concrete walls and floors.

(6) Lighting

Lighting is designed to provided sufficient illumination in radiation areas to allow quick and efficient surveillance and maintenance operations. To reduce the need for immediate replacement of defective bulbs, multiple lighting fixtures are provided in shielded cubicles. Consideration is also given to locating lighting fixtures in easily accessible locations, thus reducing the exposure time for bulb replacement.

(7) Floor Drains

Floor drains with appropriately sloped floors are provided in shielded cubicles where the potential for spills exist. Those drain lines having a potential for containing highly radioactive fluids are routed through pipe chases, shielded cubicles, or are embedded in concrete walls and floors. Smooth epoxy-type coatings are employed to facilitate decontamination when a spill does occur.

(8) SGTS Filters

The SGTS filter is located in a separate shielded cubicle and is separated by a shield wall from the exhaust fans to reduce the radiation exposure of personnel during maintenance. The dampers located in the cubicles are remotely operated, thus requiring no access to the cubicle during operation. A pneumatic transfer system is employed to remove the radioactive charcoal from the filter, requiring entry into the shielded cubicle only during the connection of the hoses to the SGTS filter unit.

12.3.1.2 Plant Design for Maintaining Exposure (ALARA)

This subsection describes features of equipment layout and design which are employed to maintain personnel exposures ALARA.

(1) Penetrations

Penetrations through shield walls are avoided whenever possible to reduce the number of streaming paths provided by these penetrations. Whenever penetrations are required through shield walls, however, they are located to minimize the impact on surrounding areas. Penetrations are located so that the radiation source cannot "see" through the penetration. When this is not possible, or to provide an added order of reduction, penetrations are located to exit far above floor level in open corridors or in other relatively inaccessible areas. Penetrations which are offset through a shield wall are frequently employed for electrical penetrations to reduce the streaming of radiation through these penetrations.

Where permitted, the annular region between pipe and penetration sleeves, as well as electrical penetrations, are filled with shielding material to reduce the streaming area presented by these penetrations. The shielding materials used in these applications include a lead-loaded silicone foam, with a density comparable to concrete, and a boron-loaded refractory-type material for applications requiring neutron as well as gamma shielding. There are certain penetrations where these two approaches are not feasible or are not sufficiently effective. In those cases, a shielded enclosure about the penetration as it exits in the shield wall, with a 90 degree bend of the process pipe as it exits the penetration, is employed.

(2) Sample Stations

Sample stations in the plant provide for the routine surveillance of reactor water quality. These sample stations are located in low radiation areas to reduce the exposure to operating personnel. Flushing provisions are included using demineralized water, and pipe drains to plant sumps are provided to minimize the possibility of spills. Fume hoods are employed for airborne contamination control. Both working areas and fume hoods are constructed of polished stainless steel to ease decontamination if a spill does occur. Grab spouts are located above the sink to reduce the possibility of contaminating surrounding areas during the sampling process.

(3) HVAC Systems

Major HVAC equipment (blowers, coolers, and the like) is located in dedicated low radiation areas to maintain exposures to personnel maintaining these equipment ALARA. HVAC ducting is routed outside pipe chases and does not penetrate pipe chase walls, which could compromise the shielding. HVAC ducting penetrations through walls of shielded cubicles are located to minimize the impact of the streaming radiation levels in adjoining areas. Additional HVAC design considerations are addressed in Subsection 12.3.3.

(4) Piping

Piping containing radioactive fluids is routed through shielded pipe chases, shielded equipment cubicles, or embedded in concrete walls and floors, whenever possible. "Clean" services such as compressed air and demineralized water are not routed through shielded pipe chases. For situations in which radioactive piping must be routed through corridors or other low radiation areas, an analysis is conducted to ensure that this routing does not compromise the existing radiation zoning.

Radioactive services are routed separately from piping containing nonradioactive fluids, whenever possible, to minimize the exposure to personnel during maintenance. When such routing combinations are required, however, drain provisions are provided to remove the radioactive fluid contained in equipment and piping. "Clean" services and radioactive piping are required at times to be routed together in shielded cubicles. In such situations, provisions are made for the valves required for process operation to be controlled remotely, without need for entering the cubicle.

Penetrations for piping through shield walls are designed to minimize the impact on surrounding areas. Approaches used to accomplish this objective are described in Subsection 12.3.1.2.1.

Piping configurations are designed to minimize the number of "dead legs" and low points in piping runs to avoid accumulation of radioactive crud and fluids in the line. Drains and flushing provisions are employed whenever feasible to reduce the impact of required "dead legs" and low points. Systems containing radioactive fluids are welded to the most practical extent to reduce leakage through flanged or screwed connections. For highly radioactive systems, butt welds are employed to minimize crud traps. Provisions are also made in radioactive systems for flushing with condensate or chemically cleaning the piping to reduce crud buildup.

(5) Equipment Layout

Equipment layout is designed to reduce the exposure of personnel required to inspect or maintain equipment. "Clean" pieces of equipment are located separately from those which are sources of radiation whenever possible. For systems that have components that are major sources of radiation, piping and pumps are located in separate cubicles to reduce exposure from these components during maintenance. These major radiation sources are also separately shielded from each other.

(6) Contamination Control

Contaminated piping systems are welded to the most practical extent to minimize leaks through screwed or flanged fittings. For systems containing highly radioactive fluids, drains are hard piped directly to equipment drain sumps, rather than to allow contaminated fluid to flow across the floor to a floor drain. Certain valves in the main steam line are also provided with leakage drains piped to equipment drain sumps to reduce contamination of the steam tunnel. Pump casing drains are employed on radioactive systems whenever possible to remove fluids from the pump prior to disassembly. In addition, provisions for flushing with condensate, and in especially contaminated systems, for chemically cleaning the equipment prior to maintenance, are provided.

The HVAC system is designed to limit the extent of airborne contamination by providing air flow patterns from areas of low contamination to more contaminated areas. Penetrations through outer walls of the building containing radiation sources are sealed to prevent miscellaneous leaks into the environment. The equipment drain sump vents are fitted with charcoal canisters or piped directly to the radwaste HVAC system to remove airborne contaminants evolved from discharges to the sump. Wet transfer of both the steam dryer and separator also reduces the likelihood of contaminants on this equipment being released into the plant atmosphere. In areas where the reduction of airborne contaminants cannot be eliminated efficiently by HVAC systems, breathing air provisions are provided, for example, for CRD removal under the reactor pressure vessel and in the CRD maintenance room.

Appropriately sloped floor drains are provided in shielded cubicles and other areas where the potential for a spill exists to limit the extent of contamination. Curbs are also provided to limit contamination and simplify washdown operations. A cask decontamination vault is located in the reactor building where the spent fuel cask and other equipment may be cleaned. The CRD maintenance room is used for disassembling control rod drives to reduce the contamination potential.

Consideration is given in the design of the plant for reducing the effort required for decontamination. Epoxy-type wall and floor coverings have been selected which provide smooth surfaces to ease decontamination surfaces. Expanded metal-type floor gratings are minimized in favor of smooth surfaces in areas where radioactive spills could occur. Equipment and floor drain sumps are stainless steel lined to reduce crud buildup and to provide surfaces easily decontaminated.

12.3.1.3 Radiation Zoning

Radiation zones are established in all areas of the plant as a function of both the access requirements of that area and the radiation sources in that area. Operating activities, inspection requirements of equipment, maintenance activities, and abnormal operating conditions are considered in determining the appropriate zoning for a given area. The relationship between radiation zone designations and accessibility requirements is presented in the following tabulation:

Zone Designation	Dose Rate (mRem/hr)	Access Description
A	≤ 0.6	Uncontrolled, unlimited access
B	< 1	Controlled, unlimited access

Zone Desig-nation	Dose Rate (mRem/hr)	Description
C	< 5	Controlled, limited access, 20 hr/wk
D	< 25	Controlled, limited access, 4 hr/wk
E	< 100	Controlled, limited access, 1 hr/wk
F	> 100	Controlled access. Authorization required.

The dose rate applicable for a particular zone is based on operating experience and represents design dose rates in a particular zone, ar should not be interpreted as the expected dose rates which would apply in all portions of that zone, or for all types of work within that zone, or at all periods of entry into the zone. Large BWR plants have been in operation for two decades, and operating experience with similar design basis numbers shows that only a small fraction of the 10CFR20 maximum permissible dose is received in such zones from radiation sources controlled by equipment layout or the structural shielding provided. Therefore, on a practical basis, a radiation zoning approach as described above accomplishes the as low as reasonably achievable objectives for doses as required by 10 CFR 20.1(c). The radiation zone maps for this plant with zone designations as described in the preceding tabulations are contained in Figures 12.3-1 through 12.3-22 and 12.3-37 through 12.3-55.

Access to areas in the plant is controlled and regulated by the zoning of a given area. Areas with dose rates such that an individual would receive a dose in excess of 100 mRem in a period of one hour are locked and posted with "High Radiation Area" signs. Entry to these areas is on a controlled basis. Areas in which an individual would receive a dose in excess of 5 mRem up to 100 mRem within a period of one hour are posted with signs indicating that this is a radiation area and include, in certain cases, barriers such as ropes or doors.

12.3.1.4 Implementation of ALARA

In this subsection, the implementation of design considerations to radioactive systems for maintaining personnel radiation exposures as low as reasonably achievable is described for the following five systems:

(1) Reactor water cleanup system;

(2) Residual heat removal system (shutdown cooling mode);

(3) Fuel pool cooling and cleanup system;

(4) Main steam; and

(5) Standby gas treatment system

12.3.1.4.1 Reactor Water Cleanup System

This system is designed to operate continuously to reduce reactor water radioactive contamination. Components for this system are located outside the containment and include filter demineralizers, a backwash receiving tank, regenerative and nonregenerative heat exchangers, pumps, and associated valves.

The highest radiation level components include the filter demineralizers, heat exchangers, and backwash receiving tank. The filter demineralizers are located in separate concrete-shielded cubicles which are accessible through shielded hatches. Valves and piping within the cubicles are reduced to the extent that entry into the cubicles is not required during any operational phase. Most of the valves and piping are located in a shielded valve gallery adjacent to the filter demineralizer cubicles. The valves are remotely operable to the greatest practical extent to minimize entry requirements into this area. The RWCS heat exchangers are also located in a shielded cubicle with valves operated remotely by use of extension valve stems, or from instrument panels located outside the cubicle. The backwash tank is shielded separately from the resin transfer pump, permitting maintenance of the pump without being exposed to the spent resins contained in the backwash tank. The pump valves are operated remotely from outside the cubicle.

The RWCS system is provided with chemical cleaning connections which can utilize the condensate system to flush piping and equipment prior to maintenance. The RWCS filter demineralizer can be remotely back-flushed to remove spent resins and filter aid material. If additional decontamination is required, chemical addition connections are provided in the piping to clean piping as well as equipment prior to maintenance. The backwash tank employs an arrangement to agitate resins prior to discharge. The tank vent is fitted with a charcoal filter canister to reduce emission of radioiodines into the plant atmosphere. The HVAC system is designed to limit the spread of contaminants from these shielded cubicles by maintaining a negative pressure in the cubicles relative to the surrounding areas.

Personnel access to the cubicles for maintenance of these components is on a controlled basis whereby specific restrictions and controls are implemented to minimize personnel exposure.

12.3.1.4.2 Residual Heat Removal System (Shutdown Cooling Mode)

In the shutdown cooling mode, the system is placed in operation to recirculate reactor coolant to remove reactor decay heat following the period of approximately 2 to 4 hours after shutdown. During power operation, the system is not in use except for flow testing to and from the suppression pool. Therefore, there is no reactor coolant flow through the system and only traces of residual radioactive contamination may exist from prior operation.

System components are located in the reactor building and include three RHR pumps and three heat exchangers, which are actively used in the shutdown cooling mode. The heat exchangers and associated pumps work independently of the other pump and heat exchangers and are located in separate concrete-shielded cubicles. The cubicles are accessible through labyrinths which reduce radiation levels outside the cubicle to acceptable levels. A knockout wall constructed of vertically and horizontally lapped concrete blocks is provided for pump removal. A concrete hatch is provided through the roof of the cubicle for heat exchanger removal. Highest radiation levels occur at the heat exchangers during the cooldown period (1/2 to 4 hours after shutdown). During all other operation and plant shutdown periods, the radiation level near these components is considerably decreased.

Access to the RHR pumps and heat exchangers for any inspection or maintenance is permitted on a controlled basis. System maintenance is performed during periods of system shutdown when no reactor coolant is being circulated through the system. Specific restrictions and controls for personnel entry into the shielded cubicles are implemented to minimize personnel exposures. Inspection of the equipment in these cubicles can be conducted from platforming about the heat exchangers to simplify inspection of this equipment and consequently reduce the exposure during inspection.

The reactor building is not used exclusively for radioactive equipment or systems. However, all components of the system, as described, are contained within shielded cubicles. This shielding is sufficient to reduce the radiation level during the shutdown mode of operation to less than 5 mR/hr in adjacent areas where clean components, materials, or equipment are located.

System control panels and instrumentation are located in the main control room. This precludes exposure to the control operator during operation of the system for plant cooldown.

12.3.1.4.3 Fuel Pool Cooling and Cleanup System

This system is designed to operate continuously to handle the spent fuel cooling load and to reduce pool water radioactive contamination.

The system components are located in the reactor building. Included are two filter demineralizer units which serve to remove radioactive contamination from the fuel pool and suppression water. These units are the highest radiation level components in the system. Each unit is located in a concrete-shielded cubicl, which is accessible through a shielded hatch. Provisions are made for remotely backflushing the units when filter and resin material are spent. This removal of radioactively contaminated material reduces the component radiation level considerably and serves to minimize exposures during maintenance. All valves (inlet, outlet, recycle, vent, and drain) to the filter demineralizer units are located outside the shielded cubicles in a separate shielded cubicle together with associated piping, headers, and instrumentation. The radiation level in this cubicle is sufficiently low to permit required maintenance to be performed. Piping potentially containing resin is continuously sloped downward to the backwash tank.

The backwash tank is shared with the RWCS (see Section 12.3.1.4.1). The system also includes two low radiation level heat exchangers and two circulation pumps. The heat exchangers' design radiation levels are low enough to locate them in an open alcove area. The pumps are located in a low radiation area adjacent to the shielded backwash tank. System piping is routed so as not to compromise zoning requirements as established in the radiation zone maps.

All of the aforementioned shielded system components are consolidated in the same section of the reactor building. Personnel access to shielded system components is controlled to minimize personnel exposure. Shielding for the components is designed to reduce the radiation level to less than 1 mR/hr in adjacent areas where normal access is permitted. Controlled areas where the new resin tank, filter aid tank, and pumps are located, are shielded to less than 5 mR/hr.

Operation of the system is accomplished from the MRC and local control panels located where designed radiation levels are less than 1 mR/hr and normal personnel access is permitted.

12.3.1.4.4 Main Steam System

All radioactive materials in the main steam system, located in the main steam-feedwater pipe tunnel of the reactor buildings, result from radioactive sources carried over from the reactor during plant operation, including high energy short-lived Nitrogen-16. During plant shutdown, residual radioactivity from prior plant operation is the radiation source.

Access to the main steam pipe tunnel in the reactor building is controlled. Entry into the reactor building steam tunnel is through a controlled personnel access door shielded by a concrete labyrinth to attenuate radiation streaming from the steam lines to adjoining areas. During reactor operation, the steam tunnel is not accessible except in the hot standby conditions under regulated access.

Leakage from selected valves on to surrounding areas is minimized by providing valve drains piped to equipment drain sumps. Floor drains are provided to minimize the spread of contamination should a leakage occur.

Penetrations through the steam tunnel walls are minimized to reduce the streaming paths made available by these penetrations. The blowout panels for the steam tunnel are located in the relatively inaccessible upper section of the RHR heat exchanger shielded cubicles which are controlled access areas. Penetrations through the steam tunnel walls, when they are required, are located so as to exit in controlled access areas or in areas that are not aligned with the steam lines. A lead-loaded silicone foam is employed whenever possible for these penetrations to reduce the available streaming area presented.

12.3.1.4.5 Standby Gas Treatment System

The standby gas treatment system treats the reactor building ventilation air in the event of the release of radioactivity to this building. The system contains radioactivity only in the event of an emergency of abnormal condition. However, it is a potential source of concentrated radioactivity following such an occurrence.

The system starts automatically on a high building ventilation radiation or LOCA signal and can also be manually started from the main control room. Operation of the system does not require entering the shielded filter cubicle.

The system consists of two parallel treatment trains, each train being located in its own shielded room. In addition, the fans for each train are shielded from the filter, which is the dominant source of radiation for the system. Each train includes high efficiency particulate filters and charcoal filters for removal of radioactivity prior to exhausting air to the outside environment.

All components are located in the reactor building, and personnel access to the shielded rooms for inspection or maintenance is on a controlled basis. A remote charcoal filter removal capability is provided to minimize exposures, which requires entry into the filter area only during the initial connection of the unit to the charcoal removal system. Sufficient space is provided around the filter unit to allow easy removal and bagging of the high efficiency filters.

The SGTS filter shielding is adequate to reduce the radiation level in fuel areas of the reactor building to less than 1 mR/hr following an isolation scram event with containment purge.

12.3.2 Shielding

12.3.2.1 Design Objectives

The primary objective of the radiation shielding is to protect operating personnel and the general public from radiation emanating from the reactor, the power conversion systems, the radwaste process systems, and the auxiliary systems, while maintaining appropriate access for operation and maintenance. The radiation shielding is also designed to keep radiation doses to equipment below levels at which disabling radiation damage occurs. Specifically, the shielding requirements in the plant are designed to perform the following functions:

(1) limit the exposure of the general public, plant personnel, contractors, and visitors to levels that are ALARA and within 10CFR20 requirements;

(2) limit the radiation exposure of personnel, in the unlikely event of an accident, to levels that are ALARA and which conform to the limits specified in 10CFR50, Appendix A, Criterion 19 to ensure that the plant is maintained in a safe condition during an accident; and

(3) limit the radiation exposure of critical components within specified radiation tolerances, to assure that component performance and design life are not impaired.

12.3.2.2 Design Description

12.3.2.2.1 General Design Guides

In order to meet the design objectives, the following design guides are used in the shielding design of the ABWR:

(1) All systems containing radioactivity are identified and shielded based on access and exposure level requirements of surrounding areas. The radiation zone maps described in Subsection 12.3.1.3 indicate design radiation levels for which shielding for equipment contributing to the dose rate in the area is designed.

(2) The source terms used in the shielding calculations are analyzed with a conservative approach. Transient conditions as well as shut down and normal operating conditions are considered to ensure that a conservative source is used in the analysis.

Shielding design is based on fission product quantities in the coolant corresponding to the design basis off-gas release, in addition to activation products. This is considered an anticipated operational occurrence, and hence represents conservatism in design. For components where N-16 is the major radiation source, a concentration based upon operating plant data is used.

(3) Effort is made to locate processing equipment in a manner which minimizes the shielding requirements. Shielded labyrinths are used to eliminate radiation streaming through access ways from sources located in cubicles.

(4) Penetrations through shield walls are located so as to minimize the impact on surrounding areas due to radiation streaming through the penetrations. The approaches used to locate and shield penetrations, when required, are discussed in Subsection 12.3.1.2 (1).

(5) Wherever possible, radioactive piping is run in a manner which will minimize radiation exposure to plant personnel. This involves:

(a) minimizing radioactive pipe routing in corridors;

(b) avoiding the routing of high-activity pipes through low-radiation zones;

(c) use of shielded pipe trenches and pipe chases, where routing of high-activity pipes in low-level areas cannot be avoided, or if these are not available and the pipe routing permits, embedding the pipes in concrete walls and floor; and

(d) separating radioactive and nonradioactive pipes for maintenance purposes.

(6) To maintain acceptable levels at the valve stations, motor-operated or diaphragm valves are used where practical. For valve maintenance, provision is made for draining and flushing associated equipment so that radiation exposure is minimized. If manual valves are used, provision is made for shielding the operator from the valve by use of shield walls and valve stem extensions, where practicable.

(7) Shielding is provided to permit access and occupancy of the control room to ensure that plant personnel exposure following an accident does not exceed the guideline values set forth in 10CFR50, Appendix A, Criterion 19. The analyses of the doses to Control Room personnel for the design basis accidents are included in Chapter 15.

(8) The dose at the site boundary as a result of direct and scattered radiation from the turbine and associated equipment is considered.

(9) In selected situations, provisions are made for shielding major radiation sources during inservice inspection to reduce exposure to inspection personnel. For example, steel platforms are provided for ISI of the RPV nozzle welds and associated piping.

(10) The primary material used for shielding is concrete at a density of 2.3 gr/cm^3. Concrete used for shielding purposes is designed in accordance with Regulatory Guide 1.69. Where special circumstances dictate, steel, lead, water, lead-loaded silicone foam, or a boron-laced refractory material is used.

(11) There is no field-routed piping in the ABWR design. Large and small piping, as well as instrument tubing, are routed by designers as indicated in the preceding paragraph (5).

12.3.2.2.2 Method of Shielding Design

The radiation shield wall thicknesses are determined using basic shielding data and proven shielding codes. A list of the computer programs used is contained in Table 12.3-1. The shielding design methods used also rely on basic radiation transport equations contained in Reference 1. The sources for basic shielding data, such as cross sections, buildup factors, and radioisotope decay information, are listed in References 2 through 10.

The shielding design is based on the plant operating at maximum design power with the release of fission products resulting in a source of 100,000 mCi/sec of noble gas after a 30 minute decay period, and the corresponding activation and corrosion product concentrations in the reactor water listed in Section 11.1. Radiation sources in various pieces of plant equipment are cited in Section 12.2. Shutdown conditions, such as fuel transfer operation, as well as accident conditions, such as a LOCA or an FHA, have also been considered in designing shielding for the plant.

The mathematical models used to represent a radiation source and associate equipment and shielding are established to ensure conservative calculational results. Depending on the versatility of the applicable computer program, various degrees of complexity of the actual physical situation are incorporated. In general, cylindrically shaped equipment such as tanks, heat exchangers, and demineralizers are mathematically modelled as truncated cylinders. Equipment internals are sectionally homogenized to incorporate density variations where applicable. For example, the tube bundle section of a heat exchanger exhibits a higher density than the tube bundle clearance circle, due to the tube density, and this variation is accounted for in the model. Complex piping runs are conservatively modelled as a series of point sources spaced along the piping run. Equipment containing sources in a parallelpiped configuration, such as fuel assemblies, fuel racks, and the SGTS charcoal filters, are modelled as parallelpiped with a suitable homogenization of materials contained in the equipment. The shielding for these sources is also modelled on a conservative basis, with discontinuities in the shielding, such as penetrations, doors, and partial walls accounted for. The dimension of the floor decking is not considered in the shielding calculation as it is part of the effective shield thickness provided by the floor slab.

Pure gamma dose rate calculations, both scattered and direct, are conducted using point kernel codes (QADF/GGG). The source terms are divided into groups as a function of photon energy, and each group is treated independently of the others. Credit is taken for attenuation through all phases of material, and buildup is accounted for using a third-order polynomial buildup factor equation. The more conservative material buildup coefficients are selected for laminated shield configuration to ensure conservative results.

For combined gamma and neutron shielding situations, discrete ordinates (ANISN) techniques are applied.

The shielding thicknesses are selected to reduce the aggregate dose rate from significant radiation sources in surrounding areas to values below the upper limit of the radiation zone specified in the zone maps in Subsection 12.3.1.3. By maintaining dose rates in these areas at less than the upper limit values specified in the zone maps, sufficient access to the plant areas is allowed for maintenance and operational requirements.

Where shielded entries to high-radiation areas such as labyrinths are required, a gamma ray scattering code (GGG) is used to confirm the adequacy of the labyrinth design. The labyrinths are designed to reduce the scattered as well as the direct contribution to the aggregate dose rate outside the entry, such that the radiation zone designated for the area is not violated.

12.3.2.3 Plant Shielding Description

Figures 12.3-1 through 12.3-11 show the layout of equipment containing radioactive process materials. The general description of the shielding is described below;

(1) Drywell

The major shielding structures located in the drywell area consist of the reactor shield wall and the drywell wall. The reactor shield wall in general consists of 0.6m of concrete sandwiched between two 3.7 cm thick steel plates. The primary function served by the reactor shield wall is the reduction of radiation levels in the drywell due to the reactor, to valves that do not unduly limit the service life of the equipment located in the drywell. In addition, the reactor shield wall reduces gamma heating effects on the drywell wall, as well as providing for low radiation levels in the drywell during reactor shutdown. Penetrations through the reactor shield wall are shielded to the extent that radiation streaming through the penetrations does not exceed the total neutron and gamma dose rates at the core midplane just outside 471.33 the reactor shield wall. The drywell is an F radiation zone during full power reactor operation and is not accessible during this period.

The drywell wall is a 2m thick reinforced concrete cylinder, which is topped by a 2.4m thick reinforced concrete cap. The drywell wall attenuates radiation from the reactor and other radiation sources in the drywell, such as the recirculation system and main steam piping, to allow occupancy of the reactor building during full power reactor operation.

(2) Reactor Building

In general, the shielding for the reactor building is designed to maintain open areas at dose rates less than 0.6 mR/hr.

Penetrations of the drywell wall are shielded to reduce radiation streaming through the penetrations. Localized dose rates outside these penetrations are limited to less than 5 mR/hr. The penetrations through interior shield walls of the reactor building are shielded using a lead-loaded silicone sleeve to reduce the radiation streaming are made available by the penetrations. Penetrations are also located so as to minimize the impact of radiation streaming into surrounding areas.

The components of the reactor water cleanup (RWC) system are located in the reactor building. Both the RWC regenerative and nonregenerative heat exchangers are located in shielded cubicles separated from the other components of the system. Neither cubicle needs to be entered for system operation.

Process piping between the heat exchangers and the filter demineralizers is routed through shielded areas or embedded in concrete to reduce the dose rate in surrounding areas. The two RWC system filter demineralizers are located in separate shielded cubicles, which allows maintenance of one unit while operating the other. The dose rate in the adjoining filter demineralizer cubicle from the operating unit is less than 6 mR/hr. Entry into the filter demineralizer cubicle, which is infrequently required, is via a stepped shield plug at the top of the cubicle. The bulk of the piping and valves for the filter demineralizers is located in an adjacent shielded valve gallery. Backflushing and resin application of the filter demineralizers are controlled from an area where dose rates are less than 1 mR/hr. The RWC system backwash receiving tank is also separately shielded from the other components of the RWC system, including the tank discharge pump, which allows maintenance of the pump without direct exposure to the spent resins contained in the backwash tank. The backwash tank cubicle is shielded to reduce the dose rate outside the entry to less than 1 mR/hr.

Shielding of the Transverse Incore Probe (TIP) is provided by locating the higher radiation components in a separate shielded room with labyrinth entry way. The TIP itself during maintenance is withdrawn into a lead shielded cask for entry into the room. The TIP location is maintained by a position sensor on the instrument which is alarmed to the control room. The TIP entry location into the room from the drywell is via the suppression pool instrumentation tunnel and then upward into the room. Area radiation monitors in both TIP room and spooler room maintain a secondary surveillance of both rooms being alarmed to both the control rooms and locally in the TIP facility. An inadvertent withdrawal of the TIP will result in alarming both the position sensor and area radiation monitors resulting in local alarms to egress the area.

(3) ECCS Components

The ECCS systems are located in separately shielded cubicles. Shield labyrinths are provided to gain entry into the cubicles, and equipment removal doors are shielded with removable horizontally and vertically lapped concrete block. Piping to and from the ECCS system is routed through shielded pipe chases. Access into the cubicles is not required to operate the systems. In general, the radiation levels in the open corridors of the reactor building are less than 1 mR/hr, except during RHR shutdown cooling mode operation, when radiation levels may temporarily range between 1 and 5 mR/hr in areas near the RHR cubicles.

The RWC system pumps are located in a shielded cubicle designed to reduce the radiation levels in the adjoining open corridor to less than 1 mR/hr. The pumps are separated by shield walls to allow operation of one of the pumps while performing maintenance on the other. Dose rates at this pump due to the operating pump and piping are less than 5 mR/hr. A shielded valve gallery is employed to permit manual operation of the valves associated with the RWC system pumps without entering the pump area. Piping for the pumps is directly routed from the steam tunnel to the RWC system pump area.

The CRD maintenance room walls are designed to reduce dose rates in the adjoining corridor to less than 1 mR/hr during all CRD maintenance operations except CRD transfer, when dose rates in the corridor temporarily range between 1 and 5 mR/hr.

The main steam lines are located in the shielded steam tunnel. The steam tunnel reduces the dose rates from the steam lines to less than 1 mR/hr in all adjoining areas except the roof of the steam tunnel, which is less that 5 mR/hr.

(4) Fuel Components

The fuel storage pool is designed to insure that the dose rate in adjoining areas is less than 1 mR/hr. During normal operation, dose rates in the pump area are less than 1 mR/hr. During an isolation transient, however, dose rates in the area temporarily increase to 700 mR/hr. Due to the nature of the event, egress from the area can be accomplished well before dose rates reach this level. Access to equipment in this area is not required during this occurrence. An individual in this area will know that the dose rate is increasing since a local-mounted area radiation monitoring sensor, converter, indicating auxiliary unit, and audio alarm are provided.

(5) Control Room

The dose rate in the control room is much less than 0.6 mR/hr during normal reactor operating conditions. The outer walls of the control building are designed to attenuate radiation from radioactive materials contained within the reactor building and from possible airborne radiation surrounding the control building following a LOCA. The walls provide sufficient shielding to limit the direct-shine exposure of control room personnel following a LOCA to a fraction of the 5 Rem limit as is required by 10CFR50, Appendix A, Criterion 19. Shielding for the outdoor air cleanup filters is also provided to allow temporary access to the mechanical equipment area of the control building following a LOCA, should it be required.

(6) The main steam tunnel extends from the primary containment boundary in the reactor building through the control building up to the turbine stop valves. The primary purpose of the steam tunnel is to shield the plant complex from N-16 gamma shine in the main steam lines. A minimum of 1.6 meters

of concrete or its equivalent (other material or distance) is required on any ray pathway from the main steam lines to any point which may be inhabited during normal operations. The design of the steam tunnel is shown on Figures 1.2-14, 1.2-15, 1.2-20, 1.2-21, and 1.2-28. The tunnel is classified as Seismic Category I in the reactor building and in the control building and is designed to UBC Seismic Standards in the turbine building. The interface between the buildings provides for bayonet connection to permit differential building motion during seismic events and shielding in the areas between buildings. The exact details on the bayonet design are not shown on the referenced arrangement drawings but requires complete shielding in the building interface area. The tunnel also serves a secondary purpose as a relief and release pathway for high energy events in the reactor building. Any high energy event (line break) in the reactor building will, through a series of blow out panels, vent into the steam tunnel and from the steam tunnel through the tunnel vent shaft to the turbine building (see Figure 1.2-28) for processing to the plant stack. See Subsection 6.2.3.3.1 for more complete description of this function.

12.3.3 Ventilation

The HVAC systems for the various buildings in the plant are discussed in Section 9.4, including the design bases, system descriptions, and evaluations with regard to the heating, cooling, and ventilating capabilities of the systems. This section discusses the radiation control aspects of the HVAC systems.

12.3.3.1 Design Objectives

The following design objectives apply to all building ventilation systems:

(1) The systems shall be designed to make airborne radiation exposures to plant personnel and releases to the environment ALARA. To achieve this objective, the guidance provided in Regulatory Guide 8.8 shall be followed.

(2) The concentration of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance shall be kept below the limits of 10CFR20 during normal power operation. This is accomplished by establishing in each area a reasonable comprimise between specifications on potential airborne leakages in the area and HVAC flow through the area. Appendix 12A to this chapter outlines the methodology by which such calculations are made.

The applicable guidance provided in Regulatory Guide 1.52 has been implemented for the ESF filter systems for the control building outdoor air cleanup system and the standby gas treatment system (STGS) as described in Subsections 6.5.1 and 9.4.1.

12.3.3.2 Design Description

In the following sections, the design features of the various ventilation systems that achieve the radiation control design objectives are discussed. For all areas potentially having airborne radioactivity, the ventilation systems are designed such that during normal and maintenance operations, airflow between areas is always from an area of low potential contamination to an area of higher potential contamination.

12.3.3.2.1 Control Room Ventilation

The control building atmosphere is maintained at a slightly positive pressure (up to 0.5 in. wg) at all times, except if exhausting or isolation are required, in order to prevent infiltration of contaminants. Fresh air is taken in via a dual inlet system, which has both intake structures on the roof of the building. The inlets are arranged with respect to the SGTS exhaust stack such that at least one of the intakes is free of contamination after a LOCA. Both inlets, however, can be submerged in contaminated air from a LOCA, but the calculated dose in the control room from such an eventuality is still below the limit of Criterion 19 of 10CFR50, Appendix A.

Outside air coming into the intakes is normally filtered by a particulate filter. If a high radiation level in the air is detected by the airborne radiation monitoring system, flow is automatically diverted to another filter train (an outdoor air cleanup unit) that has:

(1) a particular filter;

(2) a HEPA filter;

(3) a charcoal filter; and

(4) another HEPA filter.

Two redundant, divisionally separated radiation monitors and filter trains are provided. (See Subsection 9.4.1 for detailed description of the design.) Conservative calculations show that the filters keep the dose in the control room from a LOCA below the limits of Criterion 19 of 10CFR50, Appendix A.

The outdoor cleanup units are located in individual, closed rooms that help prevent the spread of any radiation during maintenance. Adequate space is provided for maintenance activities. The particulate and HEPA filters can be bagged when being removed from the unit. Before removing the charcoal, any radioactivity is allowed to decay to minimal levels, and is then removed through a connection in the bottom of the filter by a pneumatic transfer system. Air used in the transfer system goes through a HEPA filter before being exhausted. Face masks can worn during maintenance activities, if desired.

12.3.3.2.2 Drywell

Access into the drywell is not permitted during normal operation. The ventilation system inside merely circulates, without filtering, the air. The only airflow out of the drywell into accessible areas is minor leakage through the wall.

During maintenance, the drywell air is purged before access is allowed.

12.3.3.2.3 Reactor Building

The reactor building HVAC system is divided into three zones, which are separated by leaktight, physical barriers. The zones include:

(1) secondary containment (this area contains equipment that is a potential source of radioactivity and if a leak occurs, the other accessible areas of the building are not contaminated);

(2) electrical equipment area, cable tunnels, cable spreading rooms, remote control panel area, diesel generator rooms, reactor internal pump panel rooms, and the heating and ventilating equipment rooms; and

(3) steam tunnel (this room also contains a potential source of radioactive material leakage).

Air pressure in the rooms in Zone 1 is maintained slightly below outside atmospheric pressure by a fresh air supply and exhaust system. The supply air is filtered by a particulate filter. The exhaust stream is monitored for radioactivity, and if a high activity level is detected, the exhaust stream is diverted to the SGTS.

Normally, exhaust air is drawn from the corridor and various rooms. The exhaust duct has two isolation valves in series and a radiation monitor. The valves isolate the system if high airborne radioactivity is detected by the radiation monitor.

Zone 2 of the reactor building is maintained at a positive pressure during normal operation.

For a description of the reactor building HVAC system, see Subsection 9.4.5.

12.3.3.2.4 Radwaste Building

The radwaste building is divided into two zones for ventilation purposes. The control room is one zone, and the remainder of the building is the other zone. The air pressure in the first zone is maintained slightly above atmospheric, while the air pressure in the second zone is maintained slightly below atmospheric. Air in the second zone is drawn from outside the building and distributed to various work areas within the building. Air flows from the work areas and is then discharged via the reactor building stack. An alarm sounds in the control room if the exhaust fan fails. The exhaust flow is monitored for radioactivity, and if a high activity level is detected, the potentially radioactive cells are automatically isolated, but airflow through the work areas continues.

If the exhaust flow high-radiation alarm continues to annunciate after the tank and pump rooms are isolated, the work area branch exhaust ducts are selectively manually isolated to locate the involved building area. Should this technique fail, because the airborne radiation has spread throughout the building, the control room air conditioning continues, but the air conditioning for the balance of the building is shut down.

The work area's exhaust air is drawn through a filter unit consisting of a particulate filter, a HEPA filter, a charcoal filter, and then another HEPA filter, before being discharged to the reactor building stack. The air is monitored for radioactivity, and if a high level is detected, supply and exhaust is terminated, and the SGTS is started.

Maintenance provisions for the filters are similar to those for the control building HVAC system.

See Subsection 9.4.6 for a detailed discussion of the radwaste building HVAC system.

12.3.4 Area Radiation and Airborne Radioactivity Monitors

This section defines and describes the area radiation system that monitors the gamma radiation levels throughout the plant except within the containment. The gamma radiation levels within the containment (drywell and suppression chamber) are monitored continuously by the containment atmospheric monitoring system (CAMS) as described in Subsection 7.6.2. Four gamma sensitive ion chambers (two per divisions 1 & 2) are provided by CAMS to monitor for airborne radioactivity up to 10^7 rads per/hr. Those four sensors are located at the penetrations listed in Table 6.2-8. The area radiation monitoring system is classified as non-safety.

12.3.4.1 System Objectives

The purpose of the area radiation monitoring system is to warn plant personnel of excessive gamma ray levels in service areas including the areas where nuclear fuel is stored or handled, to record and indicate the monitored gamma radiation levels in the control room at selected locations within the various plant buildings, and to provide audible local alarms at key locations where abnormal radiation levels could endanger plant personnel.

12.3.4.2 System Description

The area radiation monitoring system consists of gamma sensitive detectors, associated digital radiation monitors, auxiliary units, local audible warning devices and multipoint recorders. The detector signals are digitized and optically multiplexed for transmission to the radiation monitors. Each monitor has two adjustable trip circuits for alarm initiation, one high radiation level trip and one downscale trip. The downscale trip circuit operates on loss of power or when gross equipment failure occurs. Auxiliary units are provided in local areas for radiation indication and for initiating the sonic alarms on abnormal levels. The electronics are powered from the non-1E vital 120 Vac source while the recorders are powered from the 120 Vac instrument bus.

12.3.4.3 System Design

The area radiation monitoring detectors provided in each plant building are listed in Tables 12.3-3 through 12.3-7 along with area location maps shown in Figures 12.3-56 through 12.3-73. Also, these tables specify the sensitivity range of each channel as designated below along with requirements for local area alarms.

The channel sensitivity covers the following ranges:

a) Range 10^{-2} to 10^2 mR/hr - H (High Sensitivity)

b) Range 10^{-1} to 10^3 mR/hr - M (Medium Sensitivity)

c) Range 1 to 10^4 mR/hr - L (Low Sensitivity)

d) Range 10^2 to 10^6 mR/hr - LL (Low Low Sensitivity)

e) Range 10^{-1} to 10^4 mR/hr - VL (Very Low Sensitivity)

There are two radiation detectors that are located in the fuel storage and handling area, one is positioned to monitor the radiation near the fuel pool and the other is placed in the fuel handling area to monitor the radiation that may result from accidental fuel handling. Criticality detection monitors for this area are not needed to satisfy the criticality accident requirements of 10CFR70.24, because the ABWR design utilizes specialized high density fuel storage racks that preclude the possibility of criticality accident under normal and abnormal conditions. The new fuel bund' are stored in racks that are placed at the bottom of the fuel storage pool. A full array of loaded fuel storage racks are designed to be subcritical by at least 5% delta k. Refer to Sections 9.1 and 9.2 for details.

The detectors and radiation monitors are responsive to gamma radiation over an energy range of 80 keV 7 MeV. The energy dependence will not exceed 20% of point from 100eV to 3 MeV. The overall system design accuracy is within 9.5% of equivalent linear full scale recorder output for any decade.

The trip alarm setpoints will be established in the field following equipment installation at the site. The exact settings will be based on sensor location, back ground radiation levels, expected radiation levels, and low occupational radiation exposures.

Each channel is calibrated based on a pseudo input signal to confirm accurate monitor response. The detectors are calibrated using standardized traceable radioactive source in order to establish the linearity and sensitivity of the channel for subsequent calibration. The area radiation monitoring system is designed to accommodiate periodic surveillance testing.

The area radiation monitoring instrumentation is designed and properly located to provide early detection and warning for personnel protection to insure that occupational radiation exposures will be as low as is reasonably achieved (ALARA) in accordance with guidelines stipulated in Reg Guide 8.2 and 8.8.

The area radiation monitoring system includes instrumentation provided to assess the radiation conditions in crucial areas in the reactor building (the RHR equipment areas) where access may be required to service the safety related equipment during post LOCA per Reg Guide 1.97.

12.3.5 Post-Accident Access Requirements

The locations requiring access to mitigate the consequences of an accident during the 100-day post-accident period are the control room, the technical support center, the remote shutdown panel, the primary containment sample station (post accident sample system), the health physics facility (counting room), and the nitrogen gas supply bottles. Each area has low post LOCA radiation levels. The dose evaluations in Subsection 15.6.5 are within regulatory guidelines.

Access to vital areas through out the reactor building/control building/turbine building complex is controlled via the service building. Entrance to the service building and access to the other areas are controlled via double locked secured entry ways. Access to the reactor building is via two specific routes, one for clean access and the second for controlled access. During a event such as a design basis accident, the service building/control building are maintained under filtered HVAC at a positive pressure with respect to the environment. Air infiltration is minimized by positive flow via double entry ways. Therefore, radiation exposure is limited to gamma shine from the reactor building, turbine building, main steam line access corridor, and skyline. This shine is minimized by locating highly populated areas below ground.

During a design basis accident event, access to remote shutdown panel, nitrogen bottles, and the PASS and monitor systems is controlled from the service building via the controlled access way. These corridors are not maintained under filtered positive pressure so that personal protection equipment (radiation protection suits, breathing gear, etc.) will be required in the access corridor. Primary contamination would occur from leakage through the PASS system and air infiltration from the environment. Both pathways are considered minimal and minor contamination under even the most adverse conditions is expected.

The reactor building vital areas are all located off one of of the two primary access ways except the nitrogen bottle areas which are located on the refueling floor and are accessible from the clean access corridor at the 4800 level (B1F) and up three floors to the 23500 level (3F). There are two access corri- dors, clean and dirty, with contamination in those areas limited to air infiltration from the environment and penetration leakage from the PASS system. In addition, the lines penetrating the PASS room are doubly valved permitting line isolation in the event of any potential rupture. Sources of radiation therefore are limited to minor leakage and gamma shine including the stack monitor room which contains only instrumentation and associated penetrations for monitoring stack effluent.

12.3.6 Post-Accident Radiation Zone Maps

The post-accident radiation zone maps for the areas in the reactor building are presented in Figures 12.3-12 through 12.3-22. The zone maps represent the maximum gamma dose rates that exist in these areas during the post-accident period. These dose rates do not include the airborne contribution in the reactor building.

Post-accident zone maps of the control building and turbine building are presented in Figures 12.3-54 and 55 respectively. The zone maps are designed to reflect the criteria established in Subsection 3.1.2.2.10.

12.3.7 Deleted

12.3.8 References

1. N. M. Schaeffer, *Reactor Shielding for Nuclear Engineers*, TID-25951, U.S. Atomic Energy Commission (1973).

2. J. H. Hubbell, *Photon Cross Sections, Attenuation Coefficients, and Energy Absorption Coefficients from 10 KeV to 100 GeV*, NSRDS-NBS20, U.S. Department of Commerce, August 1969.

3. *Radiological Health Handbook*, U.S. Department of Health, Education, and Welfare, Revised Edition, January 1970.

4. *Reactor Handbook*, Volume III, Part B, E.P. Blizzard, U.S. Atomic Energy Commission (1962).

5. Lederer, Hollander, and Perlman, *Table of Isotopes*, Sixth Edition, (1968).

6. M.A. Capo, *Polynomial Approximation of Gamma Ray Buildup Factors for ' Point Isotropic Source*, APEX-510, Novem .. 1958.

7. *Reactor Physics Constants*, Second Edition, ANL-5800, U.S. Atomic Energy Commission, July 1963.

8. ENDF/B-III and ENDF/B-IV Cross Section Libraries, Brookhaven National Laboratory.

9. PDS-31 Cross Section Library, Oak Ridge National Laboratory.

10. DLC-7, ENDF/B Photo Interaction Library.

Table 12.3-1

COMPUTER CODES USED IN
SHIELDING DESIGN CALCULATIONS

Computer Code	Description
QADF	A multigroup, multiregion, point kernal, gamma ray code for calculating the flux and dose rate at discrete locations within a complex source-geometry configuration.
GGG	A multigroup, multiregion, point kernal code for calculating the contribution due to gamma ray scattering in a hetrogeneous three-dimensional space
DOT.4	A discrete ordinates, two-dimensional transport code. Multigroup, multiregion neutron or gamma transport

Table 12.3-2

TYPICAL NICKEL AND COLBALT CONTENT OF MATERIALS

Material	Nickel (%)	Colbalt (%)
Carbon Steel	0.25	1% of Ni
Stainless Steel	10	1% of Ni
Ni-Cr-Fe (Inconel 600, Inconel X750)	70	1% of Ni
Stellite 6	3	58

Table 12.3-3

AREA RADIATION MONITORS
REACTOR BUILDING

No.	Location & Description	Figure #	Sensitivity Range	Local Alarms
1	Reactor area (A)-4F	12.3-62	H	X
2	Reactor area (B)-4F	12.3-62	LL	
3	Fuel storage pool area (A)-4F	12.3-62	LL	X
4	Fuel storage pocl area (B)-4F	12.3-62	LL	
5	R/B 4F south area	12.3-62	H	
6	R/B 4F SE area	12.3-62	H	X
7	R/B 3F NW area	12.3-60	H	
8	R/B 3F SE area	12.3-60	H	X
9	CUW control panel area-B3F	12.3-56	H	
10	R/B equipment hatch-B2F	12.3-57	H	X
11	HCU area (A)-B3F	12.3-56	M	X
12	HCU area (B)-B3F	12.3-56	M	X
13	SRV/MSIV valve maintenance room-3F	12.3-63	M	X
14	R/B 1F SE hatch area	12.3-49	H	X
15	RPV instrument rack room (A)-B1F	12.3-58	H	X
16	RPV instrument rack room (B)-B1F	12.3-58	H	X
17	R/B B1F SE hatch area	12.3-58	H	
18	TIP drive machine room-El 1500	12.3-57	M	X
19	TIP machine equipment room-El 1500	12.3-57	L	X
20	Core cooling water sampling room-M4F	12.3-61	M	X
21	CRD maintenance room-B2F	12.3-57	M	X
22	R/B B2F SE hatch area	12.3-57	H	X
23	R/B B2F NW hatch area	12.3-57	H	X
24	R/B B3F NW area-RHR "A" equip area	12.3-56	VL	X
25	R/B B3F SE area-RHR "B" equip area	12.3-56	VL	X

Table 12.3-4

AREA RADIATION MONITORS
CONTROL BUILDING

No.	Location & Description	Figure #	Sensitivity Range
1	Main Control Room	12.3-64	H
2	Passage Way Underneath Steam Tunnel	12.3-64	H
3	RBCW "A" Area-El -1315	12.3-64	H
4	RBCW "B" Area-El -1315	12.3-64	H
5	RBCW "C" Area-El -1315	12.3-64	H

Table 12.3-5

AREA RADIATION MONITORS
SERVICE BUILDING

No.	Location & Description	Figure #	Sensitivity Range
1	Service Building Tech. Support Center	12.3-64	H

Table 12.3-6

AREA RADIATION MONITORS
RADWASTE BUILDING

No.	Location & Description	Figure #	Sensitivity Range	Local Alarms
1	R/W Building Control Room-El 16000	12.3-68	H	
2	Maintenance area #1-El 16000	12.3-68	H	X
3	Maintenance Area #2-El 16000	12.3-68	H	X
[illegible]	R/W Building HVAC Exhaust El 1600	12.3-68	H	
5	R/W Building Truck Area-El 7300	12.3-67	H	
6	MSW Compactor Area-El 7300	12.3-67	H	
7	Corridor to Aux. Building-El 7300	12.3-67	H	X
8	Equip Rack Area #1-El -0200	12.3-66	H	
9	Equip Rack Area #2-El -0200	12.3-66	H	
10	R/W Building MSW Control Room-El -0200	12.3-66	H	
11	Rad Waste Sampling Room-El -6500	12.3-65	H	
12	MSW Equipment Area-El -5500	12.3-65	H	X
13	R/W Equipment Rack Area #1-El -6500	12.3-65	H	
14	R/W Equipment Rack Area #2-El -6500	12.3-65	H	

Table 12.3-7

AREA RADIATION MONITORS
TURBINE BUILDING

No.	Location & Description	Figure No.	Sensitivity Range	Local Alarms
1.	Condensate Pump Maintenance Area	12.3-70	M	
2.	Condensate Sampling & Control Area	12.3-70	M	X
3.	Off-Gas Sample & Control Area	12.3-70	M	X
4.	RFP 1A, 1B & 1C Area	12.3-70	H	X
5.	Filter Maintenance Area	12.3-71	M	X
6.	Demineralizer Area	12.3-71	H	
7.	SJAE A & Recombiner Area	12.3-71	H	
8.	SJAE B & Recombiner Area	12.3-71	H	
9.	HP Heaters & Drain Tank Area 1	12.3-71	H	
10.	HP Heaters & Drain Tank Area 2	12.3-71	H	
11.	MSR 1A & 1C Area	12.3-72	H	
12.	MSR 1B & 1D Area	12.3-72	H	
13.	Turbine Building Operating Floor	12.2-73	H	X
14.	Equipment Main Access Area	12.3-73	H	X

APPENDIX 12A

CALCULATION OF AIRBORNE RADIONUCLIDES

12A.1 CALCULATION OF AIRBORNE RADIONUCLIDES

This appendix presents a simplified methodology to calculate the airborne concentrations of radionuclides in a compartment. This methodology is conservative in nature and assumes that diffusion and mixing in a compartment is basically instantaneous with respect to those mitigating mechanisms such as radioactive decay and other removal mechanisms. The following calculations need to be performed on an isotope by isotope basis to verify airborne concentrations are within the limits of 10CFR20.

(1) For the compartment, all sources of airborne radionuclides need to be identified such as:

(a) Flow of contaminated air from other areas

(b) Gaseous releases from equipment in the compartment

(c) Evolution of airborne sources from sumps or water leaking from equipment

(2) Second, the primary sinks of airborne radionuclides need to be identified. This will primarily be outflow from the compartment but may also take the form of condensation onto room coolers.

(3) Given the above information the following equation will calculate a conservative concentration.

$$C_i = \frac{1}{V} \sum_j \frac{S_{ij}}{(\lambda_i + \sum_k R_{ijk})}$$

Where:

C_i = Concentration of the ith radionuclides in the room

V = Volume of room

S_{ij} = The jth source (rate) of the ith radionuclide to the room. These sources are discussed below.

R_{ijk} = the kth removal constant for the jth source and the ith radionuclide as discussed below.

λ_i = radionuclide decay constant

Evaluation Parameters

The following parameters require evaluation on a case by case basis dictated by the physical parameters and processes germaine to the modeling process.

(1) S_{ij} - is defined as the source rate for radionuclide i into the compartment. Typically these sources take the form of:

(a) Inflow of contaminated air from an upstream compartment. Given the concentration of radionuclide i, c_i, in this air and a flow rate of "r", the source rate then becomes $S_{ij} = rc_i$

(b) Production of airborne radionuclides from equipment. This typically takes two forms, gaseous leakage, and liquid leakage.

(i) For gaseous leakage sources, the source rate is equal to the concentration of radionuclide i, c_i, and the leakage rate, "r", or $S_{ij} = rc_i$.

(ii) For liquid sources, the source rate is similar but more complex. Given a liquid concentration c_i and a leakage rate, "r", the total release from the leak is rc_i. The fraction of this release which then becomes airborne is typically evaluated by a partition factor, P_f which may be conservately estimated from:

Noble Gases $P_f = 1$

All others $P_f = \dfrac{h_t - h_f}{h_s - h_f}$

where: h_t = saturated liquid enthalpy

h_f = saturated liquid enthalpy at one atmosphere = 100.10 Kcal/Kg

h_s = saturated vapor enthalpy at one atmosphere = 639.18 Kcal/Kg

Therefore the liquid release rate becomes, $r_i P_f$

(2) R_{ijk} is defined as the removal rate constant and typically consists of:

(a) Exhaust rate from the compartment. This term considers not only the exhaust of any initially contaminated air but also any clean air which may be used to dilute the compartment air.

(b) Compartment filter systems are treated by the equation:

$R_{ijk} = (1-F_i) * r_i$

where r_i = filter system flow rate

F_i = filter efficiency for radionuclide i

(c) Other removal factors on a case by case basis which may be deemed reasonable and conservative.

Example Calculation

(Values used below are examples only and should not be used in any actual evaluation.)

This example will look at I-131 in a compartment 6.1x6.1x7.6 = 282.80 m^2 = V

First all primary source of radionuclides needs to be identified and categorized.

(1) Flow into the compartment equals 424.8m^3 per hour with the input I-131 concen- tration equal to 2 x 10^{-10} μCi/ml (from upstream compartments) or 2.4 x 10^{-11} Ci/sec. No other sources of air either contaminated or clean air are assumed.

(2) The compartment contains a pump carrying reactor coolant with a maximum specified leakage rate of 0.000034 m^3 per hour at 273.6^{O}C.

(a) Conservatively it can be estimated based upon properties from steam tables (see note 1) that under these conditions 44% of the liquid will flash to steam and become airborne. Along with the flashing liquid it is assumed that a proportional amount of I-131 will become airborne therefore P_f = 0.44.

(b) Using the design basis iodine concentrations for reactor water from Table 11.1-2 of 0.016 μCi/gm of I-131, it is calculated that the pump is providing a source of I-131 of 5.0 x 10^{-11} Ci/sec to the air. (see Note 2)

Second, the sinks for airborne material need to be identified. This example include only exhaust which is categorized as flow out of the compartment at 150% per hour or 4.2 x 10^{-4} per second.

Therefore, for an equilibrium situation, the I-131 airborne concentration from this liquid source would be calculated from the following equation.

$A = S_1/(\lambda + R_1) + S_2/(\lambda + R_2)$, where

S_1 = source rate in Curies per second = 5.0 x 10^{-11} Ci/sec from liquid

S_2 = source rate from inflow = 2.4 x 10^{-11} Ci/sec

= isotope decay constant in units of per second = 9.977 x 10^{-7}/sec

R_1 = R_2 = removal rate constant per second (exfiltration) = 4.2 x 10^{-4} per second

A = 6.2 x 10^{-10} μCi/ml of I-131.

Notes:

1. The assumption of 44% flashing at 273.6^{O}C is extremely conservative, see Reference 1 for a discussion of fission product transport.

2. Water density assumed at 0.743 gm/cm^3 based upon standard tables for water at 273.6^{O}C.

12A.2 References

1. Paquette, et al, *Volatility of Fission Products During Reactor Accidents*, Journal of Nuclear Materials, Vol 130 Pg 129-138, 1985.

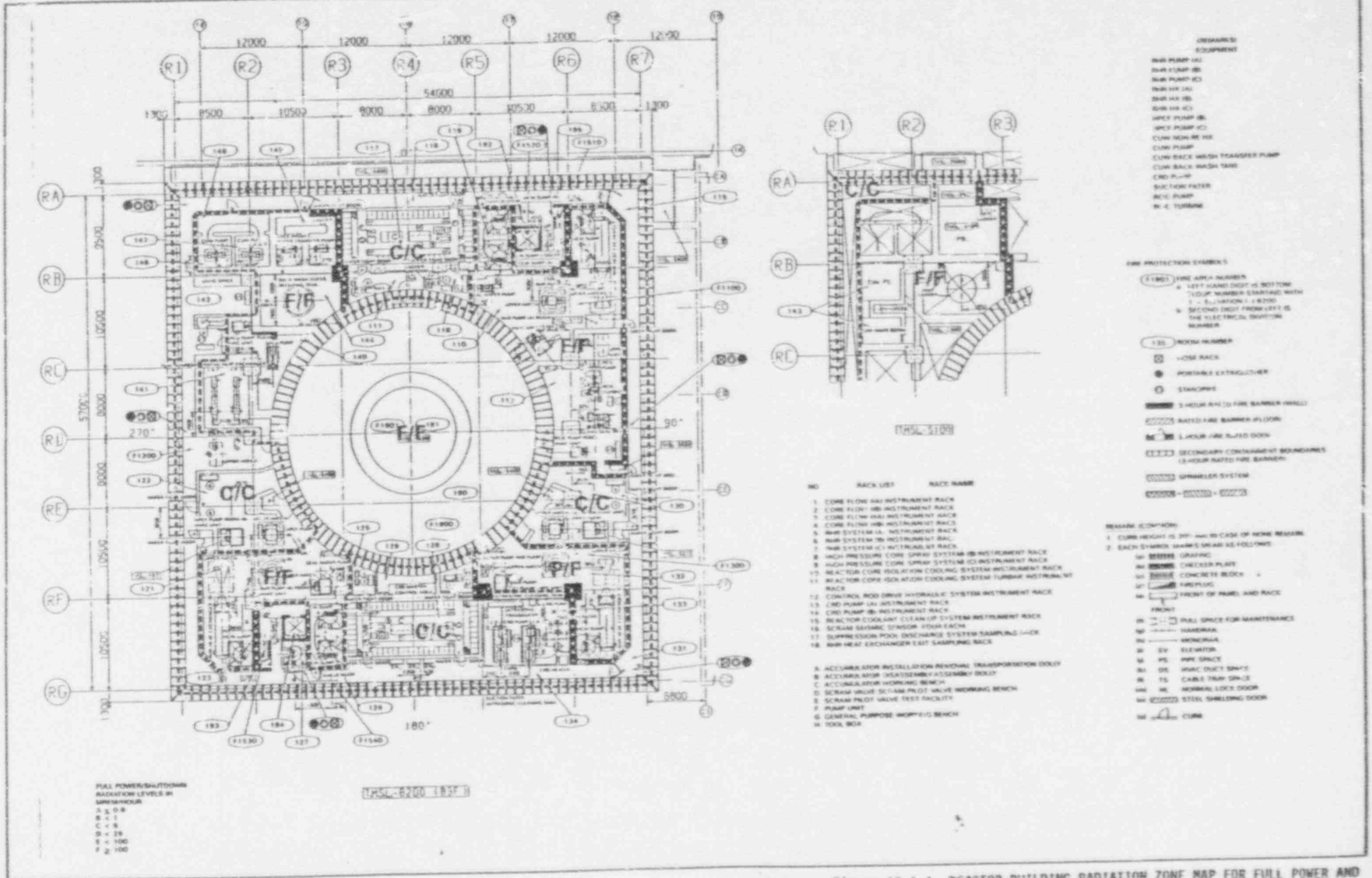


Figure 12.3-1 REACTOR BUILDING RADIATION ZONE MAP FOR FULL POWER AND SHUTDOWN OPERATIONS AT ELEVATION -8200mm (B3F)

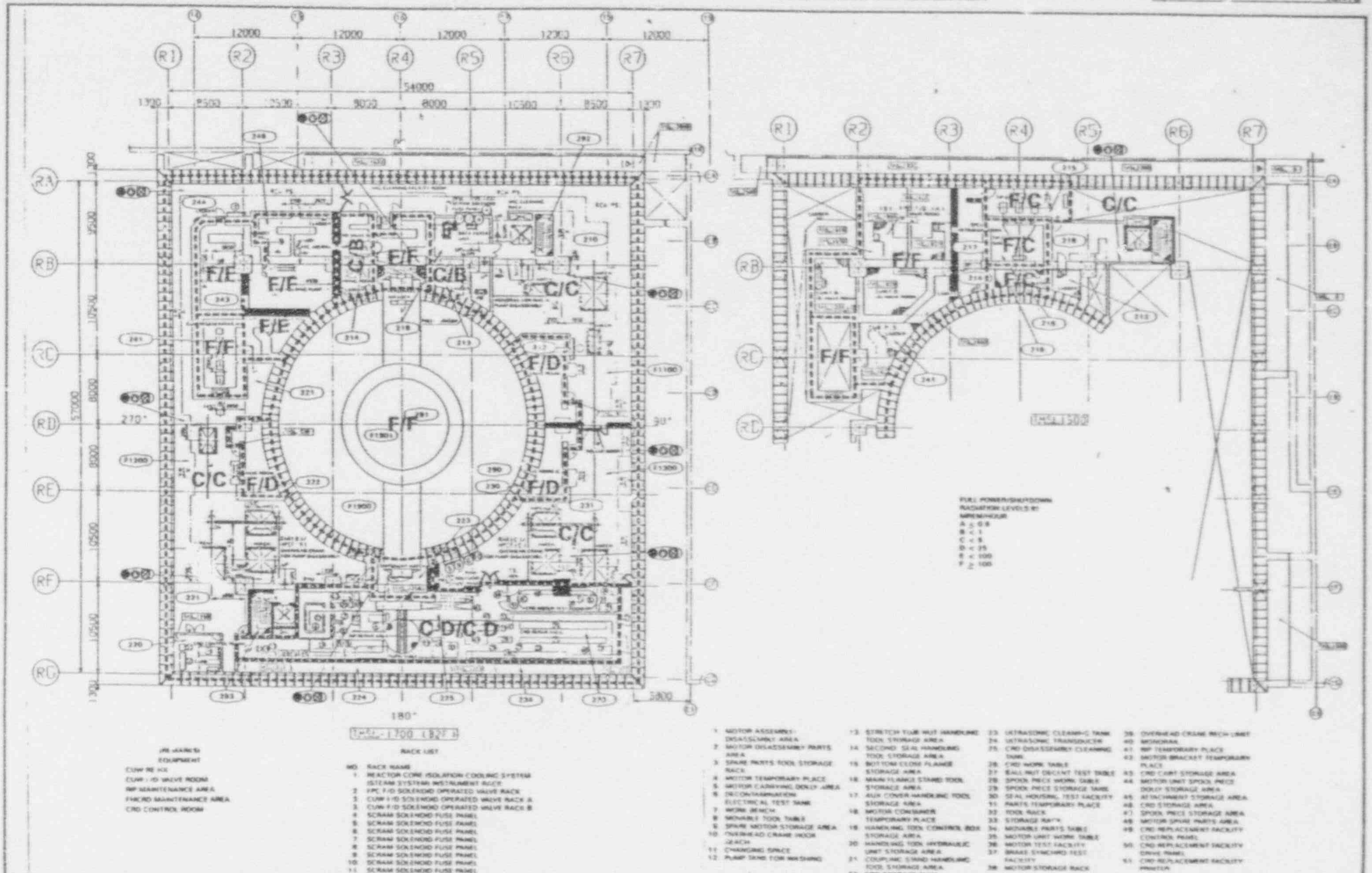


Figure 12.3-2 REACTOR BUILDING RADIATION ZONE MAP FOR FULL POWER AND SHUTDOWN OPERATIONS AT ELEVATION -1700mm (B2F)

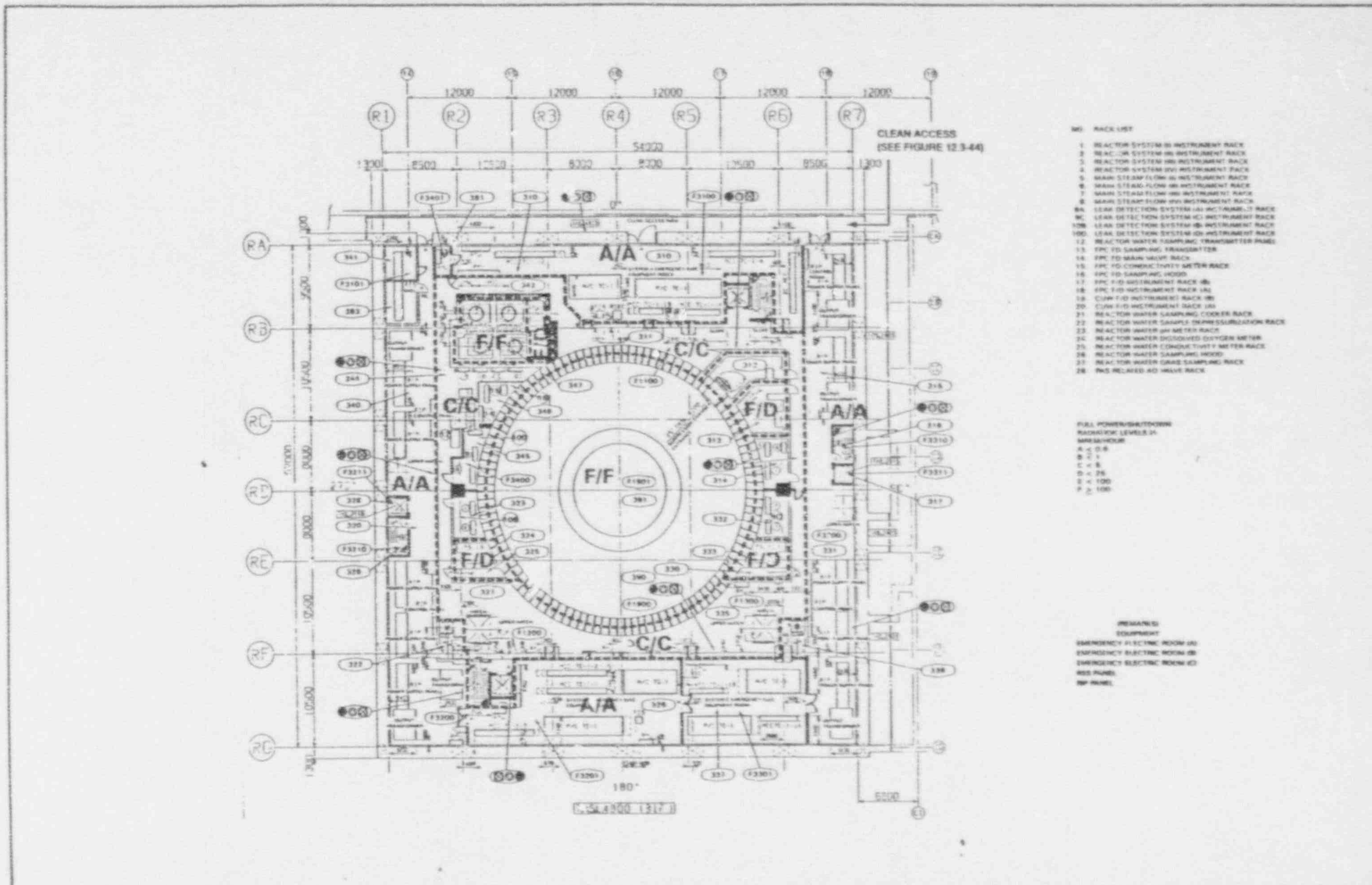


Figure 12.3-3 REACTOR BUILDING RADIATION ZONE MAP FOR FULL POWER AND SHUTDOWN OPERATIONS AT ELEVATION 4800mm (B1F)

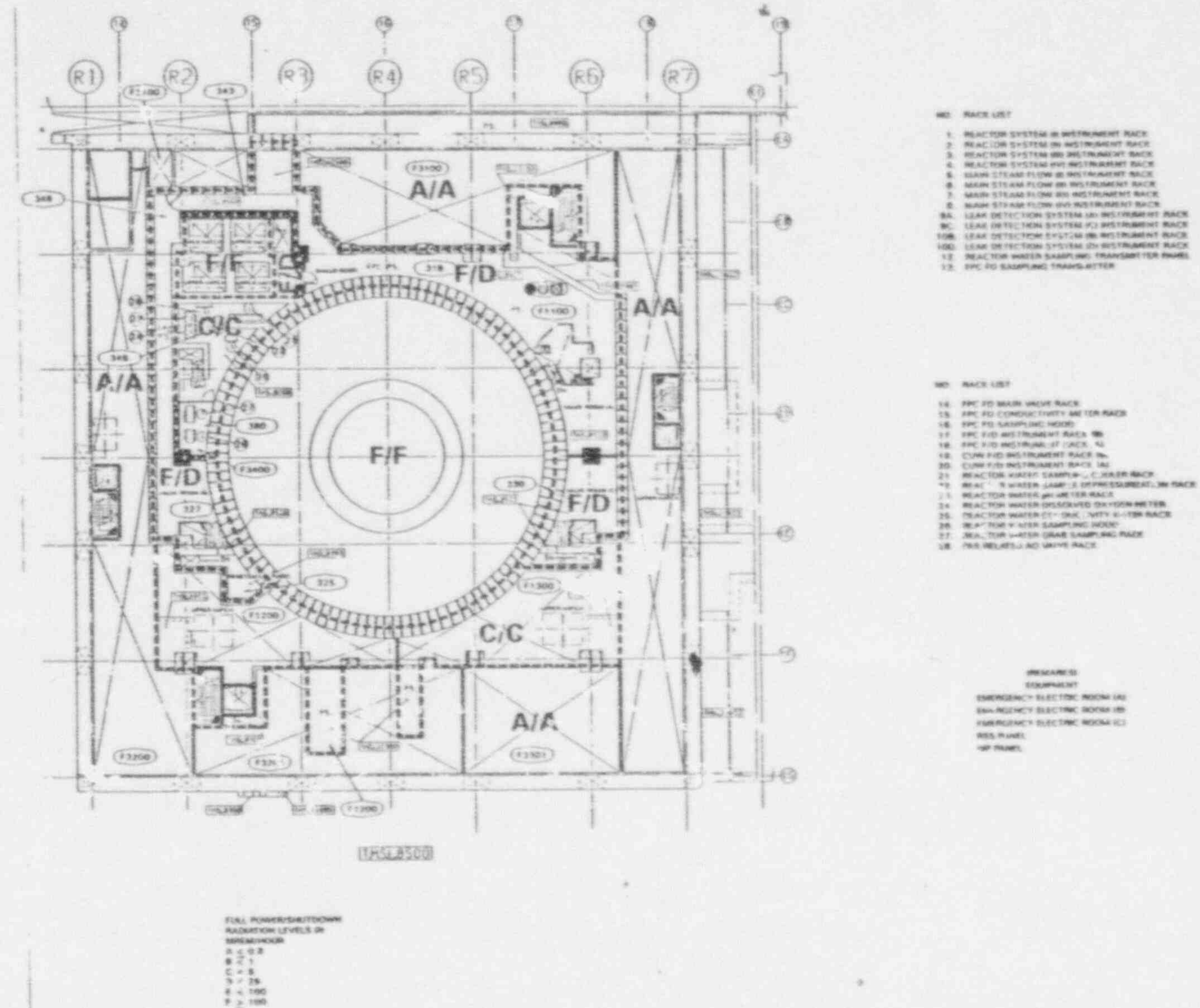


Figure 12.3-4 REACTOR BUILDING RADIATION ZONE MAP FOR FULL POWER AND SHUTDOWN OPERATIONS AT ELEVATION 8500mm (B1F)

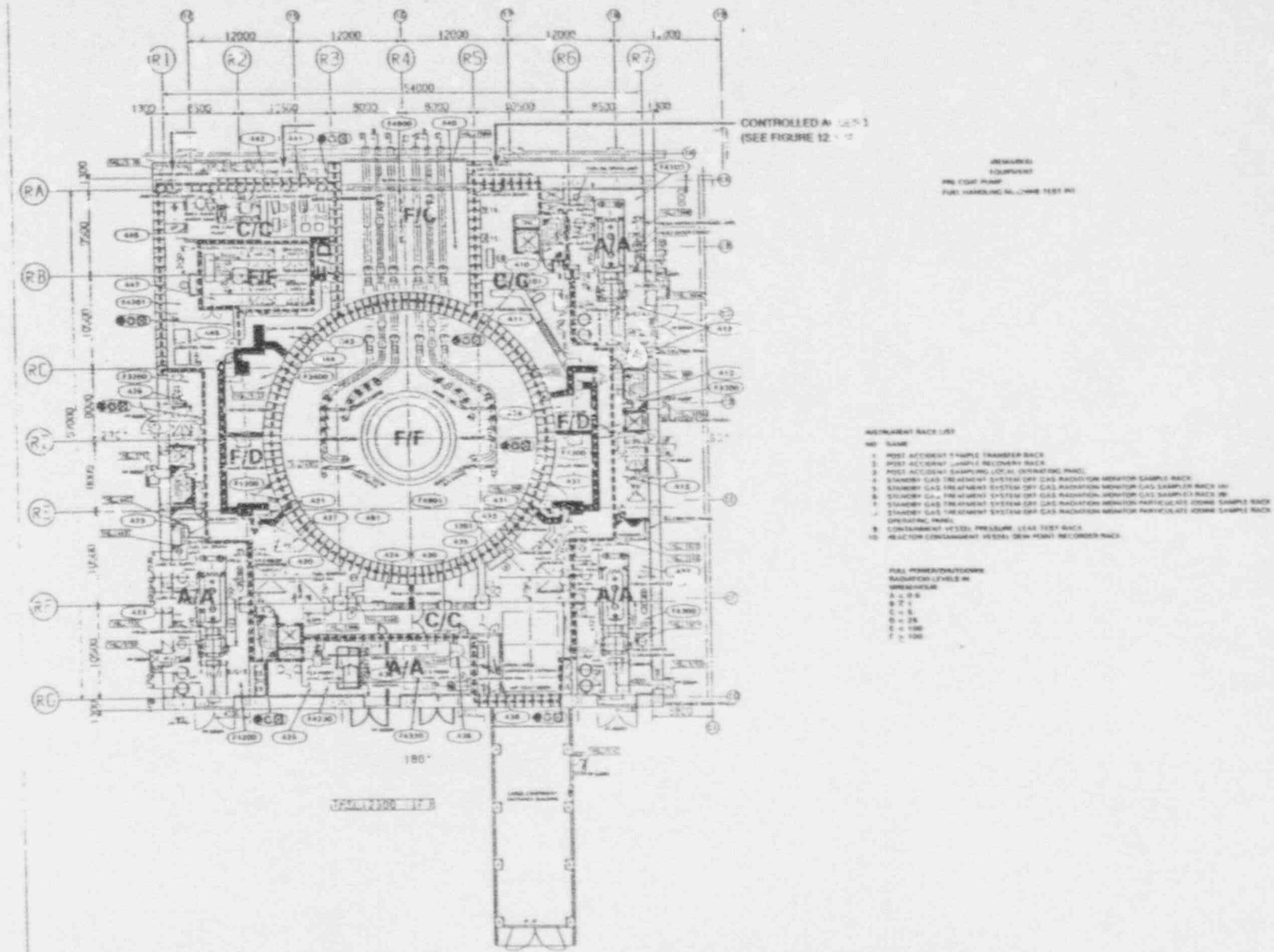


Figure 12.3-5 REACTOR BUILDING RADIATION ZONE MAP FOR FULL POWER AND SHUTDOWN OPERATIONS AT ELEVATION 12300mm (1F)

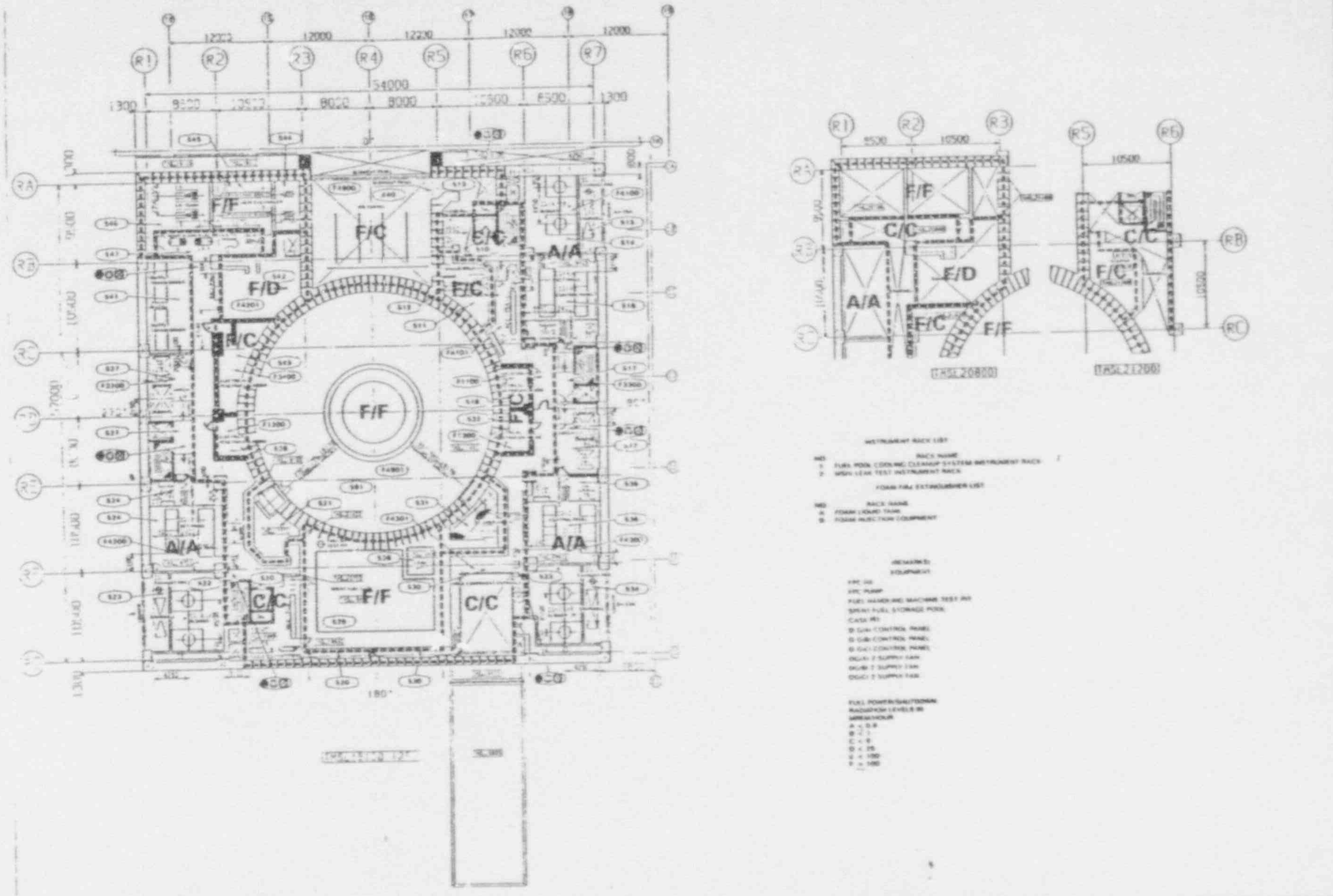


Figure 12.3-6 REACTOR BUILDING RADIATION ZONE MAP FOR FULL POWER AND SHUTDOWN OPERATIONS AT ELEVATION 18100mm (2F)

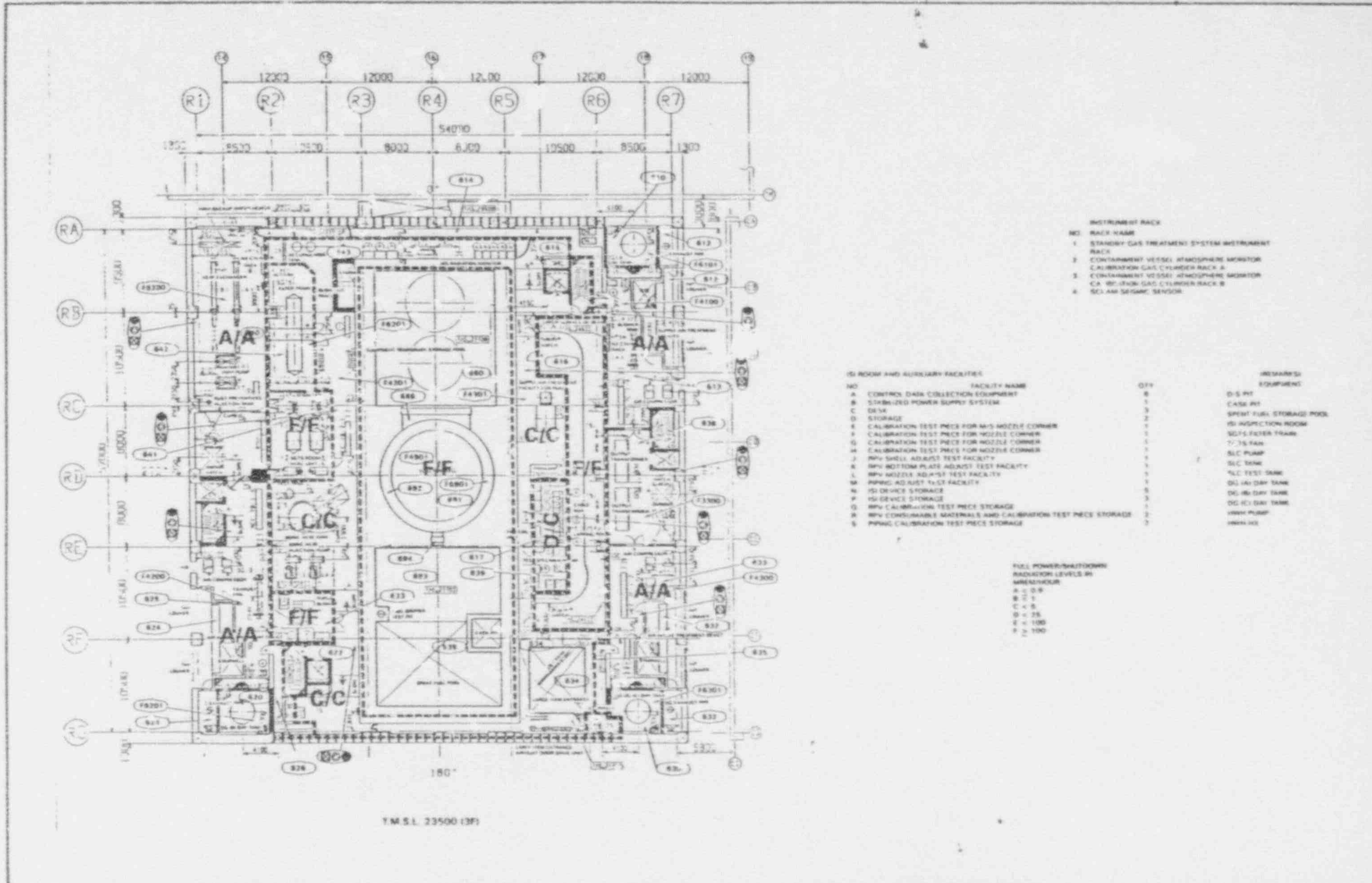


Figure 12.3-7 REACTOR BUILDING RADIATION ZONE MAP FOR FULL POWER AND SHUTDOWN OPERATIONS AT ELEVATION 23500mm (3F)

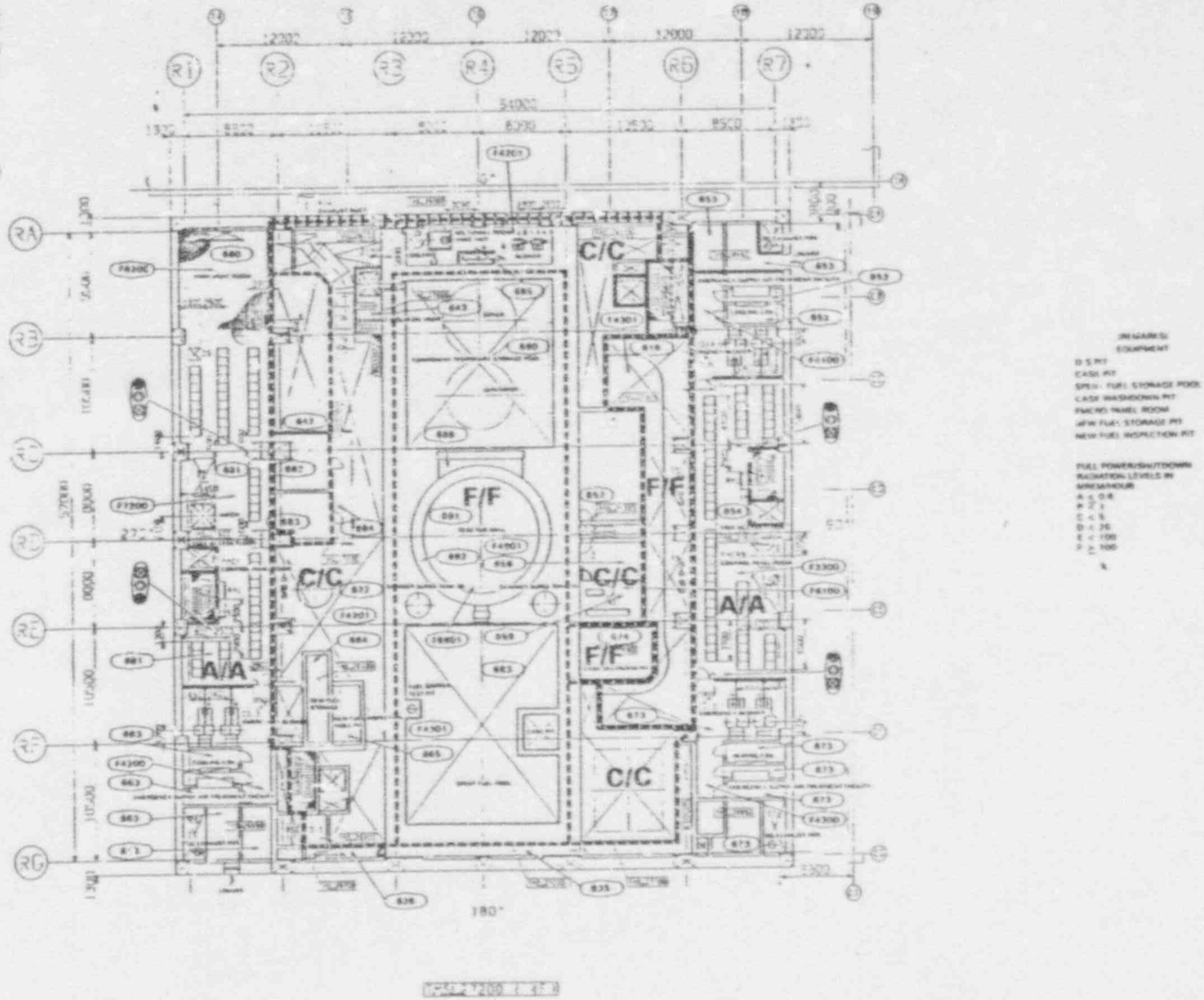
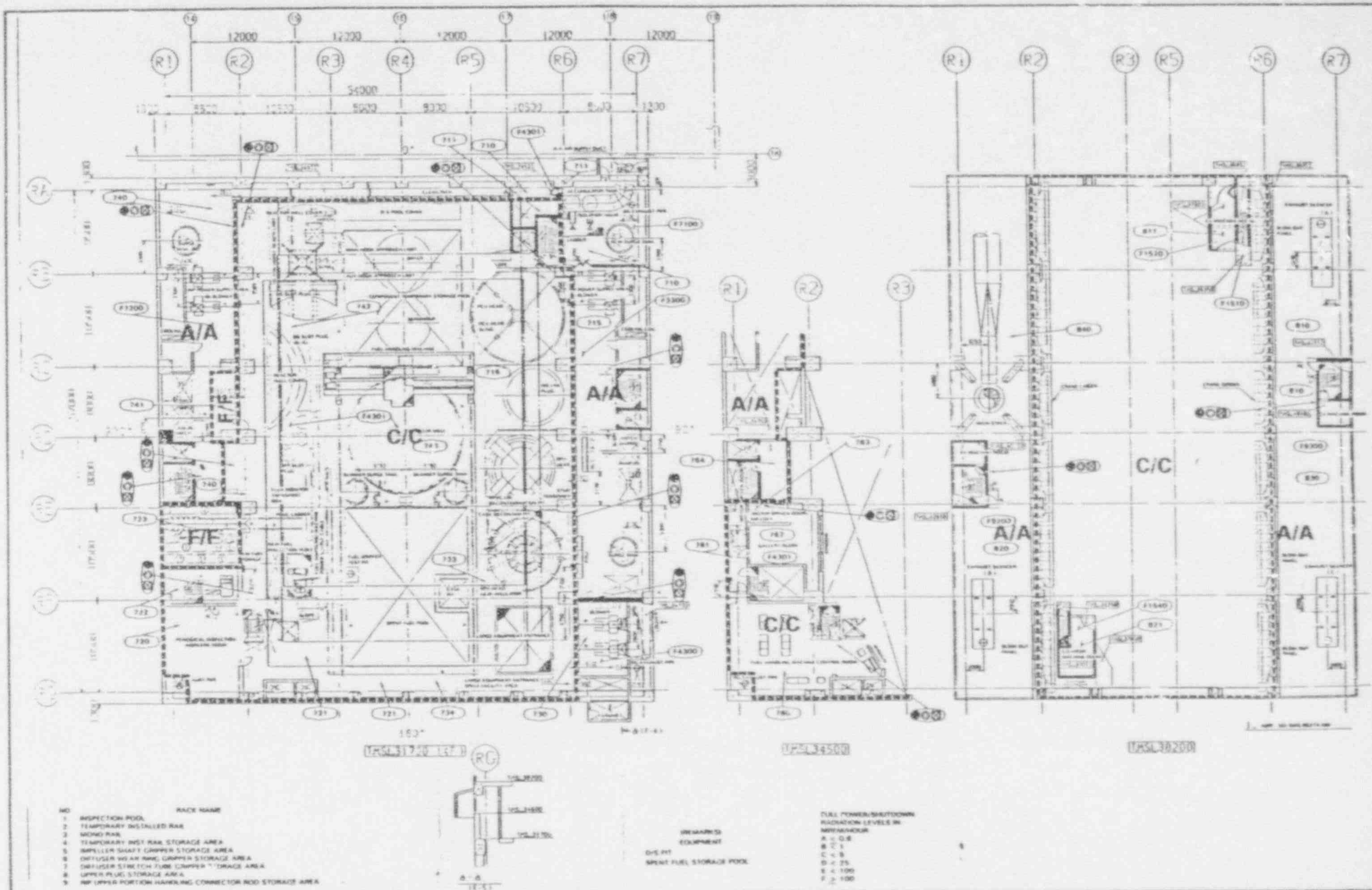


Figure 12.3-8 REACTOR BUILDING RADIATION ZONE MAP FOR FULL POWER AND SHUTDOWN OPERATIONS AT ELEVATION 27200mm (4F)



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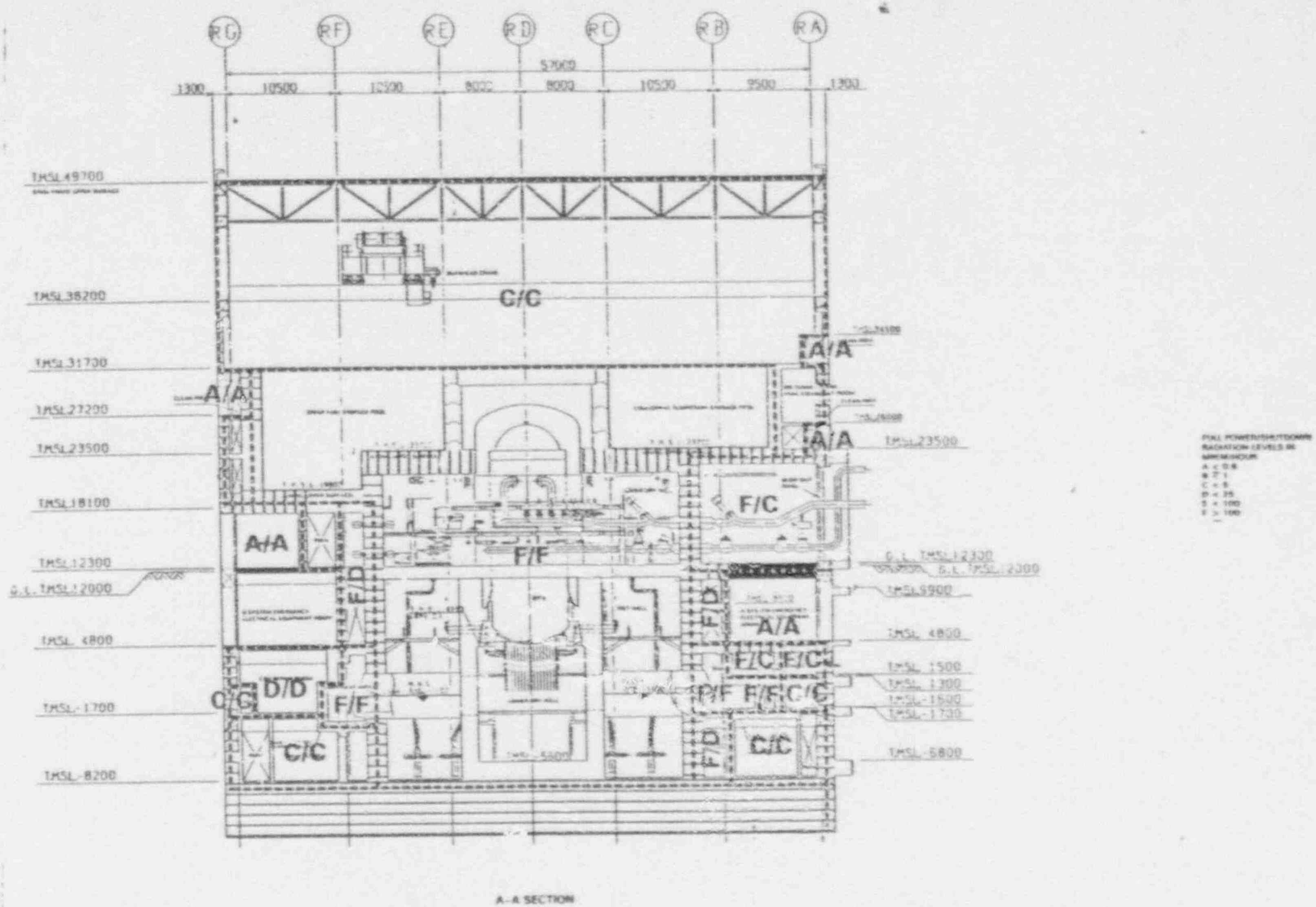
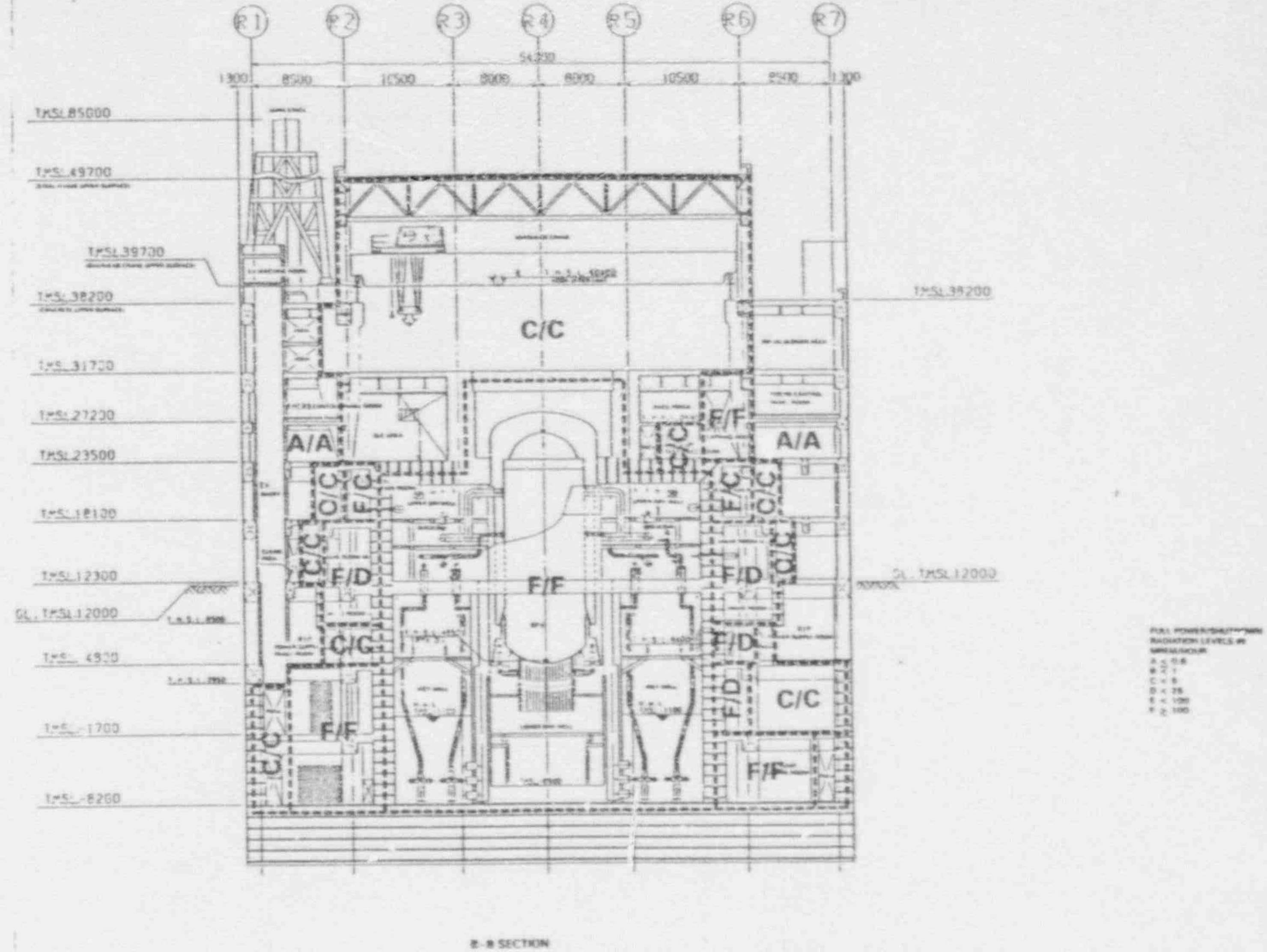


Figure 12.3-10 REACTOR BUILDING RADIATION ZONE MAP FOR FULL POWER AND SHUTDOWN OPERATIONS AT CROSS SECTION VIEW A-A



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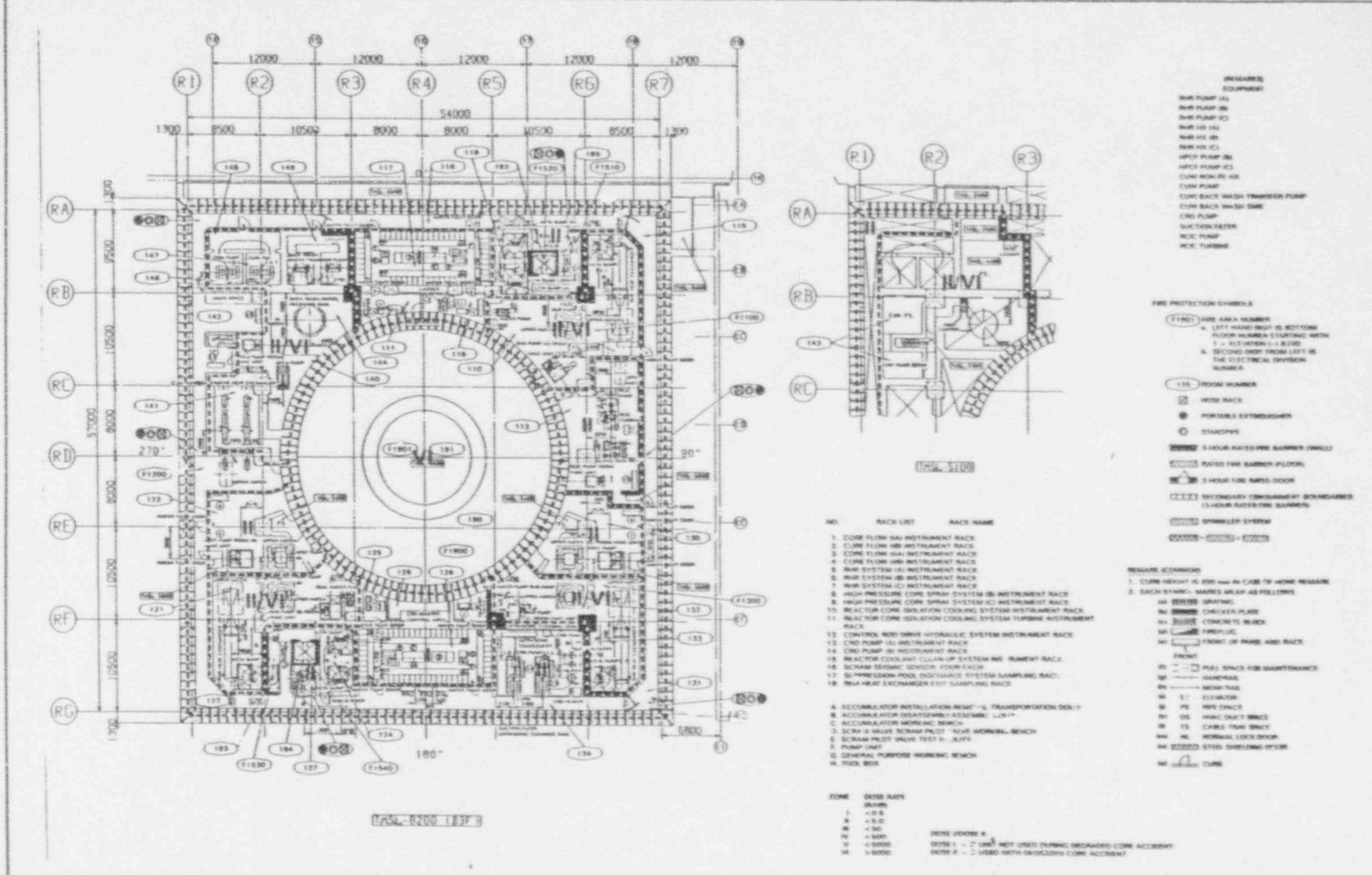


Figure 12.3–12 REACTOR BUILDING RADIATION ZONE MAP POST LOCA AT ELEVATION (–)8200mm (B3F)

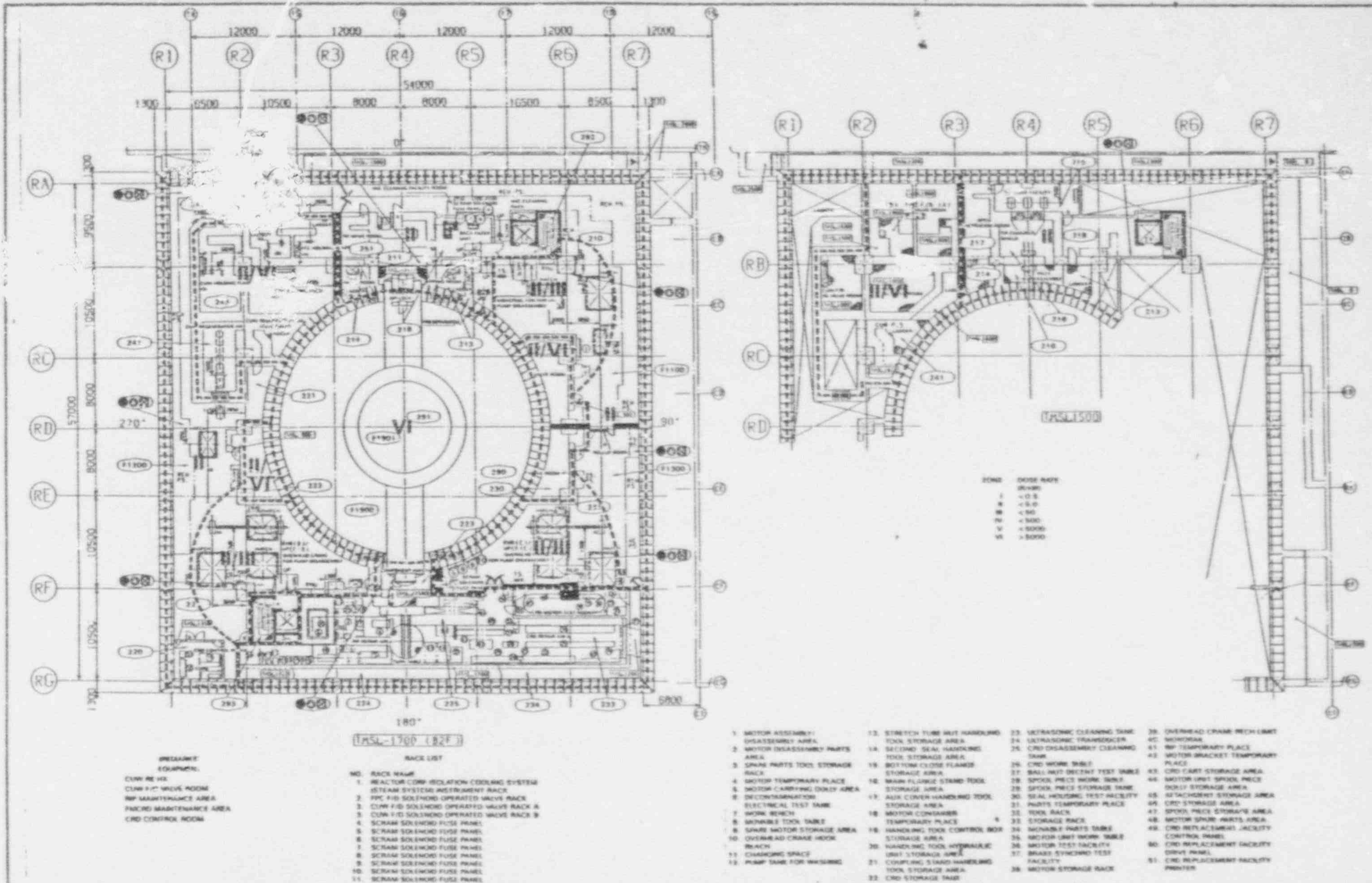
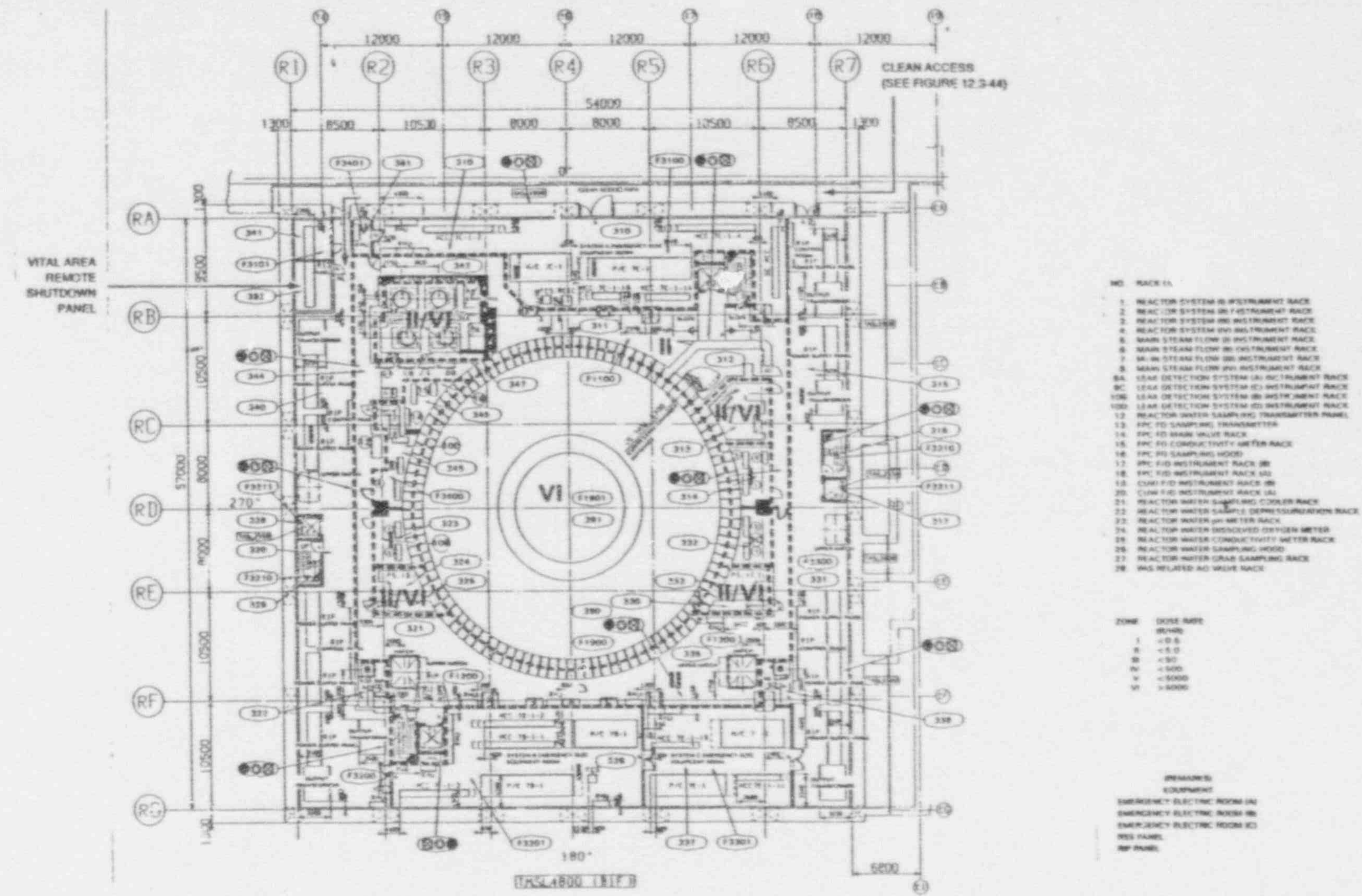


Figure 12.3-13 REACTOR BUILDING RADIATION ZONE MAP POST LOCA AT ELEVATION (–) 1700mm (B2F)



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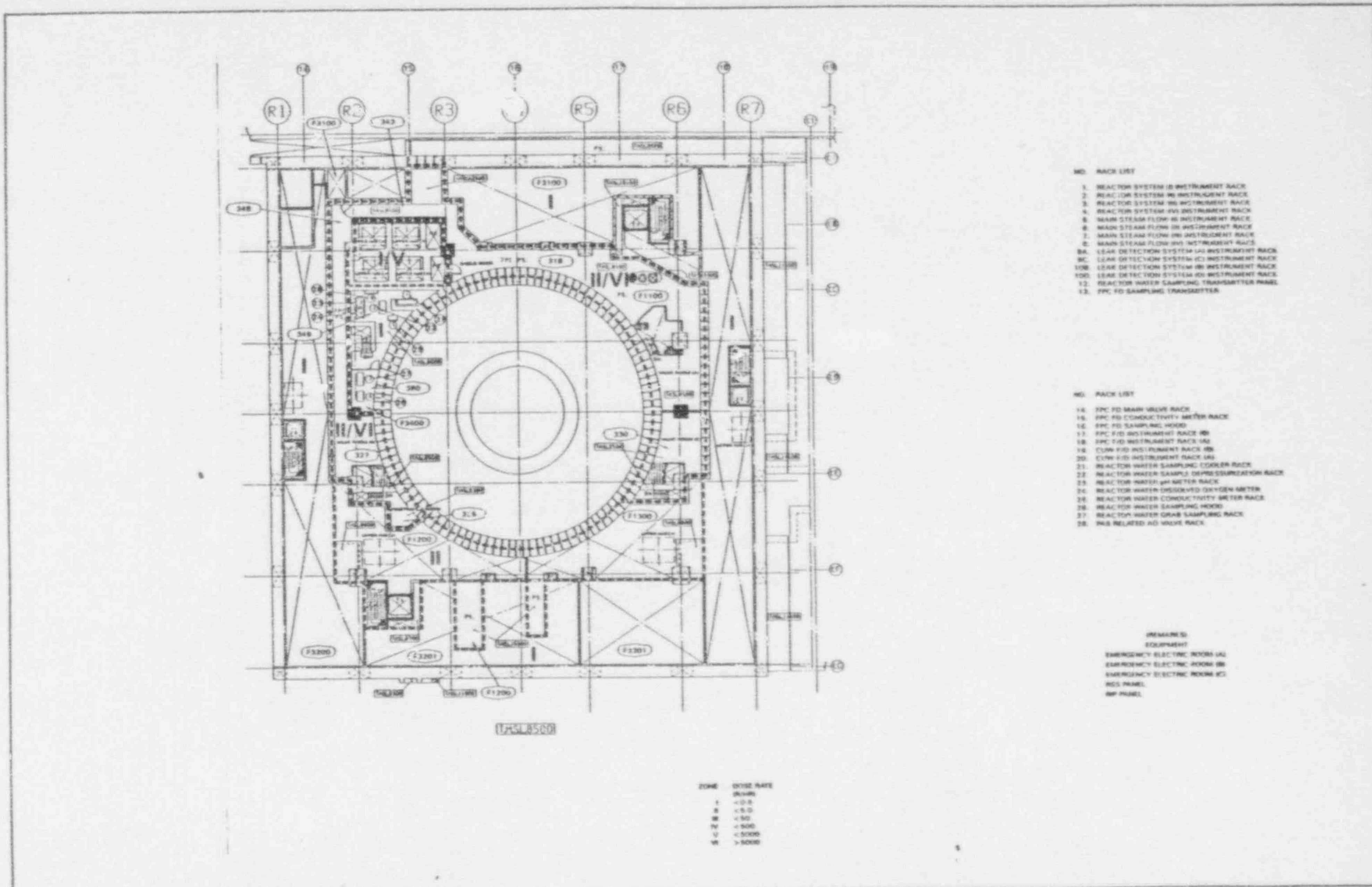

Figure 12.3-15 REACTOR BUILDING RADIATION ZONE MAP POST LOCA AT ELEVATION 8500mm

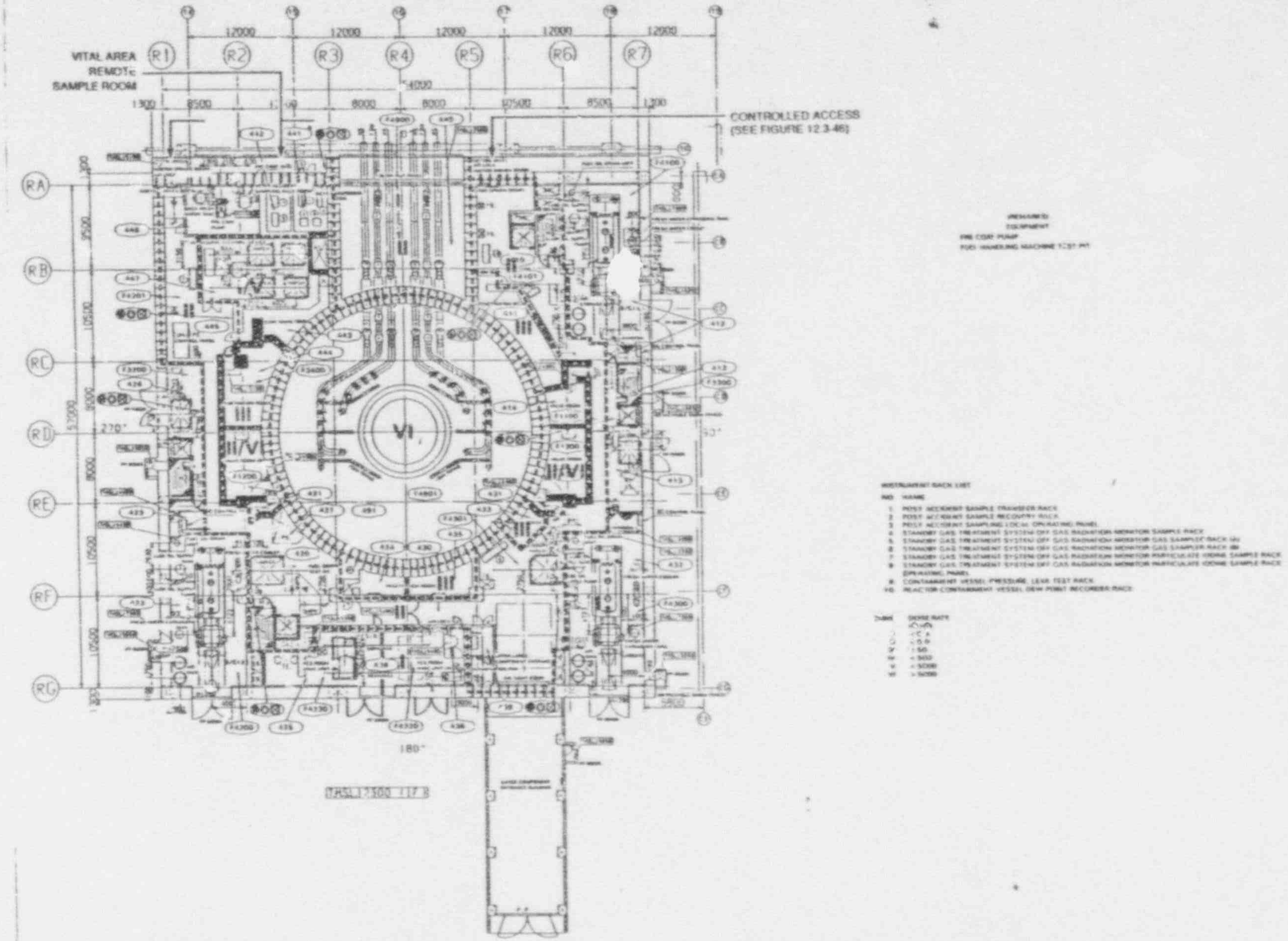

Figure 12.3-15 REACTOR BUILDING RADIATION ZONE MAP POST LOCA AT ELEVATION 12300mm (1F)

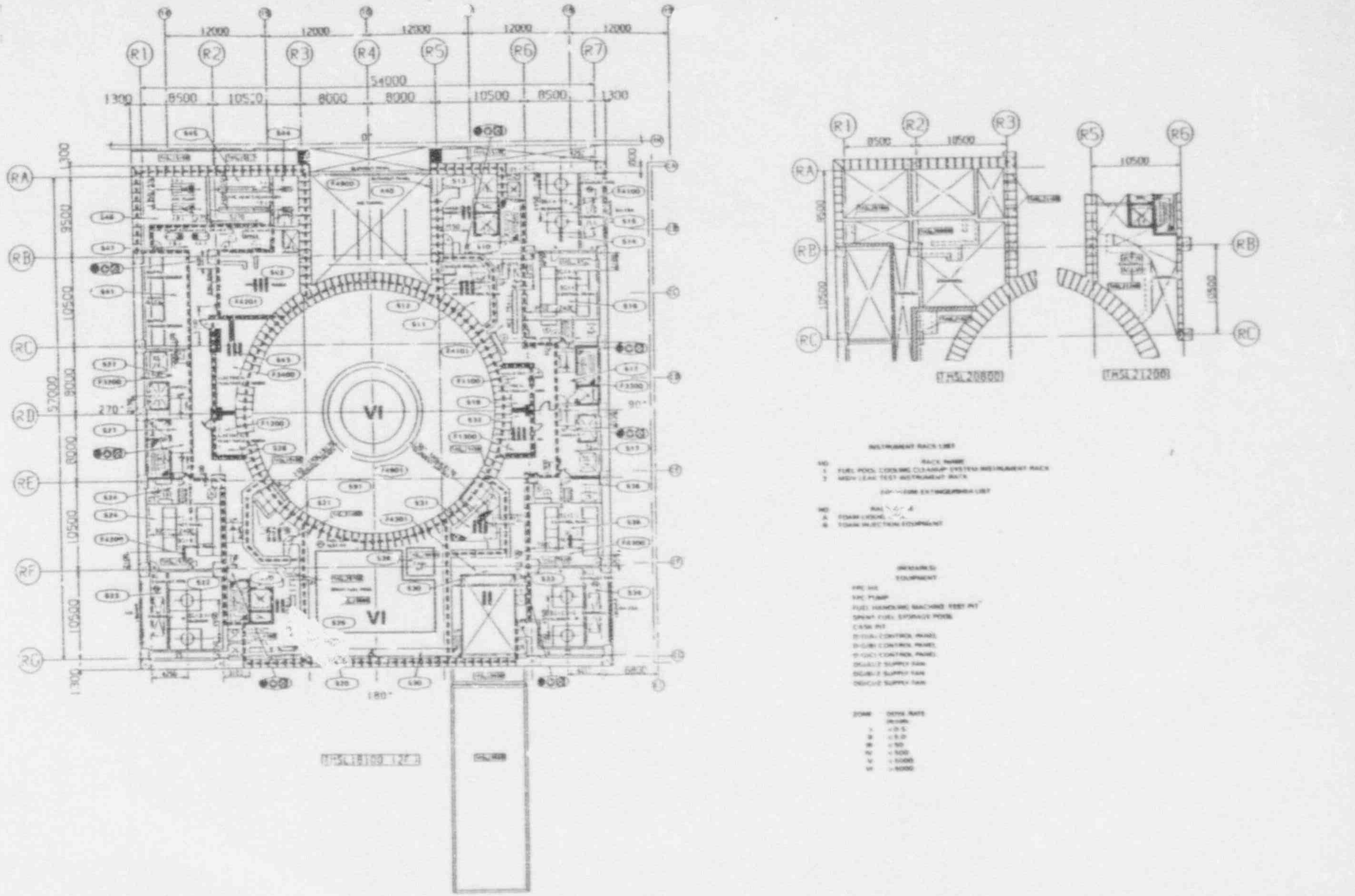

Figure 12.3-17 REACTOR BUILDING RADIATION ZONE MAP POST LOCA AT ELEVATION 18100mm (2F)

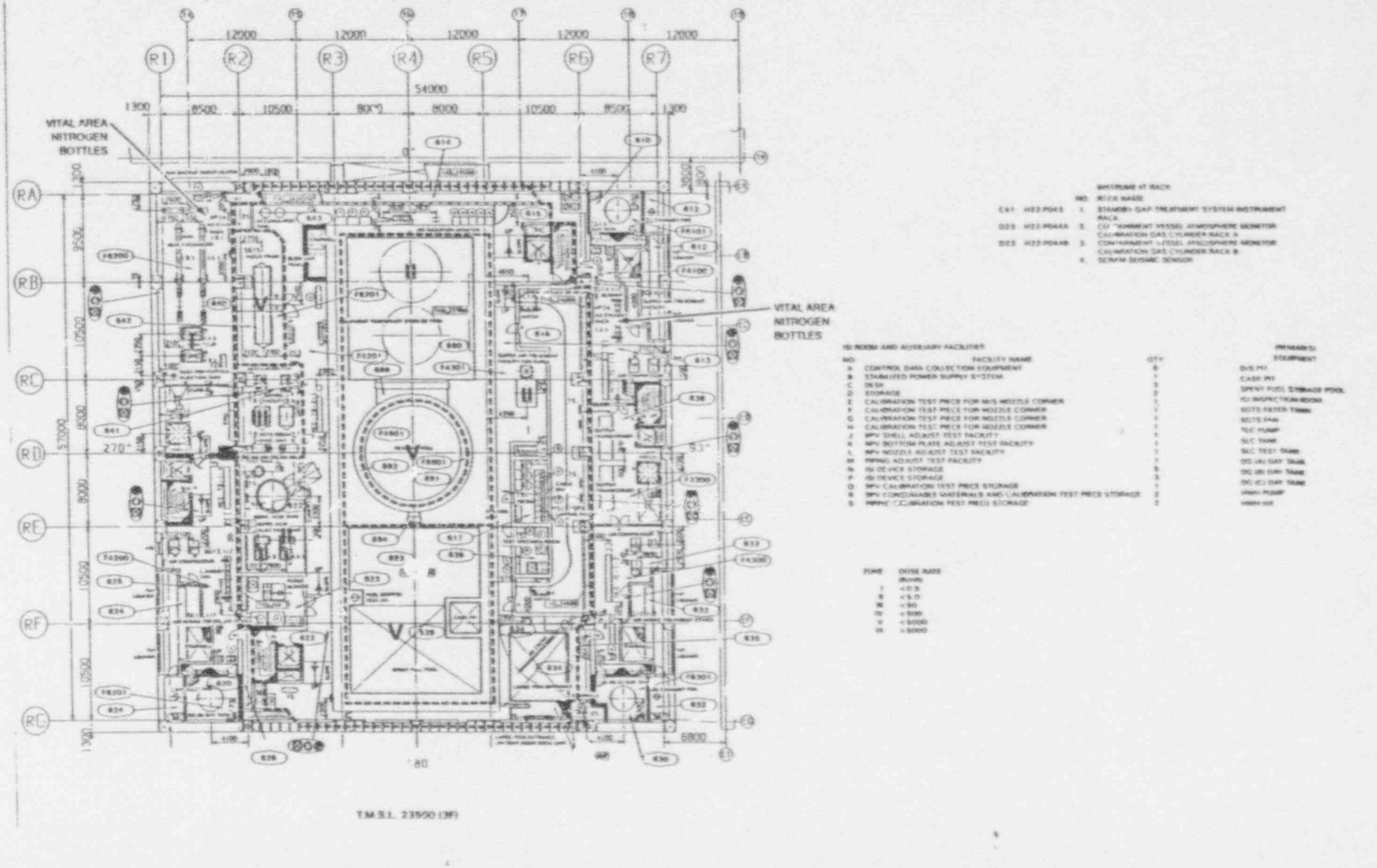

Figure 12.3–18 REACTOR BUILDING RADIATION ZONE MAP POST LOCA AT ELEVATION 23500mm (3F)

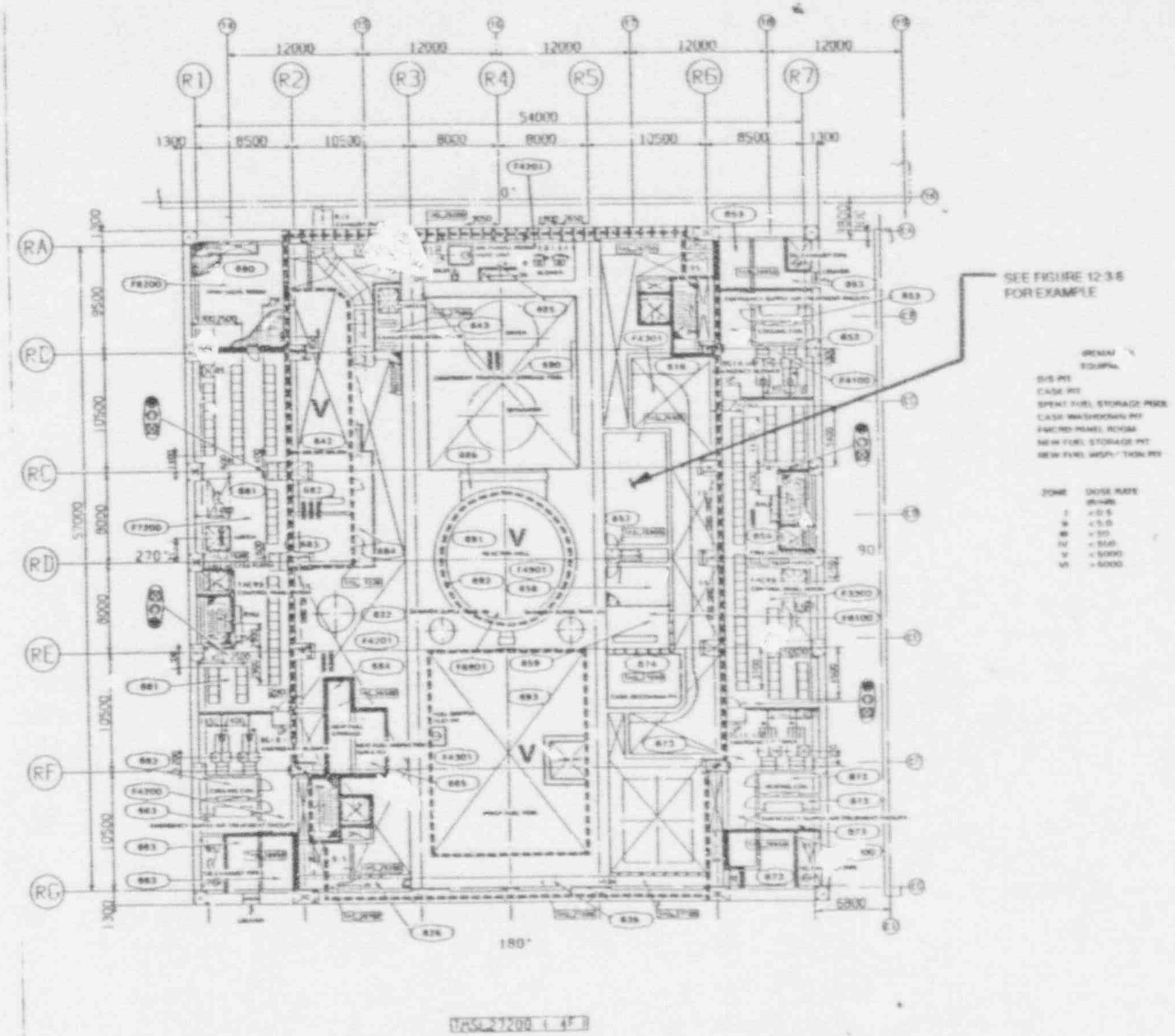

Figure 12.3–19 REACTOR BUILDING RADIATION ZONE MAP POST LOCA AT ELEVATION 27200mm (4F)

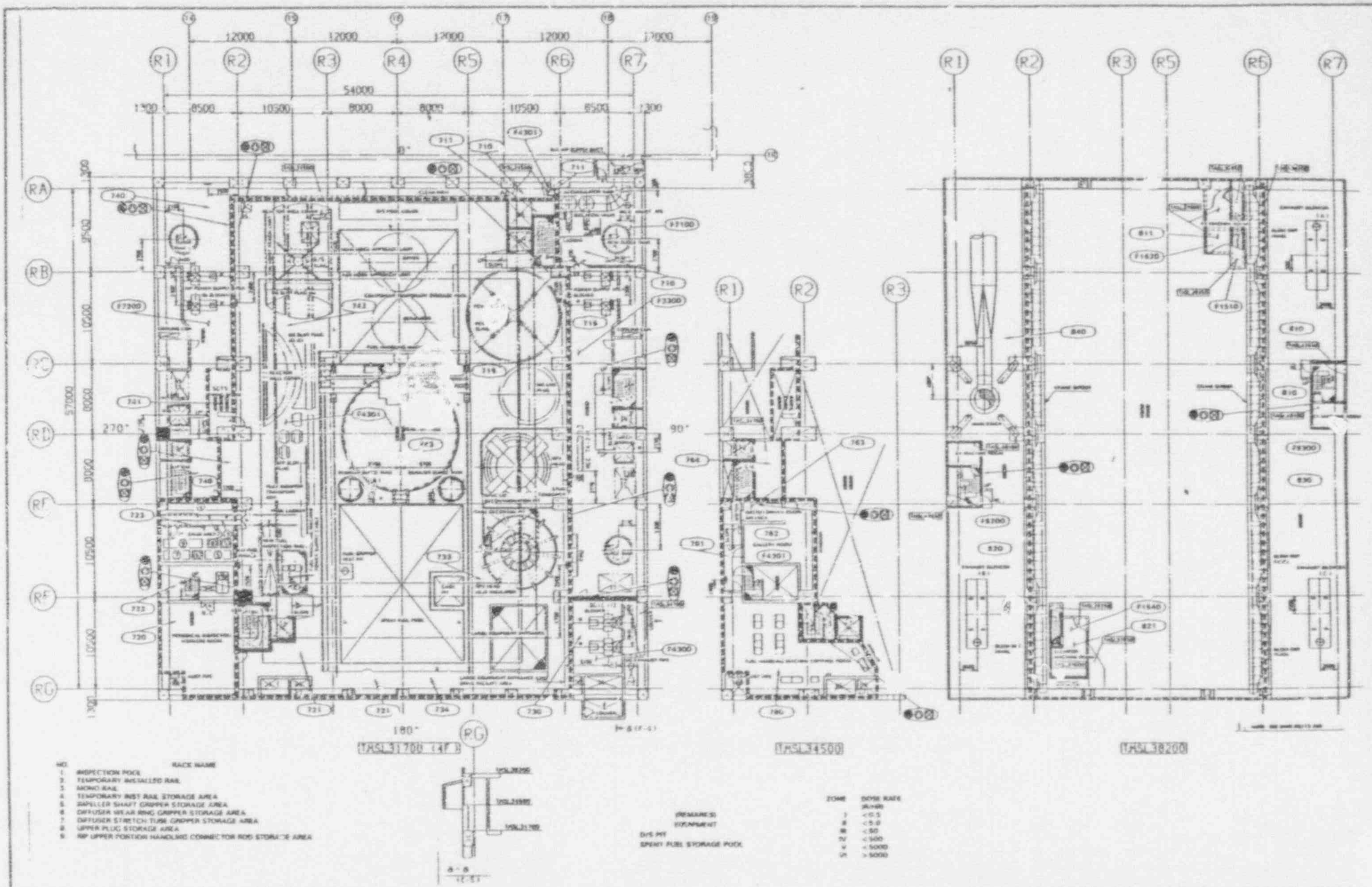

Figure 12.3–20 REACTOR BUILDING RADIATION ZONE MAP POST LOCA AT ELEVATION 31700mm

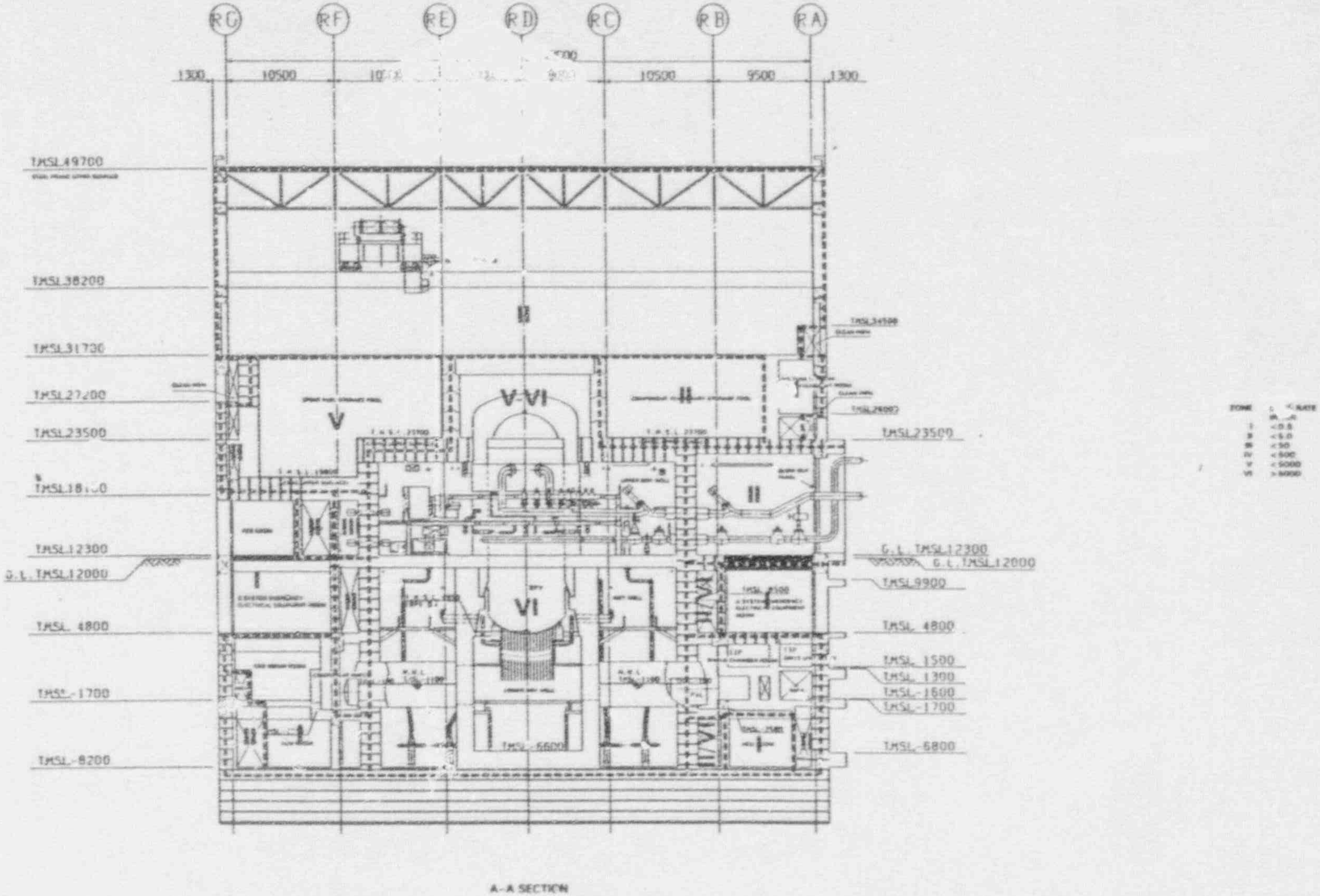

Figure 12.3–21 REACTOR BUILDING RADIATION ZONE MAP POST LOCA AT CROSS SECTION A–A

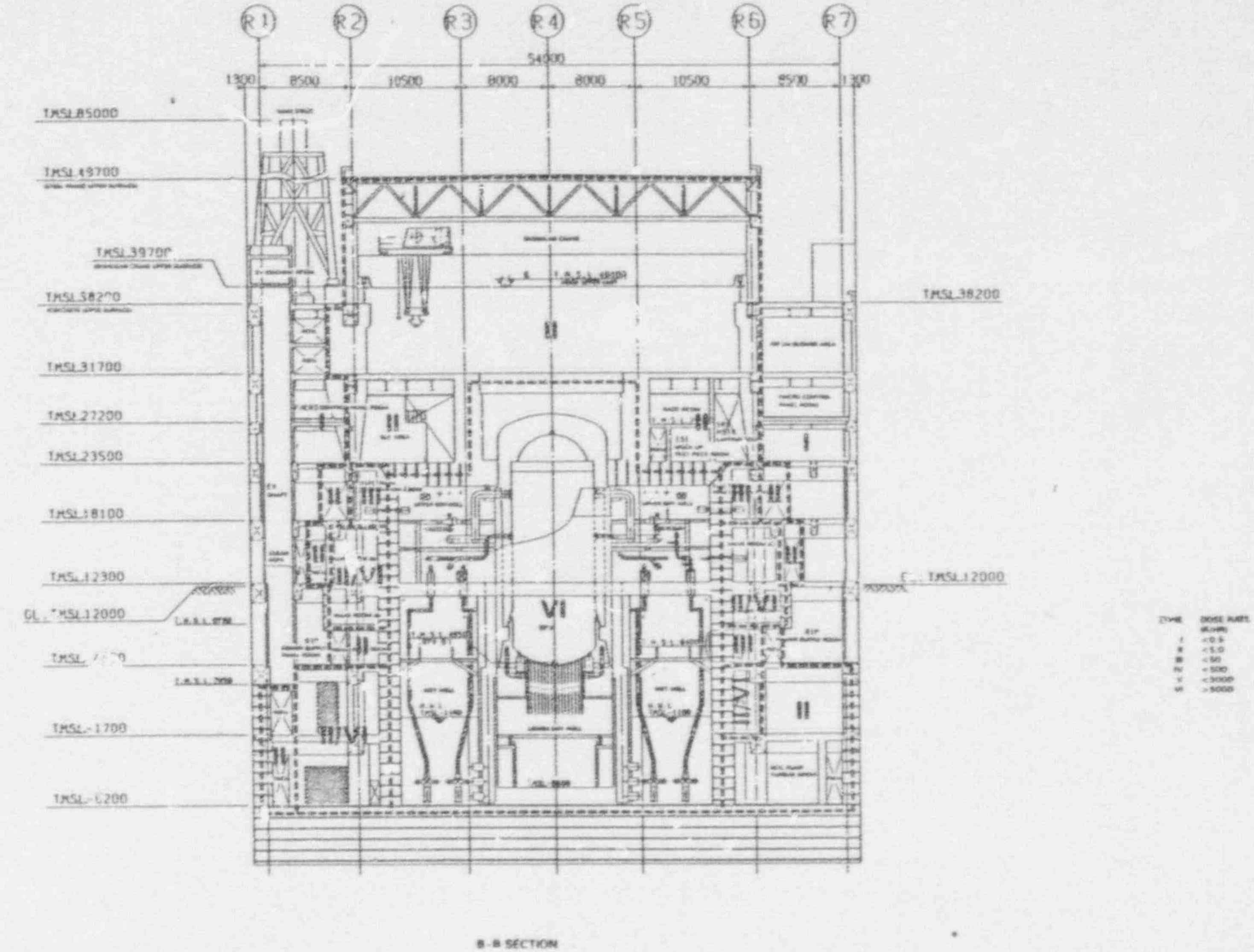

Figure 12.3-22 REACTOR BUILDING RADIATION ZONE MAP POST LOCA AT CROSS SECTION B–B

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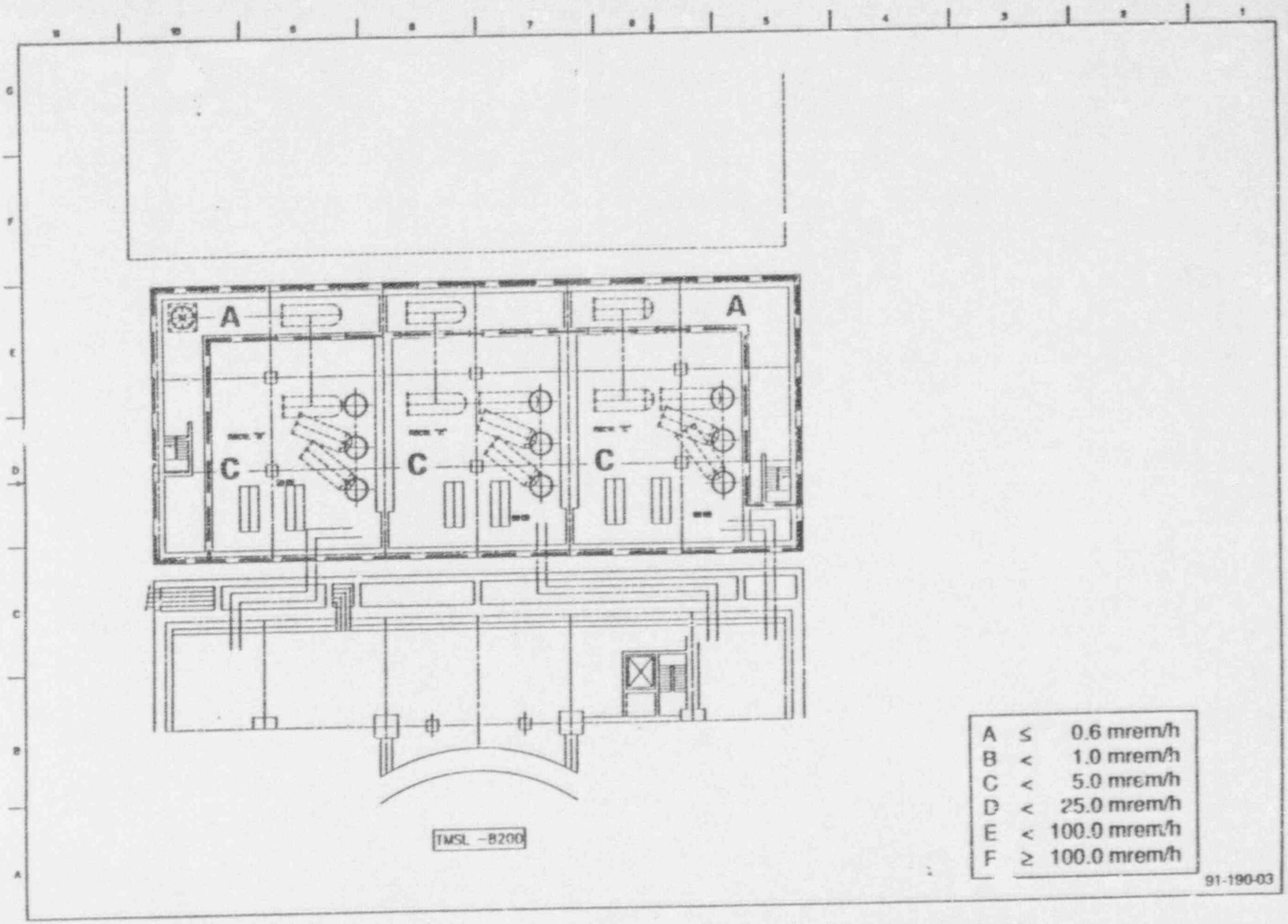

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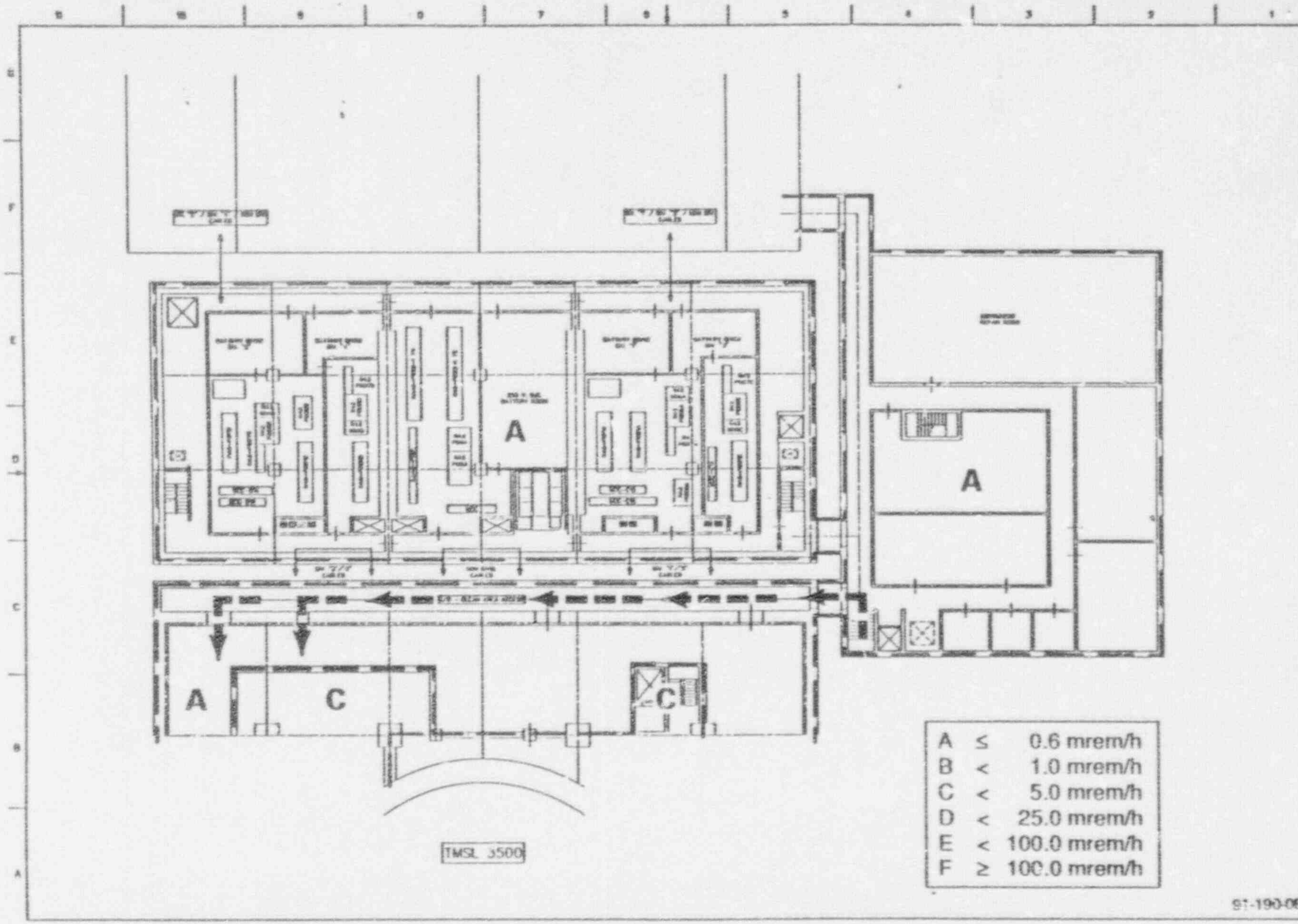

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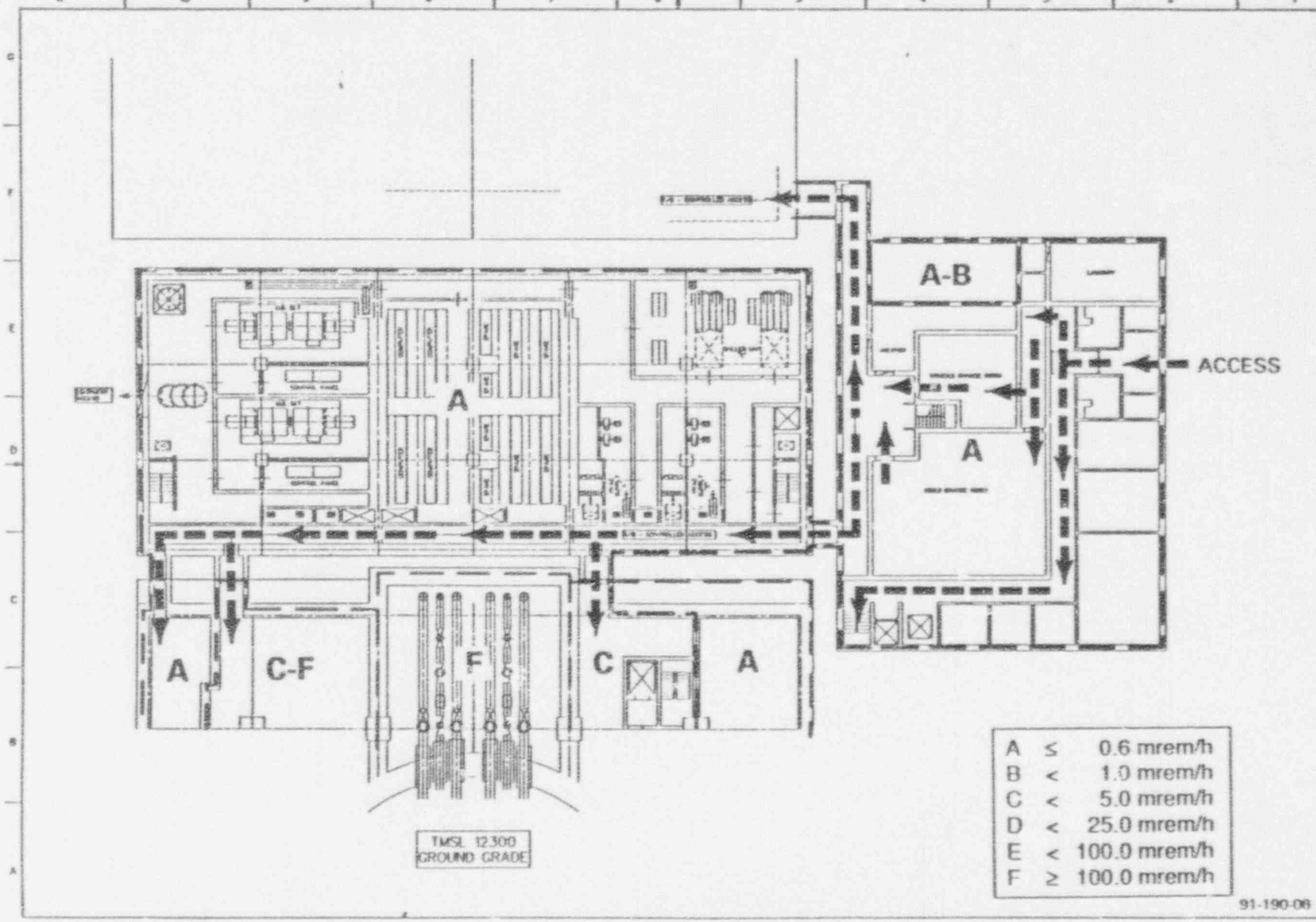


Figure 12.3-46 CONTROL BUILDING, RADIATION ZONE, NORMAL OPERATIONS AT FLOOR LEVEL 12,300mm

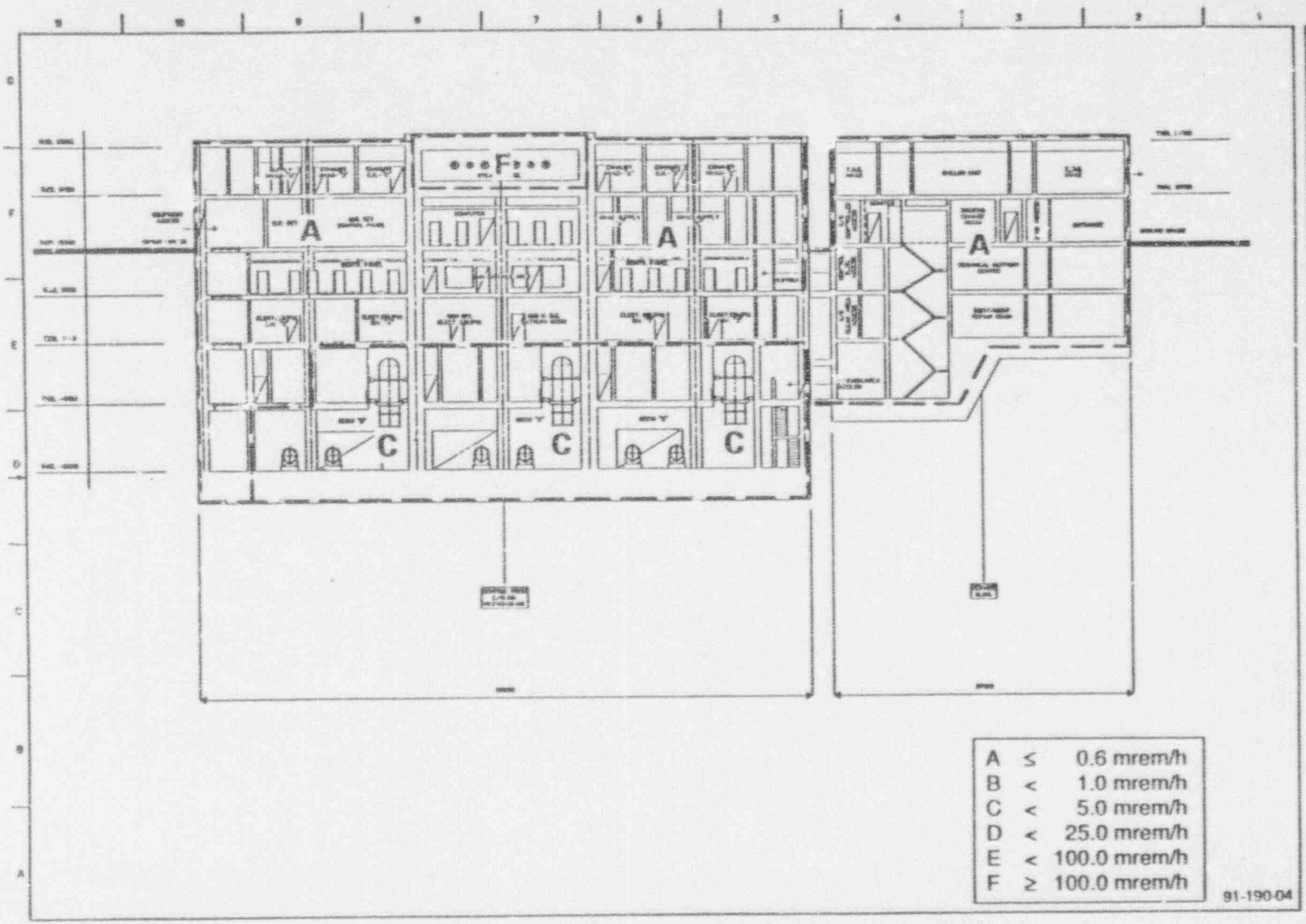


Figure 12.3-48 CONTROL BUILDING, RADIATION ZONE, NORMAL OPERATION, SIDE VIEW

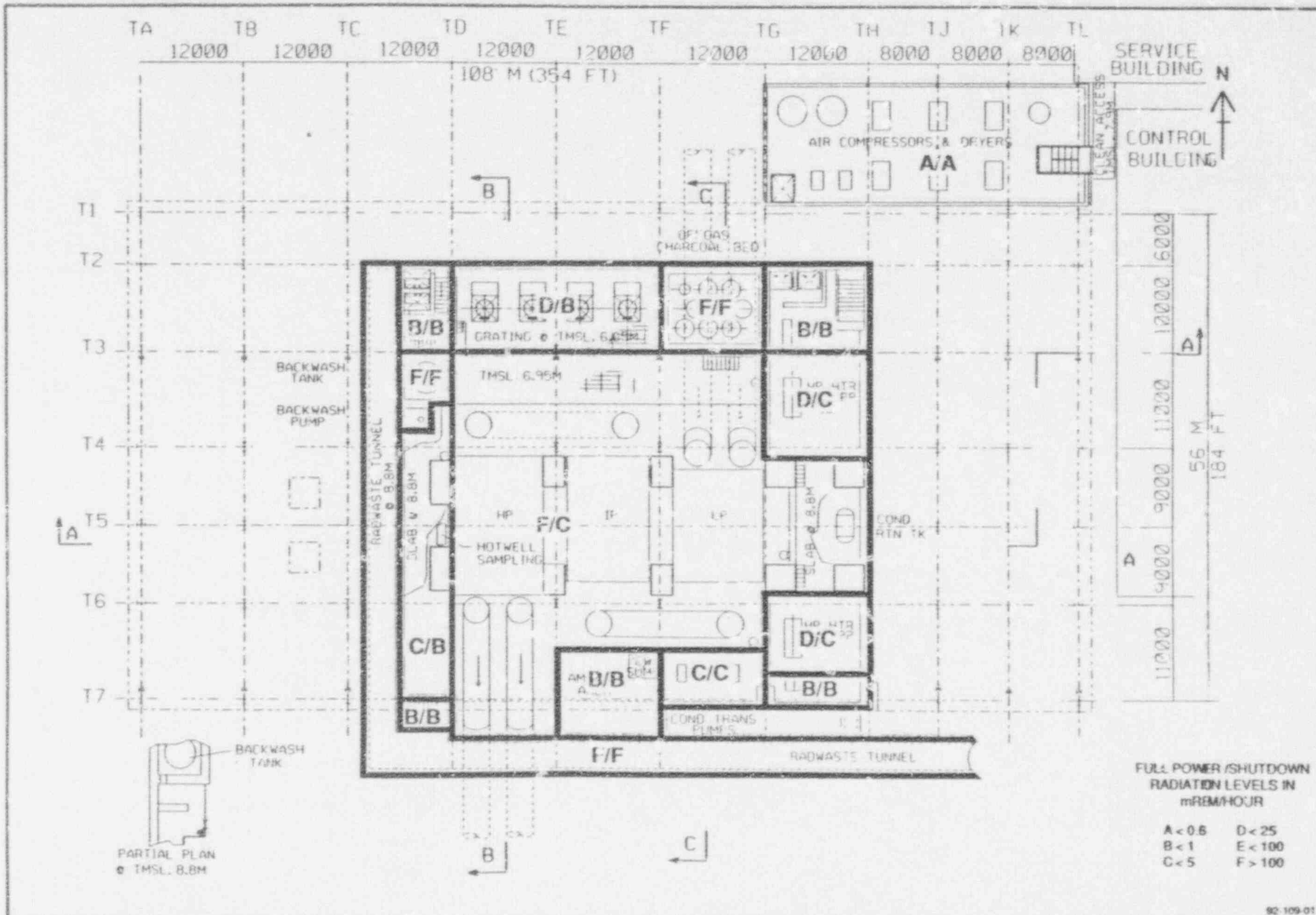
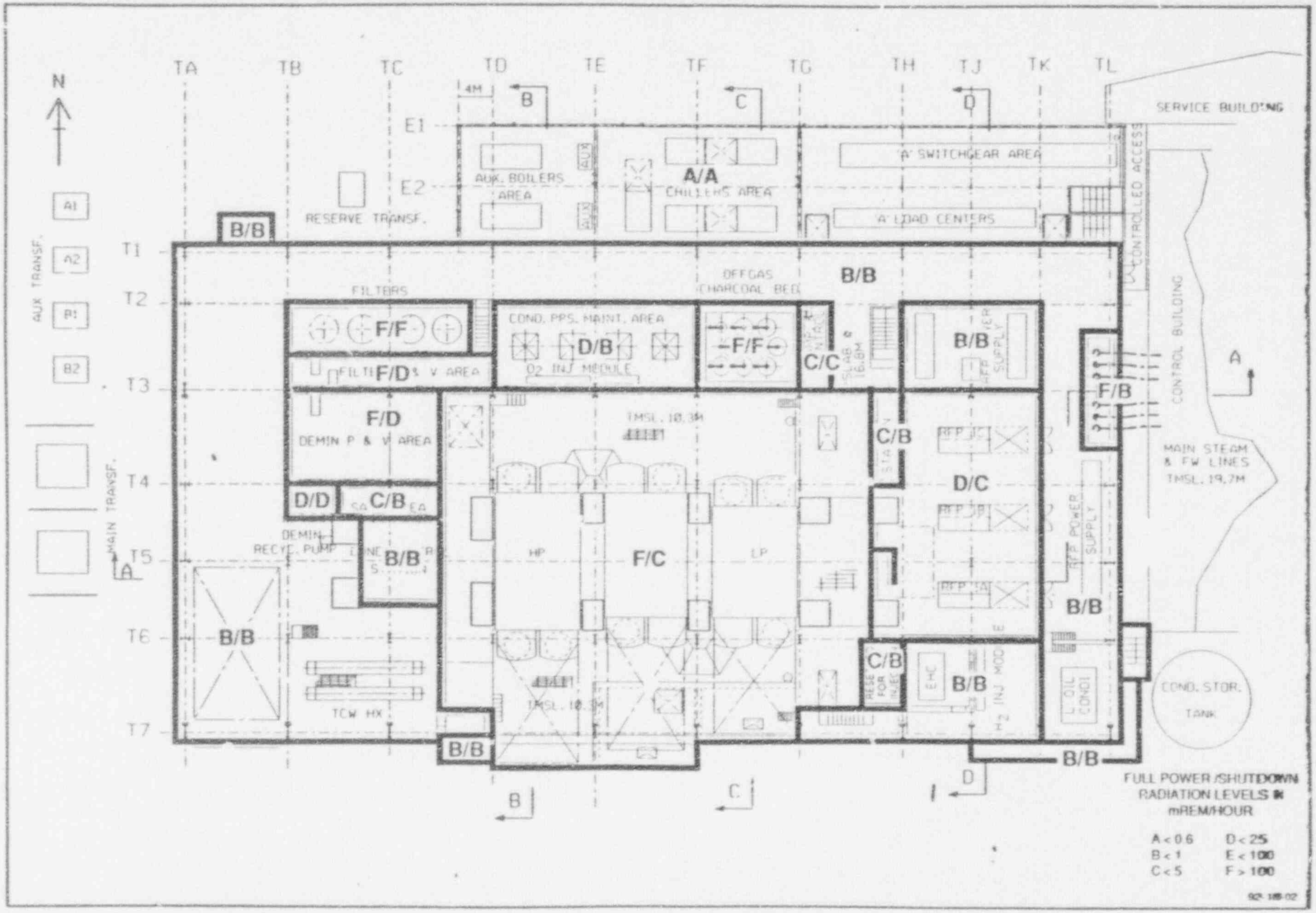


Figure 12.3-49 TURBINE BUILDING RADIATION ZONE AT ELEVATION TMSL 5.3M



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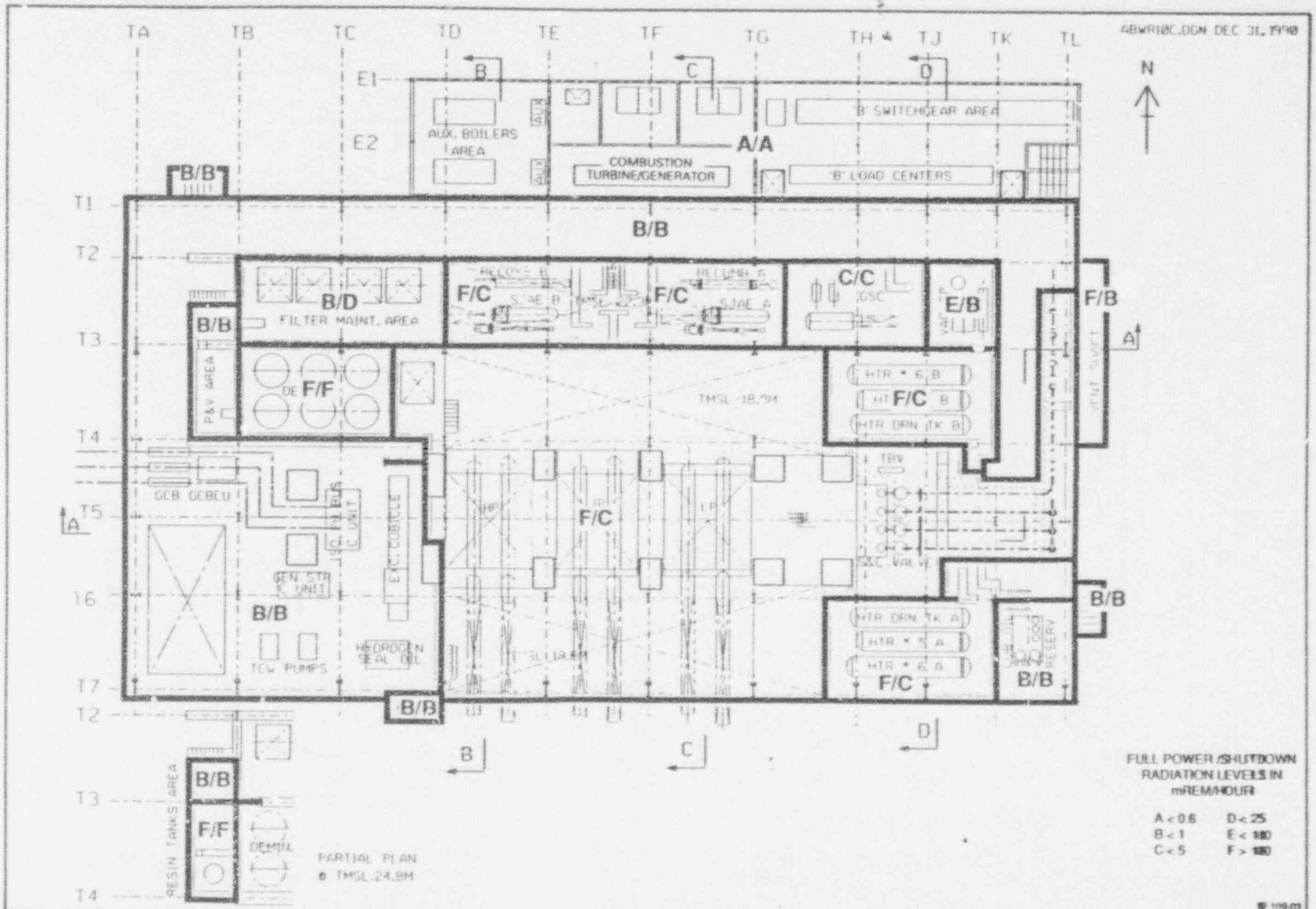

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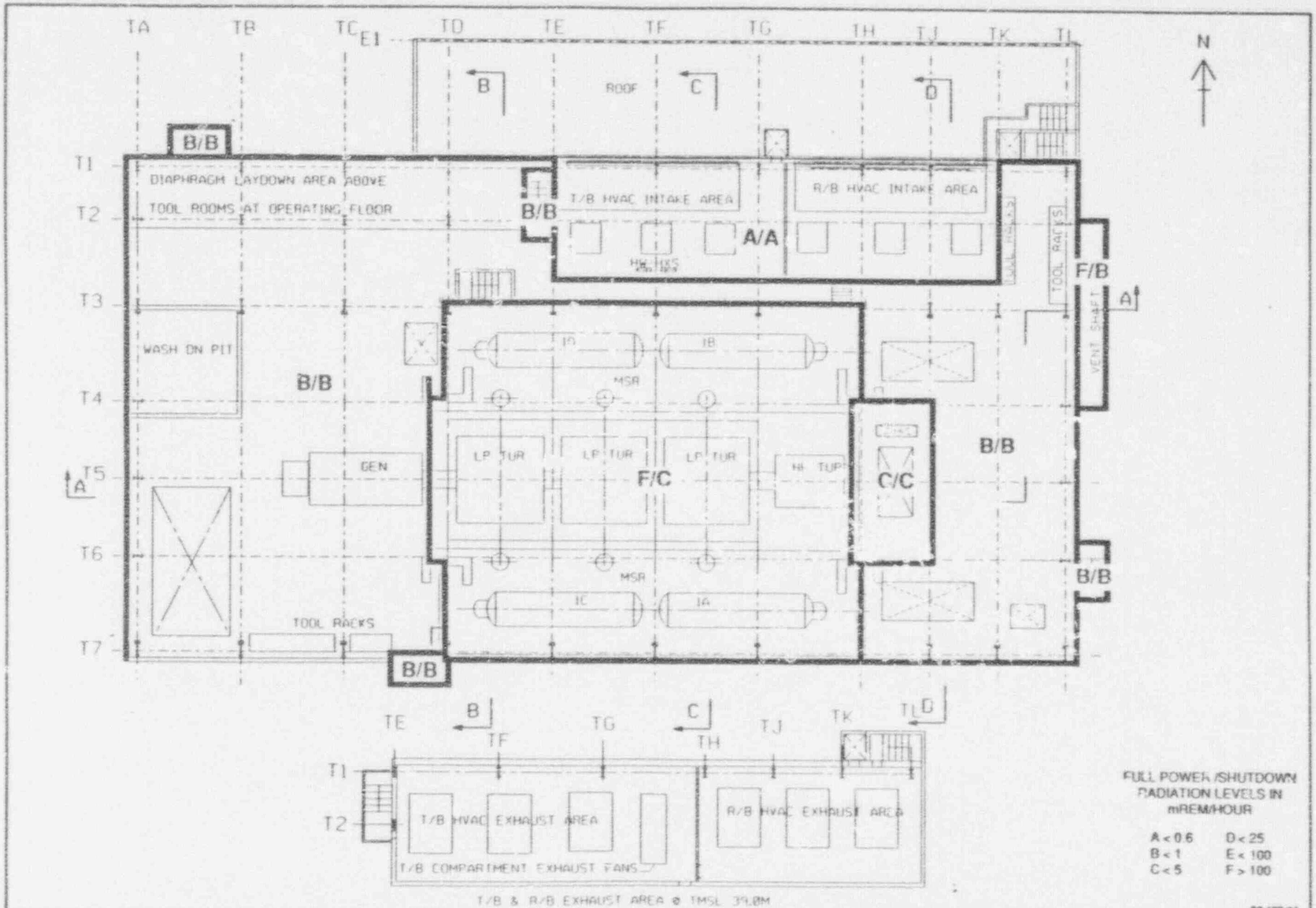

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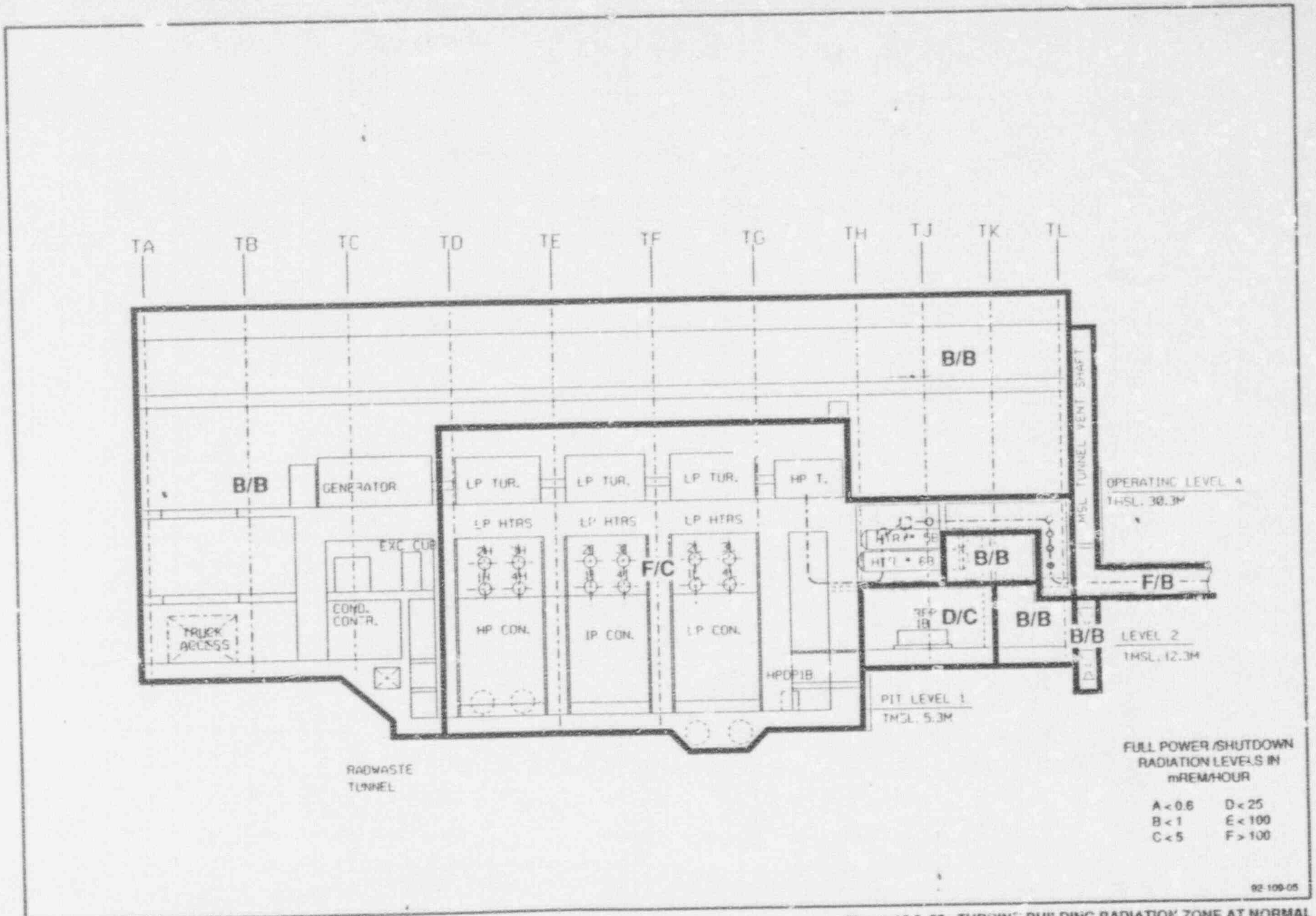
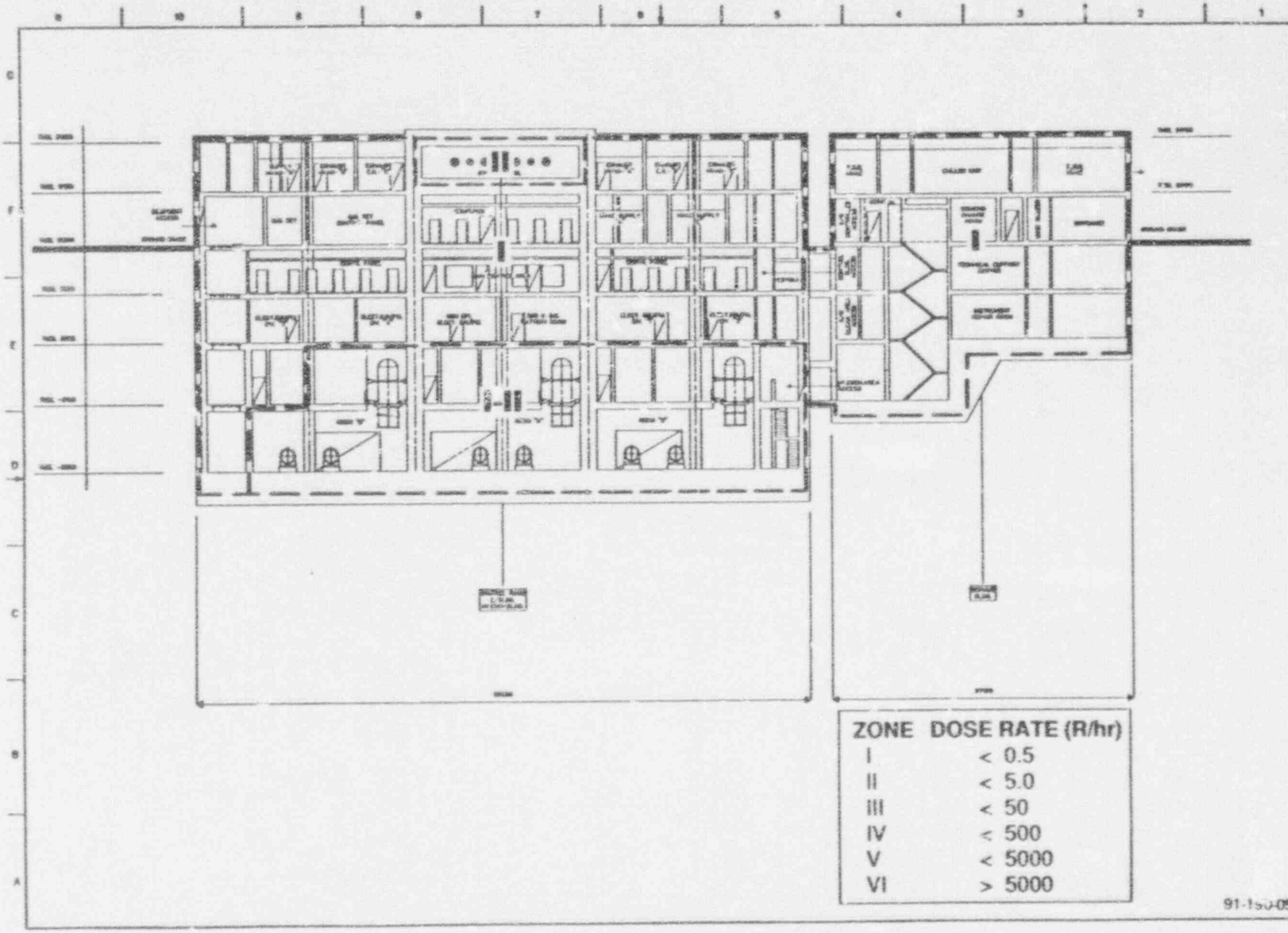


Figure 12.3-53 TURBINE BUILDING RADIATION ZONE AT NORMAL OPERATION LONGITUDINAL SECTION AA



ZONE	DOSE RATE (R/hr)
I	< 0.5
II	< 5.0
III	< 50
IV	< 500
V	< 5000
VI	> 5000

Figure 12.3–54 CONTROL BUILDING, RADIATION ZONE, POST-LOCA, SIDE VIEW

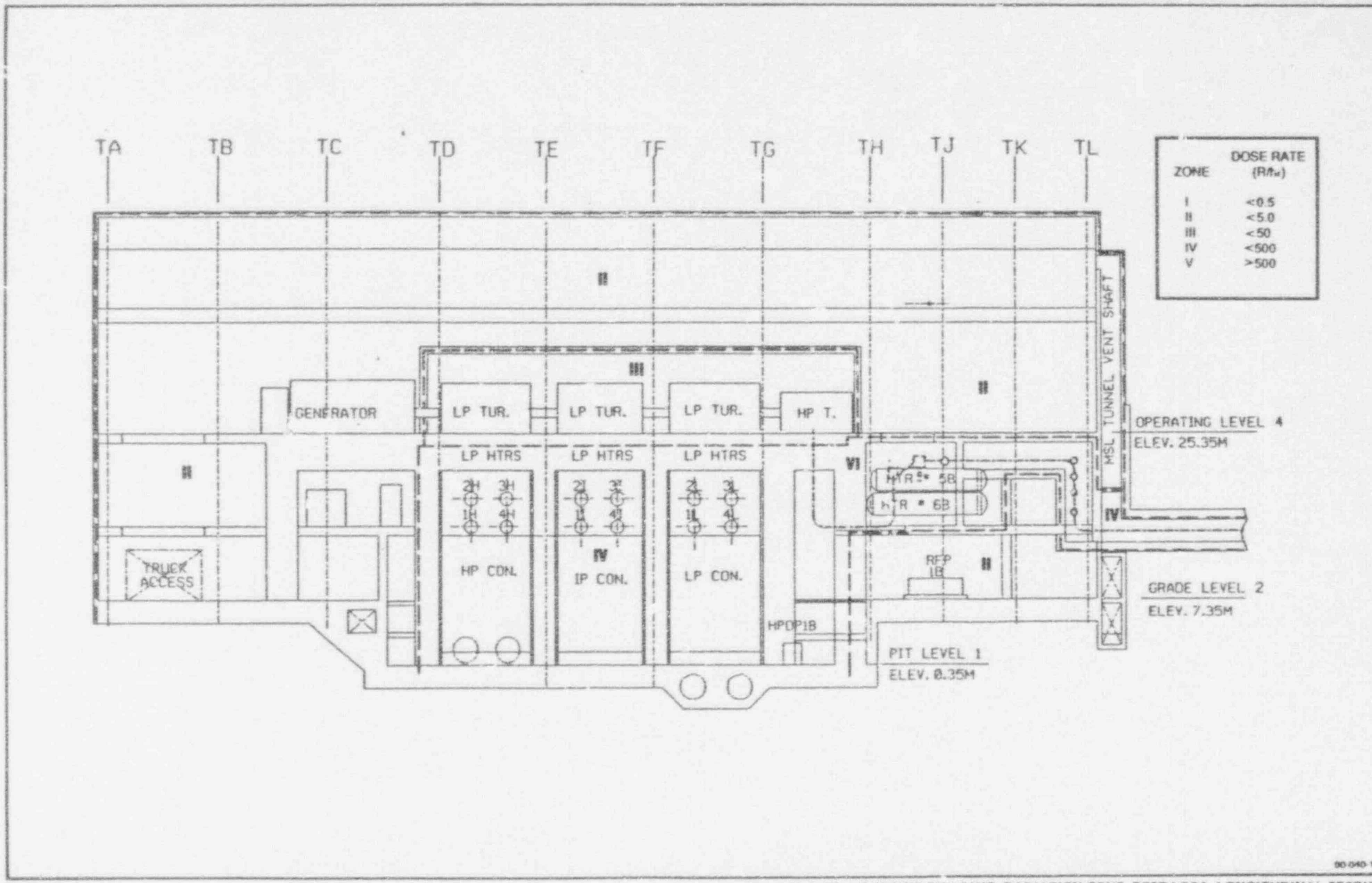


Figure 12.3–55 TURBINE BUILDING, RADIATION ZONE, POST-LOCA, LONGITUDINAL SECTION

Enclosure 2

PIPING DESIGN DAC MATERIAL

- Appendix A - Design Description and ITAAC
- Appendix B - Draft Safety Evaluation Report
- Appendix C - SSAR Material

3.3 Piping Design

Design Description

Piping associated with hydraulic and pneumatic systems is categorized as either nuclear safety related or non-safety related. Piping systems that must remain functional following a safe shutdown earthquake (SSE) are designated as Seismic Category I. Depending on the intended service conditions and system design functions, piping is further classified as ASME Code Class 1, 2, 3, or non-Code Class. NRC regulations govern piping designations and piping in the certified design may further be classified as Quality Group A, B, C, or D.

All ABWR piping components will be designed, fabricated, installed and examined to confirm full compliance with all applicable regulatory requirements and industrial codes and standards.

Inspection, Test, Analyses and Acceptance Criteria

Table 3.3 provides a definition of the inspections, tests and analyses, together with the acceptance criteria, which will be performed for ABWR piping in order to demonstrate compliance with the certified design commitments. The information in Table 3.3 is intended to be generic and to apply to all safety related piping governed by Quality Group A, B, or C and ASME Code Class 1, 2, or 3 designations. Not all of the entries in Table 3.3 apply to all piping classifications. Appropriate applicability, based on designation, will be incorporated at the time the inspections, tests, and analyses are implemented.

Table 3.3: Generic Piping Design

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The piping shall be designed for a fatigue life of 60 years. This design shall account for the cyclic stresses resulting from the expected pressure/temperature cycles and loads in the required combinations. For ASME Class 1 piping systems, a fatigue analysis will be performed in accordance with ASME Code, Section III requirements. For ASME Class 2 & 3 piping, ASME Code, Section III rules will be followed using a stress range reduction factor of 1.0, based on fewer than 7000 cycles. These fatigue analyses results shall be documented in a certified stress report.	1. An inspection of the certified stress report will be conducted to assure that the fatigue evaluation is consistent with the ASME Code, Section III requirements and with the 60 year design life.	1. ASME Code, Section III requirements shall be satisfied, including the cumulative fatigue usage factor, which shall be less than or equal to 1.0. The applied subsections of ASME Code shall be contained in the approved editions documented in 10CFR50.55a.
2. Pipe mounted equipment allowable loads and attachment interface (for example, the interface between a snubber and its embedment plate) allowable loads, accelerations and stresses shall be satisfied. The loads, accelerations, and stresses that the piping system imposes on its pipe mounted equipment and on its interfaces shall be determined by analyses of the piping systems and compared to the allowable values. The results of these analyses shall be documented as interface requirements to assure design compatibility with the equipment and interfaces.	2. Inspections of stress reports, design specifications, and design drawings will be conducted to confirm that the as-designed interface loads, accelerations and stresses are consistent with the interfacing vendor's / constructor's specified hardware allowables.	2. The allowables for pipe mounted equipment and interfacing equipment shall be met. The allowables at attachment interfaces shall be met.

Table 3.3: Generic Piping Design (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. Analytical methods for the dynamic and static analysis of piping systems and the corresponding component stress analysis shall be specified in a certified design specification for each piping system. The dynamic analysis of piping systems shall use a suitable dynamic method, such as time history or response spectrum method, or an equivalent static load method. Linear-elastic analysis or nonlinear-plastic analysis shall be used. For the applied method, the key analysis parameters shall be addressed. For example, for the response spectrum method, the following shall be defined: a. Combination of group responses when multiple response spectra are used. b. Combination of modal responses. c. Combination of response spectra analysis results with differential building movement analysis results. d. Damping coefficients. e. Cut-off frequency. f. High frequency modes.	3. Inspection (review) of the certified design specification and the certified stress report will be conducted to confirm that the piping was designed and analyzed in compliance with all regulatory (and other applicable) requirements.	3. Methods shall be in compliance with all applicable regulatory requirements.

Table 3.3: Generic Piping Design (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. Essential piping systems, including required pipe whip restraints, shall be designed to protect against the dynamic effects associated with the postulated rupture of high energy and moderate energy fluid systems. A pipe break analysis report shall be generated to confirm that the piping system is acceptable for all postulated breaks. Piping systems that are qualified for the optional leak-before-break design approach may exclude design against the dynamic effects from the postulation of breaks in high energy piping	4. Inspections of ASME Code III required documents and the pipe break analysis report, or leak-before-break justification report, will be conducted to confirm that the piping system was designed/analyzed in compliance with requirements that assure postulated pipe breaks will not unduly impact the safety of the plant.	4. The essential functions of structures, systems, and components shall not be precluded by the postulated pipe breaks. For those components required for safe shutdown, limits to meet the ASME Code requirements for faulted conditions and limits to ensure required operability shall be met.
5. All ASME Code Safety Class 1, 2, and 3 piping systems which are essential for safe shutdown, shall be designed to assure that they will maintain sufficient dimensional stability to perform their required function following application of all loads to which they will be subjected during postulated events requiring their safety function.	5. An inspection of the certified stress report will be conducted to assure that none of the stresses or deflections of the piping system exceed values which could lead to large reductions in the cross-sectional flow area.	5. ASME Code, Section III limits that protect the piping and pipe supports against primary stress failures will be compared with allowable values that preclude impairment of functional capability. In no case will stresses exceed values allowed for Service Level D in ASME Code, Section III.

Table 3.3: Generic Piping Design (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. When performing static and dynamic analysis of piping systems, the mathematical model of the piping system shall be constructed to realistically reflect the dynamic and static characteristics of the piping system. The following parameters shall be addressed: a. The model shall adequately account for modes up to the analysis cut-off frequency. b. The appropriate stiffness and mass of piping, pipe supports, and pipe mounted equipment shall be included in the piping system model. c. The appropriate stiffnesses for anchors and intermediate supports shall be included in the piping system model.	6. An inspection (verification) of the mathematical model will be performed to confirm that the boundary conditions and dynamic and static characteristics have been adequately technically addressed.	6. Analytical modeling practices shall be in compliance with all applicable regulatory requirements. The methods used for modeling will be applied to NRC benchmark problems and the results of the corresponding analyses shall be compared to the NRC benchmark and consistency shall be confirmed.
Construction Items:		
7. The piping, its appurtenances, and its supports, shall satisfy the ASME Class, Seismic Category, and Quality Group requirements commensurate with its classification.	7. Inspections will be conducted of ASME Code required documents and the Code stamp on the components.	7. Existence of ASME Code required documents and the Code stamps on the components confirms that the piping and components have been designed, analyzed, fabricated, and examined in accordance with the applicable requirements.
8. For those piping systems using ferritic materials, the ferritic materials shall not be susceptible to brittle fracture under pressure during the expected service conditions. Only intrinsically tough grades of ferritic materials conforming to the ASME Code, Section III SA specifications shall be used.	8. Fracture toughness tests will be performed in accordance with ASME Code, Section III.	8. Records of the fracture toughness tests must confirm that the requirements of ASME Code, Section III are satisfied.

Table 3.3: Generic Piping Design (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. For those piping systems using austenitic stainless steel materials, the stainless steel piping shall be selected to minimize the possibility of cracking during service. Special chemical, fabrication, handling, welding, and examination requirements that minimize cracking shall be met.	9. Inspections of ASME Code required documents and other pertinent records will be conducted to confirm that manufacture, fabrication, welding, and examination were performed in accordance with the committed requirements.	9. Records of the materials and processes must confirm that the committed requirements to avoid the potential of stainless steel to crack in service are satisfied
10. For essential systems, the as-built piping system shall be confirmed to be consistent with the as-designed piping system. All deviations shall be shown to not invalidate the design.	10. a. Pipe routing will be confirmed by inspecting isometric drawings containing verification stamps from field visual inspections. This documentation will also confirm that no interferences exist. b. The exact location, orientation, and size of snubbers and struts; the location and size of hangers; the location and weight of valves, pumps, and heat exchangers; the location and configuration of anchors; the location of guides and pipe whip restraints; and the specified clearances, will be confirmed by reviewing isometric drawings containing quality control verification stamps, or by taking the as-built measurements. c. Deviations from the as-designed condition will be documented and evaluated. If acceptance limits are not satisfied in the reevaluation, a reanalysis of the as-built condition will be performed, the stress report and design drawings will be revised, and the final stress report will be certified.	10. a. The as-built pipe routing is within the tolerances allowed on the as-designed drawings. The piping system has the minimum specified clearance from neighboring hardware. Deviations shall be addressed in compliance with c below. b. The location, size, orientation of pipe mounted components are within the tolerances allowed on the as-designed drawings. Deviations shall be addressed in clompliance with c below. c. For Safety Class 1, 2, & 3 piping, the required allowables in the applicable subsections of ASME Code, Section III shall be satisfied. The applied subsections of ASME Code, Section III shall be contained in the approved editions documented in 10CFR 50.55a.

Table 3.3: Generic Piping Design (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Combination Design and Construction Items:		
11. ASME Code Safety Class 1, 2, and 3 piping shall retain its pressure integrity under all internal pressures that will be expected during its design lifetime. Piping and piping components shall be designed and analyzed to show compliance with the pressure integrity requirements of ASME Code.	11. Inspections of ASME Code required documents will be conducted to confirm that the piping system was designed/ analyzed in compliance with requirements that assure pressure integrity. A hydrostatic test of the Safety Class 1, 2, and 3 piping will be conducted as required by, and in accordance with, the ASME Code.	11. For safety class 1, 2, & 3 piping, the required allowables in the applicable subsections of ASME Code, Section III shall be satisfied. The applied subsections of ASME Code, Section III shall be contained in the approved editions documented in 10CFR 50.55a. The results of the hydrostatic test must conform with the requirements in the ASME Code.
12. Piping shall be designed (and installed) to provide adequate clearance to prevent interference with other piping, structures, and components as the piping moves or deflects due to the thermal, dynamic, and/ or static loads which it experiences in service. Stress analyses shall be performed to calculate piping movements. These calculated movements shall be used to develop and document minimum required clearances.	12. An inspection of the certified stress report will be conducted to assure that the calculated pipe deflection values do not result in the piping exceeding its design allowables for the specified load combinations and that the minimum specified clearances adequately encompass these deflections. A field walkdown will be performed on all essential piping to measure the "As-installed" piping clearances and confirm the actual clearances are within allowable values.	12. The design allowables for piping clearance in both the axial and lateral directions shall be met.

SAFETY EVALUATION
OF THE ABWR PIPING DESIGN
AND RELATED ITAAC
(DRAFT)

May 14, 1992

Division of Engineering Technology
Office of Nuclear Reactor Regulation

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SAFETY EVALUATION OF THE ABWR PIPING DESIGN AND RELATED ITAAC

1 INTRODUCTION

In 10 CFR 52.47(a)(2), it is required that an application for design certification contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. It is also required in 10 CFR 52.47(a)(1) that an application for design certification contain proposed inspections, tests, and analyses, and acceptance criteria (ITAAC) which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria are met, a plant which references the design is built and will operate in accordance with the design certification. In SECY-92-053 dated February 19, 1992, the staff proposed to the Commission a method for using design acceptance criteria (DAC) together with detailed design information during the 10 CFR Part 52 process for reviewing and approving designs. The approach for using DAC was proposed in the design certification review of the ABWR, in part, because of difficulties experienced by the ABWR vendor, General Electric (GE), in obtaining as-built or as-procured information to finalize its piping design and analyses. GE believed that the use of preliminary information to establish an initial design would not have been cost-effective because of the many design changes that would likely occur when the as-built and as-procured information would be available.

The DAC are a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies in making a final safety determination to support a design certification. The DAC are objective (measurable, testable, or subject to analysis using pre-approved methods), and must be verified as a part of the ITAAC performed to demonstrate that the as-built facility conforms to the certified design. The use of DAC regarding how the acceptance criteria for design will be demonstrated by the COL holder during construction. The concept of DAC would enable the staff to make a final safety determination, subject only to satisfactory design implementation and verification by the COL holder through related ITAAC.

This report provides the staff's safety evaluation of the proposed DAC approach for the ABWR piping design. Consistent with the above position, the staff reviewed the details of the ABWR piping design approach, and our evaluation of the analysis methods, design procedures, and acceptance criteria to be used by the COL holder to complete the ABWR piping design are discussed herein.

The staff's review of the ABWR piping design was performed using the Standard Review Plan guidelines to evaluate the information in the ABWR SSAR and included a detailed audit of the piping design criteria and sample calculations using the ABWR design criteria. The review evaluated the adequacy of the structural integrity and functional capability of safety-related piping systems in the ABWR standardized plant design. The review was not limited to ASME Code Class 1, 2, and 3 piping and supports, but also included buried piping, instrumentation lines, the interaction of non-Category I piping with Category I piping, and any safety-related piping designed to industry standards other than the ASME Boiler and Pressure Vessel Code. The staff's evaluation of the adequacy of the ABWR piping design methods, acceptance criteria, and related ITAAC that are to be used for the completion and verification of the ABWR piping design is provided in the following sections of this safety evaluation. The staff's evaluation includes:

(1) applicable codes and standards
(2) analysis methods to be used for completing the piping design
(3) modeling techniques
(4) pipe stress analyses criteria
(5) pipe support design criteria
(6) high energy line break criteria
(7) leak-before-break approach applicable to the ABWR
(8) generic piping design ITAAC

The staff must arrive at a final safety determination that, upon successful completion by the COL holder of (1) the piping design and analyses and (2) the inspections, tests, analyses, and their acceptance criteria (ITAAC) as required in 10 CFR Part 52 using the design methods and acceptance criteria discussed herein, there is adequate assurance of the piping systems performing their safety-related functions under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events.

2 CODES AND STANDARDS

2.1 ASME Boiler and Pressure Vessel Code

For the ABWR design certification, GE has established that the ASME Boiler and Pressure Vessel Code, Section III is to be used for the design of ASME Code Class 1, 2, and 3 piping systems. The specific edition and addenda has not been specified for the piping during the ABWR design certification review, in part, because of the evolving technical content of the Code which might result in inconsistencies between design and construction practices. The ASME Code involves a consensus process to reflect the evolving design and construction practices of the industry. Although the reference to a specific edition of the Code for the design of piping systems is entirely feasible during the design certification stage, the construction practices and examination methods of an updated Code that would be effective at the COL stage might not be consistent with the earlier design practices used at the design certification stage.

In order to avoid this potential inconsistency for the ABWR piping systems, the staff finds that the specification of the ASME Boiler and Pressure Vessel Code without a commitment to a specific edition and appropriate addenda is sufficient because the regulations in 10 CFR 50.55a provide the means for the staff to revise or supplement specific portions of the updated Code editions and addenda to reflect their application to the certified designs. In this manner, the specific edition and addenda to be used at the time of the COL application is ensured to be consistent with the latest design, construction, and examination practices at that time. However, the staff finds that there is a need to adopt certain information from a specific Code edition or addenda during its design certification review particularly when that information is of importance to verify some aspect of the design or is used by the staff to reach its final safety determination. Such considerations are reflected in the various sections of this safety evaluation.

Therefore, all ASME Code Class 1, 2 and 3 piping and piping supports shall be designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III as endorsed in 10 CFR 50.55a in effect at the time of plant construction.

2.2 ASME Code Cases

The only acceptable ASME Code Cases that may be used for the design of ASME Code Class 1, 2, and 3 piping systems in the ABWR standard plant are those either conditionally or unconditionally approved in RG's 1.84 and 1.85 in effect at the time of plant construction. However, as noted above the staff has reviewed the acceptability of several proposed Code Cases that are currently endorsed in RG 1.84 and 1.85 in order to reach a final safety determination on the ABWR certified design.

In RG 1.84, the staff has conditionally endorsed ASME Code Case N-397, "Alternative Rules to the Spectral Broadening Procedure," for use on a case-by-case basis only. For the ABWR at this time, Code Case N-397 has not been requested for use and is not applicable.

In RG 1.84, the staff has conditionally endorsed ASME Code Case N-411, "Alternative Damping Values for Response Spectra Analysis of Classes 1, 2, and 3 Piping, Section III, Division 1." This Code Case is acceptable for the ABWR. The acceptability of the Code Case and its application is further discussed in Section 5.4 of this safety evaluation.

Other ASME Code Cases requested by GE that are applicable to the ABWR piping and support design are listed below:

ASME Code Case N-71-15, "Additional Materials for Subsection NF, Classes 1, 2, 3 and MC Component Supports Fabricated be Welding, Section III, Division I." This Code Case has been endorsed by the staff in RG 1.85.

ASME Code Case N-122, "Stress Indices for Structure Attachments, Class 1, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.

ASME Code Case N-247, "Certified Design Report Summary for Component Standard Supports, Section III, Division 1, Class 1, 2, 3 and MC." This Code Case has been endorsed by the staff in RG 1.84.

ASME Code Case N-249-9, "Additional Material for Subsection NF, Classes 1, 2, 3 and MC Component Supports Fabricated Without Welding, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.85.

ASME Code Case N-309-1, "Identification of Materials for Component Supports, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.

ASME Code Case N-313, "Alternate Rules for Half-Coupling Branch Connections, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.

ASME Code Case N-316, "Alternate Rules for Fillet Weld Dimensions for Socket Welded Fittings, Section III, Division 1, Class 1, 2, 3." This Code Case has been endorsed by the staff in RG 1.84.

ASME Code Case N-318-3, "Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping, Section III, Division 1." This Code Case has been conditionally endorsed by the staff in RG 1.84 and is discussed further in Section 5.15 of this safety evaluation.

ASME Code Case N-319, "Alternate Procedure for Evaluation of Stress in Butt Weld Elbows in Class 1 Piping, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.

ASME Code Case N-391, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.

ASME Code Case N-392, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Classes 2 and 3 Piping, Section

III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.

ASME Code Case N-393, "Repair Welding Structural Steel Rolled Shaped and Plates for Component Supports, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.

ASME Code Case N-414, "Tack Welds for Class 1, 2, 3 and MC Components and Piping Supports." This Code Case has been endorsed by the staff in RG 1.84.

ASME Code Case N-430, "Requirements for Welding Workmanship and Visual Acceptance Criteria for Class 1, 2, 3 and MC Linear-Type and Standard Supports." This Code Case has been endorsed by the staff in RG 1.84.

ASME Code Case N-416, "Alternative Rules for Hydrostatic Testing of Repair or Replacement of Class 2 Piping." This Code Case has been endorsed by the staff in RG 1.147.

ASME Code Case N-463, "Evaluation Procedures and Acceptance Criteria for Flaws in Class 1 Ferritic Piping That Exceed the Acceptance Standards of IWB-3514-2." This Code Case has been endorsed by the staff in RG 1.147.

2.3 Design Specification

The ASME Boiler and Pressure Vessel Code, Section III requires that a design specification be prepared for Class 1, 2, and 3 components such as pumps, valves, and piping systems. The design specification is intended to become a principal document governing the design and construction of these components and should specify loading combinations, design data, and other design data inputs. The Code also requires a design report for ASME Code, Class 1, 2, and 3 piping and components. In the ABWR SSAR, GE committed to construct all safety-related components, such as vessels, pumps, valves and piping systems, to applicable requirements of the ASME Code, Section III. During its review of the ABWR SSAR, the staff reviewed selected documents related to design specifications and design reports. Those documents were not specifically for the ABWR, but were provided by GE and reviewed by the staff as a demonstration of how design specifications and design reports will be prepared for ABWR plants. The staff determined that the demonstration documents, with modifications, would meet code requirements. However, because the documents were not specifically for the ABWR, they will have to be modified before the staff can conclude that the design specification and design report requirements in ASME Code, Section III, Subsection NCA have been met. In order for the staff to reach this conclusion, it will perform plant-specific design documentation audits for plants referencing the ABWR design. In Section 3.9.7, "Interfaces," GE made a commitment that utility applicants referencing the ABWR design will make available to the staff design specifications and design reports required by the ASME Code for vessels, pumps, valves, and piping systems for the purpose of audit. The staff finds this commitment to be an acceptable COL action item.

2.4 Conclusions

On the basis of its review of Section 3.9.3.1 of the SSAR, the staff finds that GE met 10 CFR 50.55a and GDC 1 with respect to the codes and standards specified for ASME Code, Class 1, 2, and 3 components by ensuring that systems and components important to safety are designed to quality standards commensurate with their importance to safety.

3 ANALYSIS METHODS

The staff reviewed the information in Section 3.9.1 of the SSAR relative to the design transients and methods of analysis used for all seismic Category I piping and pipe supports designated as ASME Code Class 1, 2, and 3 under ASME Code, Section III, and those not covered by the code. It reviewed the assumptions and procedures used for the inclusion of transients in the design and fatigue evaluation of ASME Code Class 1 and CS components. It also reviewed the computer programs used in the design and analysis of seismic Category I components and their supports, as well as experimental and inelastic analytical techniques.

3.1 Experimental Stress Analysis

In Section 3.9.1.3 of the SSAR, GE identified several components for which experimental stress analysis is performed in conjunction with analytical evaluation. In the piping area these components include the piping seismic snubbers and pipe whip restraints. The staff's evaluation of the analysis methods used to qualify these components are discussed in further in Sections 6 and 7 of this safety evaluation. The staff's position is that experimental stress analysis methods shall be in compliance with Appendix II of the ASME Code, Section III.

3.2 Modal Response Spectrum Method

GE performed the system and subsystem analyses on an elastic basis. Modal response spectrum and time history methods form the basis for the analyses of all major seismic Category I piping systems and components. In a response spectrum method of analysis, the mode shapes and the natural frequencies are calculated first. The dynamic response of the system is then calculated for each mode using the floor response spectrum as the input to the analysis. For a piping system that is supported at points with different dynamic excitations, the response analysis is usually performed using an enveloped response spectrum.

Three components of earthquake motion shall be considered in the analysis. The maximum response due to each of the three components should be combined by the square-root-of-the-sum-of-squares (SRSS) of the maximum codirectional responses caused by each of the three components of earthquake motion. For other dynamic events, the SRSS method can be used provided it can be demonstrated that there is no phase relationship between the three perpendicular excitation direction.

3.3 Independent Support Motion Method

As an alternative to the enveloped response spectrum method, the multiple-support excitation analysis method may be used. When this method is used, the staff's position is that the responses due to motions of supports between two or more different support groups may be combined by the SRSS method if a support group is defined by supports that have the same time history input. This usually means all supports located on the same floor, or positions of a floor, of a structure. In Amendment 11 to the SSAR, GE committed to use this definition for the design of ABWR piping systems. Therefore, the staff finds this alternative to the enveloped response spectrum method to be acceptable.

This method should be implemented in accordance with the information and recommendations in Sections 2.3 and 2.4 of NUREG-1061, "Report of the U.S.N.R.C. Piping Review Committee," Volume 4.

3.4 Time-History Method

A time history analysis is performed using either the direct integration or modal superposition method. Based on the GE documents reviewed by the staff during its audit at the offices of General Electric on March 23-26, 1992, only the direct integration method is addressed. This method of analysis is primarily used by GE for systems subjected to short duration and high frequency excitation such as those systems in the suppression pool subjected to the direct SRV and LOCA loads. The loadings may be applied either as an external load onto the pipe or as an internal fluid hydraulic transient load.

An appropriate integration time step, Δt, shall be selected to ensure stable integration. This is generally achieved when smaller time steps introduce no more than a 10% error in the total dynamic response. In addition, consideration should be given in the analysis of expected variations of piping properties, damping, and loadings — comparable to the peak broadening in developing seismic floor response spectra.

The method for combining the three-dimensional effects may utilize the approach described in Section 3.2 of this safety evaluation or may be combined algebraically at each time step.

3.5 Inelastic Analysis Method

GE has not provided any information on the use of inelastic analysis methods for the ABWR piping analyses. If inelastic methods are to be used in any ABWR piping analyses, then the staff requires that the details of the inelastic method and its acceptance criteria as well as the scope and extent of its application shall be submitted to the staff for approval prior to its use.

3.6 Small-Bore Piping Method

At this time, GE has not provided the staff any specific information with respect to the method to be used for the structural design of small-bore piping systems and instrumentation lines in the ABWR standard plant. This information is required to be included in the SSAR in order for the staff to reach a final safety determination on the adequacy of the ABWR small-bore piping design. This is considered an open item.

With respect to the use of EPRI-6628, "Procedure for Seismic Evaluation and Design of Small Bore Piping," (NCIG-14), the staff finds that the approach incorporates, in part, the use of a seismic experience-based approach for the qualification of safety-related piping. The staff has not accepted this experienced-based approach for the design or qualification of safety-related piping in nuclear power plants at this time. Currently, the staff accepts a suitable dynamic analysis or a suitable qualification test except where the use of an equivalent static analysis has been demonstrated to be adequate for the design of piping systems. The staff position is that the NCIG-14 approach is not acceptable for the design of safety-related, small-bore piping in the ABWR standard plant.

3.7 Non-Seismic/Seismic Interaction (II/I)

All non-seismic Category I piping (or other systems and components) should be isolated from Category I piping. This may be achieved by designing a seismic constraint or barrier or by locating the two sufficiently apart to preclude any interaction. If it is impractical to isolate the Category I piping system, the adjacent non-seismic Category I system should be evaluated to the same criteria as the Category I system. The use of other methods should be submitted to the staff for review and approval prior to its use.

For non-seismic Category I piping systems attached to seismic Category I piping systems, the dynamic effects of the non-seismic Category I system shall be considered in the analysis of the Category I piping. In addition, the non-seismic Category I piping from the attachment point to the first anchor shall be evaluated to ensure that under all loading conditions, it will not cause a failure of the seismic Category I piping system.

3.8 Main Steam Line and By-Pass Line in the Turbine Building

For the ABWR plant design, GE proposed to eliminate the main steam isolation valve leakage control system. Instead, it proposes to rely on the use of an alternate leakage path which takes advantage of the large volume and surface area in the main steam piping, by-pass line, and condenser to hold up and plate out the release of fission products following core damage. In this manner, the main steam piping, by-pass line, and condenser are used to mitigate the consequences of an accident and are required to remain functional during and after a safe shutdown earthquake.

For this reason, the staff position is that the main steam piping beyond the second outermost isolation valve up to the seismic interface restraint and connecting branch lines up to the first normally-closed valve be classified as Quality Group B (Safety Class 2) and Seismic Category I. The main steamline from the seismic interface restraint up to but not including the turbine stop valve (including branch lines to the first normally-closed valve) shall be classified as Quality Group B and inspected in accordance with the applicable portions of ASME Section XI, but may be classified as non-seismic Category I if they have been analyzed using a dynamic seismic analysis method to demonstrate their structural integrity under SSE loading conditions. However, all pertinent quality assurance requirements of 10 CFR Part 50, Appendix B are applicable to ensure that the quality of the piping material is commensurate with its importance to safety during normal operational, transient, and accident conditions. For assuring the integrity of the main steam by-pass line from the first valve to the main condenser hot-well, the staff position is that (1) the main steam by-pass line from the first valve up to the condenser inlet and (2) the main steam piping between the turbine stop valve and the turbine inlet is not required to be classified as safety-related nor as Seismic Category I, but should be analyzed using a dynamic seismic analysis to demonstrate their structural integrity under SSE loading conditions.

Lastly, the main steam piping and by-pass line in the turbine building shall be protected from the collapse of any non-seismic Category I structure in the event of an SSE. As a final confirmatory measure, the staff requires that a plant-specific walkdown be performed prior to operation to assess the potential failures of non-seismically designed systems, structures, and components overhead, adjacent to, and attached to the alternate leakage path (i.e., the main steam piping, by-pass line, and the main condenser). This walkdown should be performed as a part of the ITAAC verification of non-seismic/seismic interaction.

GE has proposed a revision to its SSAR to reflect the above staff position. Contingent upon the SSAR revision incorporating the above staff position, the staff finds that the methods of analyses described above to be used to assure the structural integrity of the alternate leakage path provides an acceptable methodology to ensure the structural integrity of the main steam piping and by-pass line in the turbine building during and following an SSE.

3.9 Buried Piping

Section 3.7.3.12 of the SSAR outline criteria that will be used in the analysis of buried seismic Category I piping systems. These criteria conform to the applicable guidelines in SRP Section 3.9.2. However, GE has not provided any detailed information on how the criteria are to be applied in the design of buried piping. Specifically, GE should address, as a minimum, (1) the maximum bearing loads, (2) the categorization of seismic stresses in the Code evaluation, and (3) the allowable stress limits for the piping. In order for the staff to complete its review of the ABWR buried piping, the staff requires that this information be included in the SSAR.

3.10 ASME Section III, Appendix N

The staff has not endorsed the use of ASME Code Section III, Appendix N, "Dynamic Analysis Methods." This is a non-mandatory appendix which is still evolving and does not currently agree with some regulatory positions. Therefore, for the ABWR piping design, when the methodology in Appendix N is not consistent with regulatory positions discussed herein, then the regulatory positions shall be used.

3.11 Conclusions

Contingent upon GE providing an acceptable revision to its SSAR that addresses the lack of information identified above and reflects the staff's positions as indicated, and on the basis of its review of the SSAR Section 3.9.2 and its audit of the specific design procedures for the ABWR piping systems, the staff concludes that the analysis methods to be used for all Seismic Category I piping systems as well as non-seismic Category I piping systems that are important to safety utilize a suitable dynamic analysis method or an equivalent static analysis method. The analysis methods utilize piping design practices that are commonly used in the industry and provide an adequate margin of safety to withstand the loadings due to normal operating, transient, and accident conditions.

4 PIPING MODELING

4.1 Computer Codes

This section addresses the computer codes to be used to analyze piping systems in the ABWR design. All computer programs used by GE for static and dynamic analyses to determine the structural and functional integrity of seismic Category I Code and non-Seismic Category I code items are included in Appendix 3D to the SSAR. Design control measures to verify the adequacy of the design of safety-related components are required by 10 CFR Part 50, Appendix B. In Section 3.9.1.2 of the SSAR, GE stated that the quality of the programs and the computer results are controlled either by GE or by outside computer program developers. In addition, the programs are verified by one or more of the methods recommended in SRP Section 3.9.1.

The staff is currently performing an independent confirmatory piping stress analysis of representative piping systems in the ABWR standard plant. The purpose of these analyses is to verify the adequacy of the GE computer program used to generate the sample piping analyses that were audited by the staff on March 23-26, 1992 at GE's offices in San Jose, California. The results of the confirmatory analysis will be discussed in a supplement to this safety evaluation.

4.2 Dynamic Piping Model

For the dynamic analysis of seismic Category I piping, each system is idealized as a mathematical model consisting of lumped masses connected by elastic members. The stiffness matrix for the piping system is determined using the elastic properties of the pipe. This includes the effects of torsional, bending, shear, and axial deformations as well as change in stiffness due to curved members.

The staff reviewed the method for selecting the number of masses or degrees of freedom in the mathematical piping model to determine its dynamic response. Based on the staff's March 23-26, 1992 audit of GE's internal documents, pipe and fluid masses are lumped at nodes which are selected to coincide with the locations of large masses (e.g., valves, pumps, and tanks) and with locations of significant geometric changes (e.g., pipe elbows, reducers, and tees). Additional mass points are selected to ensure that the spacing between any two adjacent piping nodes and masses is no greater than an idealized value. This value corresponds to the length of a simply supported beam with a uniformally distributed mass whose undamped natural frequency is equal to the cut-off frequency. Since this approach in effect would capture all modes up to the cut-off frequency, the staff finds that the ABWR method for locating mass points is acceptable. The SSAR should be revised to reflect the above described approach.

The effect of pipe supports on the piping response shall be considered in the analytical model by including its actual stiffness properties. If default or generic stiffnesses are used in the piping model, then justification shall be developed to validate the stiffness values used in the piping model. The justification shall include verification that the generic values are representative of the types of pipe supports used in the piping system. This alternative approach to use generic stiffness values and its bases should be submitted to the staff for review and approval prior to its use. Additionally, because the amplified response spectra are generally specified at discrete building node points, any additional flexibility between these points and the pipe support (e.g., supplementary steel) shall also be addressed. The SSAR should be revised to incorporate the above information.

When piping terminates at non-rigid equipment (e.g., tanks, pumps, or heat exchangers) then the analytical piping model shall consider the flexibility and mass effects from these equipment. The SSAR should be revised to address how the flexibility and masses of equipment attached to the piping are to be modelled.

4.3 Piping Benchmark Program

The NRC staff is currently reviewing the adequacy of the GE computer program used in the representative ABWR piping analyses that were audited by the staff on March 23-26, 1992 at GE's offices in San Jose, California. The staff is performing an independent confirmatory piping analysis and will compare the results of its analysis with those provided by GE. Contingent upon an acceptable resolution of this confirmatory analysis, the staff concludes that the computer program used by GE for the ABWR piping analysis is adequate.

To verify the adequacy of the computer program used by the COL holder to complete the ABWR piping system design and analyses, mathematical models of representative piping systems in the ABWR standardized plant will be established by the NRC staff to be used in a benchmark program. The mathematical models are based on the dynamic piping model described in Section 4.2 and on the piping design criteria in Section 5 of this safety evaluation. The benchmark program verifies the adequacy of linear-elastic, dynamic piping analysis methods using the enveloped response spectrum method, multiple response spectrum method, and time-history method of analyses.

The benchmark program essentially consists of constructing mathematical models of the ABWR feedwater piping system inside containment and a safety-relief valve (SRV) discharge line inside the suppression pool wetwell area using the COL holder's computer program. The piping configuration for the piping models are described in NUREG-XXXX (currently under preparation by the NRC staff) and include (1) piping dimensions, (2) pipe sizes, (3) materials, (4) valve weights, (5) support and anchor stiffnesses, and (6) support locations. The piping input parameters for the benchmark analyses are also specified in NUREG-XXXX and include (1) damping values, (2) loading definitions, and (3) load combinations.

When the COL holder's dynamic piping analyses are completed, the results of the analyses shall be compared with the results of the benchmark problems provided in NUREG-XXXX. The piping results to be compared and evaluated include the system modal frequencies, the maximum pipe moments, the maximum support loads and equipment reactions, and the maximum pipe deflections. The acceptance criteria or range of acceptable values are specified in NUREG-XXXX and shall be satisfied. Any deviations from these values as well as the justification for their deviations shall be documented and submitted to the NRC staff for review and approval prior to initiating final certified piping analyses.

The benchmark program provides assurance that the computer program used to complete the ABWR piping design and analyses produces results that are consistent with results considered acceptable to the NRC staff.

4.4 Decoupling Criteria

When analyzing piping systems, the size of the mathematical model might exceed the capacity of the computer program when large and small bore piping are included. Thus, the small bore branch lines are generally decoupled from the large bore main piping. Currently, the SSAR does not provide any criteria for the decoupling of the piping systems in the analysis model. However, in a letter from P. Marriott (GE) to USNRC dated February 24, 1992, GE has provided a decoupling criteria in a GE document entitled, "ABWR SSAR Main Steam, Feedwater and SRVDL Piping Systems Design Criteria and Analysis Methods," (draft), Revision 0, dated February 1992. This document stated that when the ratio between pipe diameters of the branch line to main line is less than one-third, the branch line can be excluded from the piping model of the main line.

For GE to utilize this criteria for all piping systems in the ABWR plant, the basis for the one-third ratio needs to be reviewed by the staff. GE also needs to define how the mass effect of the decoupled line is accounted for in the model of the main line and how the frequency ratio effect (or resonant amplification of the main line) is accounted for in the modeling and analysis of the branch line. GE should revise its SSAR to include this information.

4.5 Conclusions

Contingent upon GE providing an acceptable revision to its SSAR that reflects the staff's positions as indicated above, the staff concludes that GE met 10 CFR Part 50, Appendix B, and GDC 1 by submitting information that demonstrates the applicability and validity of the design methods and computer programs used for the design and analysis of seismic Category I piping designated as ASME Code Class 1, 2, and 3 and those not covered by the code within the present state-of-the-art limits and by having design control measures that are acceptable for ensuring the quality of its computer programs. Although utility applicants referencing the ABWR design are not required to use the GE computer programs, the staff will require that computer programs used by the COL holder to complete its analyses of the ABWR piping systems be validated using the piping benchmark program discussed herein.

5 PIPE STRESS ANALYSIS CRITERIA

5.1 Seismic Input (Envelope Vs. Site-Specific Soil Properties)

The ABWR standardized plant is designed for a safe shutdown earthquake (SSE ground motion defined by a RG 1.60 response spectrum anchored to a peak ground acceleration of 0.3g. Amplified building response spectra are generated for the ABWR standard plant to account for varying soil properties in the U.S. by enveloping 14 site conditions. GE has proposed that the COL holder use these enveloping amplified building response spectra provided in the SSAR to complete the design and analyses of the ABWR piping systems.

The staff recognizes that the enveloping amplified building response spectra for the ABWR plant contain conservatism that might be excessive for certain specific site conditions. Accordingly, the staff's position is that when the SSE response spectrum is defined by a RG 1.60 response spectrum anchored to a peak ground acceleration of 0.3g, the type of soil properties applicable to the site may be considered in generating the amplified building response spectra. The method used to generate the amplified building response spectra shall be consistent with the method described in the SSAR as approved by the staff.

The staff's evaluation of the method used by GE for generating the amplified building response spectra will be provided in the staff's final safety evaluation of the ABWR standard plant design.

5.2 Design Transients

In Table 3.9-1 of the SSAR, GE lists the design transients for five plant operating conditions and the number of either plant operating events or cycles for each of the design transients that will be used in the design and fatigue analyses of the ASME Code Class 1 piping systems. For a design life of 60 years, the number of cycles for each transient shall be increased by a factor of 1.5. The SSAR should be revised to reflect this factor. The operating conditions included the following:

(1) ASME Service Level A - normal conditions

(2) ASME Service Level B - upset conditions - incidents of moderate frequency

(3) ASME Service Level C - emergency conditions - infrequent incidents

(4) ASME Service Level D - faulted conditions - low-probability postulated events

(5) testing conditions

The number of events or cycles resulting from each of the listed design transients that are applicable to other ASME Code Class piping systems is to be documented by the COL holder in its design specification and/or stress report for each component.

5.3 Loadings and Load Combinations

The staff reviewed the methodology used for load combinations and the selected values of allowable stress limits. GE provided the design criteria for all ASME Code, Class 1, 2, and 3 piping and piping supports using the load combinations and stress limits given in Section 3.9.3.1 of the SSAR. The method used in the combination of dynamic responses of piping loadings shall be in accordance with NUREG-0484, "Methodology for Combining Dynamic Responses," Revision 1, dated May 1980.

The staff reviewed this information and concludes that appropriate combinations of normal, operating transients, and accident loadings is specified to provide a conservative design envelope for the design of piping systems. The load combinations are consistent with the guidelines provided in SRP Section 3.9.3 and is thus acceptable.

5.4 Damping Values

RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," contains recommended values of damping to be used in the seismic analysis of structures, systems, and components. In addition, RG 1.84, "Design and Fabrication Code Case Acceptability, ASME Section III, Division 1," Revision 25, May 1988, conditionally endorses ASME Code Case N-411-1, "Alternative Damping Values for Response Spectra Analysis of Classes 1, 2, and 3 Piping, Section III, Division 1." The damping values used by GE are the same as those specified in either RG 1.61 or those specified in ASME Code Case N-411-1 as permitted by RG 1.84, and, therefore, the staff finds these criteria to be acceptable.

GE proposed to use the damping values specified in ASME Code Case N-411 with the independent support motion (ISM) method of response spectrum analysis. The staff's position on the application of N-411 damping values to the ISM method of analysis is that it is acceptable when the ISM method is used in accordance with the information and recommendations in Sections 2.3 and 2.4 of NUREG-1061, Volume 4.

The staff's position on the use of N-411 damping values with ASME Code Case N-420, "Linear Energy Absorbing Supports for Subsection NF, Classes 1, 2, and 3 Construction, Section III, Division 1," is that the two Code Cases may only be used in separate analyses as a further condition of RG 1.84 because the damping values established in Code Case N-411 might not be entirely appropriate for the damping characteristics of the linear energy absorbing supports. Therefore, the two Code Cases are not to be used in the same analysis.

5.5 Combination of Modal Responses

For the response spectrum method of analysis, the modal responses are combined by the square-root-of-the-sum-of-the-squares (SRSS) method. Closely spaced modes are combined using the criteria of RG 1.92. GE considers all modes with frequencies below 33 Hz in computing equipment and component response for seismic loadings. The staff finds that this method is consistent with the applicable guidelines of SRP Section 3.9.2 and is thus acceptable.

5.6 High Frequency Modes

For seismic analysis, consideration of high-frequency modes to preclude missing mass effects shall be included. The staff's guidelines for this is provided in SRP Section 3.7.2, Appendix A. The SSAR should be revised to reflect the above staff position or if an alternative method is used, then the details of its basis shall be submitted to the staff for review and approval prior to its use.

For the analyses of vibratory loads (other than seismic) with significant high frequency input, i.e., 33 to 100 Hz, the staff's positions are as follows:

a. The methodology for the combination of high frequency modal results has not been addressed by GE at this time for the ABWR piping design. The staff's position is that the high frequency modes shall be combined using the guidelines provided in RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis." Use of other combination methods such as the algebraic modal combination method for combining high frequency modes will require further justification and staff approval prior to their use.

b. Non-linear analyses used to account for gaps between pipe and its supports when subjected to vibratory loads with significant high frequency has not been addressed by GE at this time for the ABWR piping design. The staff position is that the description of and justification for such analyses must be submitted to the staff for review and approval prior to their use.

5.7 Fatigue Evaluation for ASME Code Class 1 Piping

The ASME Code, Section III requires that the cumulative damage due to fatigue be evaluated for all ASME Code Class 1 piping. The cumulative fatigue usage factor should take into consideration all cyclic effects caused by the plant operating transients listed in Table 3.9-1 of the ABWR SSAR. For a 60-year design life, the number of cycles for each transient listed in Table 3.9-1 shall be multiplied by a factor of 1.5. However, recent test data indicates that the effects of the reactor environment could significantly reduce the fatigue resistance of certain materials. A comparison of the test data with the Code requirements indicates that the margins in the ASME Code fatigue design curves might be less than originally intended. The staff is currently developing an interim position to account for the environmental effects in the fatigue design of the affected materials which will be available at a later date.

For the ABWR, GE discussed with the staff its tentative procedure that it is currently using for a foreign boiling water reactor plant design. The information was provided to the staff during an audit held at the GE offices in San Jose, California on March 23-26, 1992. The specified material for the ASME Code Class 1 piping in the ABWR is carbon steel. Using the GE position, additional fatigue evaluations would not be required when certain conditions are met, such as when the fluid temperature is below 245°C, the oxygen content is below 0.3 ppm, or the tensile stress hold time does not exceed 10 seconds. The exemption rules also extend to piping elbows and tees and valve bodies when these components are conservatively designed and analyzed using the stress index method. Thus, only the circumferential girth butt welds in piping are considered to be critical by GE and are evaluated for environmental effects. The approach used by GE to account for the environmental effects on the girth butt welds is to modify the local peak stress through four factors: (1) the notch factor, (2) the mean stress factor, (3) the environmental correction factor and (4) the butt weld strength reduction factor.

The staff is currently reviewing the approach used by GE for accounting for the environmental effects on the fatigue life of the ASME Code Class 1 components. The results of the staff's review will be provided in a supplement to this safety evaluation. GE should include in its SSAR the proposed approach for accounting for the environmental effects in its fatigue analyses.

5.8 Fatigue Evaluation of ASME Class 2 and 3 Piping

Section 3.9.3.1 in the SSAR states that the design life for the ABWR is 60 years. In response to a staff request, GE provided a commitment in Sections 3.9.3.1 and 3.9.7.2 that applicants referencing the ABWR design will identify all ASME Code, Class 2, 3, and Quality Group D components that will be subjected to loadings that could result in thermal or dynamic fatigue and provide the analyses required by ASME Code, Section III, Subsection NB (ASME Class 1). These analyses will include the appropriate operating vibration loads and will account for the effects of mixing hot and cold fluids. Examples of such piping components shall include, as a minimum, the safety-relief valve (SRV) discharge piping in the wetwell airspace, the SRV quencher devices, and the connection of the residual heat removal/reactor water cleanup piping to the main feedwater piping outside containment. The staff finds this commitment to be acceptable.

Based on current data, the staff is of the opinion that the margins built in the ASME fatigue design curves might not be sufficient to account for variations in the original fatigue test data due to various environmental effects. Therefore, consistent with the staff position discussed in Section 5.7 for ASME Class 1 piping, the staff's position for ASME Code Class 2 and 3 piping for which a fatigue analysis is performed is that the environmental effects shall be considered in the fatigue analysis.

5.9 Thermal Stresses in Piping Connected to the Reactor Coolant System

In accordance with NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," the staff is requesting that licensees and applicants review systems connected to the reactor coolant system to determine whether any sections of such piping that cannot be isolated can be subjected to stresses from temperature stratification or temperature oscillations that could be induced by leaking valves. The staff requested that GE review the ABWR design to determine if this phenomenon could occur.

In response to this request, GE stated that in the ABWR design, the systems connected directly to the reactor coolant system or the reactor pressure vessel (RPV) are the nuclear boiler system and the emergency core cooling systems. In the nuclear boiler system, the feedwater subsystem that will supply makeup water to the RPV was thoroughly reviewed by GE from the point of view of temperature stratification. The design requirements for temperature stratification of feedwater piping were satisfactorily defined in system specification and piping cycle diagrams.

In the design of the emergency core cooling systems, both the residual heat removal (RHR) system and high-pressure core flooder (HPCF) have piping that is directly connected to the RPV. In the unisolable sections of RHR piping, leaking toward the RPV cannot occur because the pressure will always be higher on the reactor side during normal plant operation when the upstream pumps are not operating. In the HPCF system design, the only unisolable piping connected to the RPV is the section of pipe between the reactor nozzle and the upstream isolation check valve. Cold water in this system is at the upstream of the injection valve (gate valve) that is outside the primary containment. The region upstream of the injection valve will operate at a pressure lower than reactor pressure except when the HPCF safety function is required. Therefore, cold water will not flow to the unisolable pipe section and stratification will not be a problem in the HPCF system.

On the basis of the above information, the staff concludes that the ABWR design adequately addresses the potential problems described in Bulletin 88-08.

5.10 Safety-Relief Valve Design, Installation, and Testing

The staff reviewed Section 3.9.3.3 in the SSAR with respect to the design, installation, and testing criteria applicable to the mounting of pressure-relief devices used for the overpressure protection of ASME Code, Class 1, 2, and 3 components. This review which was conducted in accordance with SRP Section 3.9.3 included an evaluation of the applicable loading combinations and stress criteria. The design review extended to consideration of the means provided to accommodate the rapidly applied reaction force when a safety valve or relief valve opens, and the transient fluid-induced loads applied to the piping downstream of a safety valve or relief valve in a closed discharge piping system. The information in Section 3.9.3.3, Amendment 3 to the SSAR, meets the applicable guidelines of SRP Section 3.9.3 and is, therefore, acceptable.

In accordance with Item II.D.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," both pressurized water reactor and boiling water reactor licensees and applicants are required to conduct testing to qualify the reactor coolant system relief and safety valves and associated piping and supports under expected operating conditions for design-basis transients and accidents. GE's response to Item II.D.1 is briefly discussed in Section 1A.2.9, Appendix 1A of the SSAR. This section states that the safety/relief valve models that will be used for ABWR plants have been tested under ABWR steam discharge conditions. It further states that if the ABWR design should contain any safety/relief valves or discharge piping that is not similar to those that have been tested, the valves will be tested in accordance with NUREG-0737, Item II.D.1. The staff finds this commitment to be acceptable.

In performing the hydraulic transient piping analyses associated with the safety and relief valve (SRV) discharge, GE assumed a minimum rise time of 20 msec. Rise times faster than this value could result in higher loads than analytically predicted. The assumed rise time is based on past SRV designs and existing test data. Contingent upon the commitment described above to retest the SRVs if the COL applicant should purchase any SRV or install its SRV piping in a configuration that is not similar to those that have been tested, then this approach is acceptable to the staff.

The COL applicant should confirm that any safety-relief valves or discharge piping installed in the ABWR standard plant that is not similar to those that have been tested, will have been tested in accordance with NUREG-0737, Item II.D.1. This is a COL action item.

5.11 Functional Capability

In Note (6) to Table 3.9-2 of the SSAR, GE stated that all ASME Code Class 1, 2, and 3 piping systems which are essential for safe shutdown under the postulated events are designed to meet the requirements of GE's topical report NEDO-21985, "Functional Capability Criteria for Essential Mark II Piping," dated September 1978. This report has been previously approved by the NRC staff in its evaluation of the topical report dated July 17, 1980 and its results are still applicable to the ABWR piping. Thus, the staff finds the methodology for ensuring the functional capability of essential piping systems is acceptable.

5.12 Combination of Inertial and Seismic Motion Effects

The piping analyses shall consider the effects caused by the relative building movements at supports and anchors (seismic anchor motion) as well as the seismic inertial loads. This is necessary when piping is supported at multiple locations within a single structure or is attached to two separate structures.

The effects of relative displacements at support points shall be evaluated by imposing the maximum support displacements in the most unfavorable combination. This can be performed using a static analysis procedure. Relative displacements of equipment supports (e.g., pumps or tanks) shall be included in the analysis along with the building support movements.

When required for certain evaluations, such as support design, the responses due to the inertia effect and relative displacement effect should be combined by the absolute sum method.

In lieu of the above method, time histories of support excitations may be used in which case both inertial and relative displacement effects are already included.

5.13 Cut-off Frequency for Hydrodynamic Loadings

As discussed in SSAR Section 3.9.2.2.1, the minimum cut-off frequency for dynamic analysis of suppression pool hydrodynamic loads is 60 Hertz which was based on a generic study using the missing strain energy method for representative BWR equipment under high-frequency input loadings. This cut-off frequency was previously used in the hydrodynamic analyses for currently operating BWR plants. Because the hydrodynamic load methodology used for the ABWR is the same as that used for the operating BWR plants, the staff finds that the cut-off frequency is also appropriate for the ABWR and is thus acceptable.

5.14 OBE as a Design Load

The NRC staff is currently proposing rulemaking to revise 10 CFR Part 100, Appendix A to decouple the operating basis earthquake (OBE) from the safe shutdown earthquake (SSE) or possibly eliminate the OBE from design altogether for advanced light water reactors. For the ABWR, GE proposed that the OBE be equal to one-third of the SSE. For the evaluation of the ABWR plant components, GE will use the maximum OBE ground motion equal to one-half of the maximum SSE ground motion. However, the rulemaking is not expected to be finalized until after the design certification of the ABWR. Thus, at this time, the staff position is that ABWR shall include the OBE as a design load and it shall be equal to one-half of the SSE. The loads and load combinations in the SSAR currently include the OBE as a design loading for the ABWR and are thus acceptable.

5.15 Welded Attachments

For the analysis of local stresses at welded attachments to piping (e.g., lugs, trunnions, or stanchions), GE proposed in its SSAR to use several ASME Code Cases. Code Case N-318-3, "Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping, Section III, Division 1," is acceptable to the staff and is endorsed in RG 1.84. It should be noted that in RG 1.84, the Code Case is conditionally approved based on the applicant specifying in its SAR (1) the method of lug attachment, (2) the piping system involved, and (3) the location in the system where the Case is to be applied. The staff finds that for the ABWR design certification, this information is not needed to reach a safety conclusion and therefore is not required.

Code Cases N-391, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping, Section III, Division 1," and N-392, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 2 and 3 Piping, Section III, Division 1," are endorsed by the staff in RG 1.84 and are thus acceptable.

5.16 Modal Damping for Composite Structures

The issue of modal damping for composite structures was reviewed by the staff in the audit conducted on March 23-26, 1992, at GE's offices in San Jose, California. The GE SSAR does not describe the application of modal damping for composite structures in the analysis of piping systems. However, a review of a GE internal document entitled, " Piping Systems Design Criteria and Analysis Methods," contained a table of damping values for various types of piping supports. The damping values for the piping supports (e.g., snubbers and struts) were higher than the damping values tabulated for the piping.

GE indicated that these values were presented because modal damping for composite structures could be used in a response spectrum analysis as an option. If GE plans to use the modal damping for composite structures as an option for piping analysis, then a description and justification of the approach must be provided in the SSAR for staff review and approval prior to its use.

5.17 Minimum Temperature for Thermal Analyses

GE has not provided the staff any information that would establish a minimum temperature at which an explicit piping thermal expansion analysis would be required. Unless GE provides this information in the SSAR, the staff requires that thermal analyses will be performed for all temperature conditions above ambient.

5.18 Conclusions

Contingent upon GE providing an acceptable revision to its SSAR that addresses the lack of information identified above and reflects the staff's positions as indicated, the staff concludes the following:

On the basis of its review of Section 3.9.1 of the SSAR, the staff concludes that the design transients and resulting load combinations with appropriate specific design and service limits for mechanical components and supports are acceptable and meet the applicable portions of GDC 1, 2, 14, and 15; 10 CFR Part 50, Appendix B; 10 CFR Part 100, Appendix A; and SRP Section 3.9.1.

GE met GDC 14 and 15 by demonstrating that the design transients and resulting loads and load combinations with appropriate specific design and service limits that GE used for designing ASME Code Class 1 piping and supports provide a complete basis for the design of the reactor coolant pressure boundary piping for all conditions and events expected over the service lifetime of the plant.

GE met GDC 2 and 10 CFR Part 100, Appendix A, by including seismic events in design transients that serve as the design basis for withstanding the effects of natural phenomena.

On the basis of its review of the SSAR Section 3.9.2 and its audit of the specific design procedures for the ABWR piping systems, the staff concludes that GE met GDC 2 with respect to ensuring the design adequacy of all seismic Category I piping systems and their supports to withstand earthquakes by meeting the positions of RGs 1.61 and 1.92 or acceptable alternatives and by providing acceptable seismic analysis procedures and criteria that are consistent with applicable guidelines in SRP Section 3.9.2.

On the basis of its review of Sections 3.9.3.3 and 1A.2.9 of the SSAR, the staff finds that GE met GDC 1, 2, and 3 with respect to the criteria to be used for the design and installation of ASME Code, Class 1, 2, and 3 overpressure-relief devices by ensuring that safety and relief valves and their installations will be designed to standards that are commensurate with their safety functions, and that they will accommodate the effects of pressure relief caused by normal operation as well as postulated events such as LOCAs and the dynamic effects resulting from the SSE. GE also met GDC 14 and 15 with respect to ensuring that the reactor coolant pressure boundary design limits for normal operation, including anticipated operational occurrences, will not be exceeded. The criteria used by GE in the design and installation of ASME Code, Class 1, 2, and 3 safety and relief valves provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure-relief devices provides a conservative basis for the design and installation of the devices for ensuring that the devices will withstand these loads without loss of structural integrity or impairment of the overpressure-protection function.

On the basis of its review of Section 3.9.3.1 of the SSAR, the staff finds that GE met GDC 2 and 4 with respect to the design and service load combinations and associated stress and deformation limits specified for ASME Code, Class 1, 2, and 3 components by ensuring that these systems and components can accommodate the effects of normal operation as well as postulated events such as LOCAs and the dynamic effects resulting from earthquakes. The specified design and service combinations of loading as applied to ASME Code, Class 1, 2, and 3 pressure-retaining components in systems designed to meet seismic Category I standards provide assurance that, in the event of an earthquake affecting the site or other service loading caused by postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting stresses under such loading combinations provides a conservative design basis for ensuring that the system components will withstand the most adverse combination of loading events without loss of structural integrity.

6 PIPE SUPPORT CRITERIA

6.1 Applicable Codes

The staff's review of Sections 3.9.3.4 and 3.9.3.5 of the SSAR relates to the methodology used in the design of ASME Code Class 1, 2, and 3 component supports. The review included an assessment of the design and structural integrity of the supports. It addressed three types of supports: plate and shell, linear, and component standard types. All ASME Code Class 1, 2, and 3 component supports for the ABWR standard plant shall be constructed in accordance with ASME Code, Section III, Subsection NF, "Component Supports." In addition, GE states in its SSAR that the design is augmented by the application of Code Case N-476, Supplement 89.1 which governs the design of single-angle members. Also, when eccentric loads or other torsional loads are not accommodated by designing the load to act through the shear center or does not meet "Standard for Steel Support Design," then analyses will be performed in accordance with the torsional analysis methods such as " Torsional Analysis of Steel Members, USS Steel Manual," Publication T114-2/83. The staff position is that Subsection NF is an acceptable code for the design of piping supports. However, the rules shall be augmented by acceptable guidelines governing the design of single-angle members of supports and the methodology used to accommodate torsional loads. At this time, although Code Case N-476 has not yet been endorsed by the staff in RG 1.84, the staff finds that it provides adequate design rules for the single-angle members. For torsional analysis of steel members, the staff's review of the GE proposed documents finds that they provide sufficient technical guidelines to perform a torsional analyses of steel members and are thus acceptable.

The staff has not yet endorsed the use of ANSI/AISC N-690, "Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities," in lieu of Subsection NF.

6.2 Jurisdictional Boundaries

In Section 3.9.3.4 of the SSAR, GE committed to define the jurisdictional boundaries between pipe supports and interface attachment points such as structural steel in accordance with the ASME Code, Section III, Subsection NF, 1989 Edition. The staff's review of the jurisdictional boundaries described in the 1989 Edition finds that they are sufficiently defined to ensure a clear division between the pipe support and the structural steel and are thus acceptable.

6.3 Loads and Load Combinations

In Section 3.9.3.4.1 of the SSAR, GE states that the loading combinations for the design of piping supports correspond to those used for the design of the supported pipe. The staff's evaluation of the load combinations for the supported pipe is contained in Section 5.3 of this safety evaluation. The stress limits for pipe supports are in accordance with the ASME Code, Section III, Subsection NF and Appendix F. The supports are generally designed or qualified by the load rating method as described in paragraph NF-3260 or by the stress limits specified in paragraph NF-3231. The staff's review of these methods and limits as specified in the 1989 Edition of the ASME Code, Section III finds them to be acceptable.

6.4 Pipe Support Baseplate and Anchor Bolt Design

Section 3.9.3.4 states that concrete anchor bolts that will be used for pipe support base plates will be designed to the applicable factors of safety defined in Office of Inspection and Enforcement (IE) Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," Revision 1, June 21, 1979. Loading combinations for component supports are discussed in Section 6.3 above. The staff finds that, in general, the factors of safety for anchor bolts are acceptable.

However, GE has not discussed the use of specific types of anchor bolts to be used in the ABWR standard plant. For example, under-cut type anchor bolts behave in a ductile manner but the staff's position is that the safety factors in IE Bulletin 79-02 shall still be applicable unless justification for alternative safety factors is provided. GE has not provided any guidelines for use of under-cut type anchor bolts in the ABWR piping systems. Therefore, the use of safety factors for anchor bolts other than those provided in IE Bulletin 79-02 shall be justified and submitted to the staff for review and approval prior to their use.

Irrespective of the type of concrete anchor bolt used for piping supports, the action item in IE Bulletin 79-02 relative to pipe support baseplate flexibility shall be implemented.

6.5 Use of Energy Absorbers

GE has not provided to the staff any details regarding the specific analysis methods or procedures to be used for the ABWR pipe support design. In order for the staff to complete its review, GE should include in the SSAR additional details of the pipe support design that addresses the use of seismic restraints other than snubbers and their modeling assumptions.

6.6 Use of Snubbers

GE has not provided to the staff any details regarding the specific analysis methods or procedures to be used for the ABWR pipe support design. In order for the staff to complete its review, GE should include in the SSAR additional details of the pipe support design that addresses the types of snubbers and their character[illegible]cs to be used in the ABWR standard plant.

In addition, the dynamic qualification testing and periodic functional testing of large-bore hydraulic snubbers are important to verify that the snubbers are adequately designed and maintained for the life of the plant as discussed in Generic Issue 113, " Dynamic Qualification and Testing of Large Bore Hydraulic Snubbers." In order to complete its review, the staff requires that GE submit in its SSAR additional details addressing the environmental (including dynamic) qualification and inservice inspection and testing requirements for large-bore hydraulic snubbers (rated at 50 kips or greater) if they are intended to be used in the ABWR standard plant.

6.7 Pipe Support Stiffnesses

GE has not provided to the staff any details regarding the specific analysis methods or procedures to be used for the ABWR pipe support design. In order for the staff to complete its review, GE should include in the SSAR additional details of the pipe support design that addresses the pipe support stiffness values and support deflection limits used in the piping analyses.

6.8 Seismic Self-Weight Excitation

GE has not provided to the staff any details regarding the specific analysis methods or procedures to be used for the ABWR pipe support design. In order for the staff to complete its review, GE should include in the SSAR additional details of the pipe support design that addresses how the seismic excitation of the pipe supports (especially large frame-type structures) are to be considered in the design of the pipe support anchorage.

6.9 Design of Supplementary Steel

In Section 3.9.3.4 of the SSAR, GE provided its design criteria for the design of pipe supports using supplementary steel. The building structure component supports are designed in accordance with AISC, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings." The use of this specification is standard industry practice and has been proven to provide adequate design guidelines for the design of structural steel for use as pipe supports. Thus, the staff finds the specification to be acceptable.

6.10 Consideration of Friction Forces

GE has not provided to the staff any details regarding the specific analysis methods or procedures to be used for the ABWR pipe support design. In order for the staff to complete its review, GE should include in the SSAR additional the coefficient of friction to be used for considering friction forces between the pipe and the steel frames.

6.11 Pipe Support Gaps and Clearances

GE has not provided to the staff any details regarding the specific analysis methods or procedures to be used for the ABWR pipe support design. In order for the staff to complete its review, GE should include in the SSAR additional details of the pipe support design that addresses the hot and cold gaps to be used between the pipe and the box frame-type of support.

6.12 Instrumentation Line Support Criteria

GE has not provided the staff any information on the design criteria for the structural design of instrumentation line supports.

The industry has taken the position that ANS/AISC N-690 is useful in the design of instrumentation sensing line supports and has recommended that the industry be allowed to use it. Its use would have the effect of reducing the QA recordkeeping requirements and Code stamping required by Section NF of the

ASME Boiler and Pressure Vessel Code, Section III. The staff's position on this issue is that for construction of ASME component supports, ANS/AISC N-690 alone is not an acceptable standard. ASME Code, Section III, Subsection NF should be used. However, the staff is currently participating in the ASME effort to incorporate N-690 into Subsection NF. Subsequent to a staff-endorsed version of NF incorporating N-690 into it, Subsection NF will also specify the rules acceptable to the staff for construction of ASME Class supports. When this staff-approved version is available, the COL holder seeking to use it may submit a request to the staff for approval on a plant-specific basis.

6.13 Pipe Deflection Limits

GE has not provided to the staff any details regarding the specific analysis methods or procedures to be used for the ABWR pipe support design. In order for the staff to complete its review, GE should include in the SSAR additional details of the pipe support design criteria that will ensure that the maximum deflections of the piping at support locations for static and dynamic loadings are within an allowable limit to preclude failure of the pipe supports and hangers.

6.14 Conclusions

GE has not provided to the staff any details regarding the specific analysis methods or procedures to be used for the ABWR pipe support design. In order for the staff to complete its review, GE should include in the SSAR additional details of the pipe support design that addresses as a minimum (1) the jurisdictional boundaries, (2) the design of supplementary steel (e.g., frames) for pipe supports, (3) pipe support stiffnesses, (4) types of baseplate anchor bolts to be used and their safety factors, (5) types of piping seismic restraints to be used and their modeling assumptions, (6) consideration of friction forces, (7) seismic self-weight excitation, (8) pipe support gaps and clearances, and (9) deflection limits.

Contingent upon GE providing an acceptable revision to its SSAR to address the above issues and to reflect the above staff positions, and on the basis of its review of Section 3.9.3.4 in the SSAR, the staff finds that GE met 10 CFR 50.55a and GDC 1, 2, and 4 with respect to the design and service load combinations and associated stress and deformation limits specified for ASME Code, Class 1, 2, and 3 component supports by ensuring (1) that component supports important to safety will be designed to quality standards commensurate with their importance to safety and (2) that these supports will accommodate the effects of normal operation as well as postulated events such as LOCAs and the dynamic effects resulting from the SSE. The combination of loadings (including system operating transients) considered for each component support within a system, including the designation of the appropriate service stress limit for each loading combination, will be in accordance with the SRP Section 3.9.3. The specified design and service loading combinations used for the design of ASME Code Class 1, 2, and 3 component supports in systems classified as seismic Category I provide assurance that, in the event of an earthquake or other service loadings caused by postulated events or system operating transients, the resulting combined stresses imposed on system components and component supports will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative design basis for ensuring that component support will withstand the most adverse combination of loading events without loss of structural integrity.

7 HIGH ENERGY LINE BREAK CRITERIA

GDC 4 requires that structures, systems, and components important to safety be designed to be compatible with and to accommodate the effects of the environmental conditions resulting from normal operations, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. It also requires that they be adequately protected against dynamic effects (including the effects of missiles, pipe whipping, and discharging fluids) that may result from equipment failures and from events and conditions outside the nuclear power plant.

The staff reviewed, in accordance with SRP Section 3.6.2, Revision 2, June 1987, the criteria and methodology proposed by GE for the COL holder to use to analyze the effects that breaks in high-energy fluid systems would have on adjacent safety-related structures, systems, or components with respect to pipe whip and jet impingement loadings. The criteria and methodology discussed herein shall be used by the COL holder to ensure adequate protection against the dynamic effects of postulated ruptures of piping in the ABWR standard design.

7.1 High Energy Piping Systems

Pipe whip need only be considered for those high-energy piping systems having fluid reservoirs with sufficient capacity to develop a jet stream. The criteria for determining high- and moderate-energy lines in SRP Section 3.6.1, Branch Technical Position (BTP) ASB 3-1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," were adequately defined by GE in Section 3.6.2.1 of the SSAR. All high-energy systems are listed in Tables 3.6-3 and 3.6-4 of the SSAR.

7.2 Pipe Break Criteria Within the Containment Penetration Areas

In the ABWR breaks are not postulated in those portions of high-energy piping between the containment isolation valves outside and inside the containment that are designed to meet ASME Code, Section III, Article NE-1120, and the additional design guidelines in SRP Section 3.6.2, including BTP MEB 3-1, Revision 2, June 1987. These guidelines recommend that an augmented inservice inspection program be implemented for those portions of piping within the break exclusion region. For the ABWR, the COL holder is committed to perform a 100 percent volumetric examination of circumferential and longitudinal pipe welds in the break exclusion region during each inspection interval as defined in Article IWA-2400, ASME Code, Section XI. The staff finds that the above commitment for the design and examination of high-energy piping in the containment penetration area meets SRP Section 3.6.2 and is acceptable.

7.3 Pipe Break Criteria Outs[illegible] Containment Penetration Areas

Fo. ASME Code, Class 1, 2, and [illegible], and non-ASME seismic Category I high- and moderate-energy lines that are not in the containment penetration area, GE, in Section 3.6.2 of the SSAR, presented the criteria for determining postulated rupture and crack locations and the methodology used to evaluate the dynamic effects of pipe whip, jet thrust, and jet impingement that result from such breaks.

SRP Section 3.6.2 also states that for the final design approval, GE should include the following in the SSAR:

- o Sketches of applicable piping systems showing the location, size, and orientation of postulated pipe breaks and the location of pipe whip restraints and jet impingement barriers.

- o A summary of the data developed to select postulated break locations. This should include calculated stress intensities, cumulative usage factors and stress ranges as delineated in SRP Section 3.6.2, BTP MEB 3-1.

In the SSAR, Section 3.6.4.1, "Interfaces," Amendment 10, GE identified the above information as a confirmatory-type interface that must be submitted by the applicants that reference the ABWR design. The staff concludes that this is an acceptable COL action item for the ABWR standard plant.

During an audit of the high energy piping criteria and sample calculations performed at the offices of GE in San Jose, California on March 23-27, 1992, the staff found that the description of the pipe whip analyses being performed and the design requirements for the pipe whip restraints being considered for the ABWR were not in accordance with the description and commitments in Sections 3.6.2.2.2 and 3.6.2.3.3 of the SSAR, respectively. Section 3.6.2.2.2 of the SSAR states that the pipe whip analyses were performed using the PDA computer program. Section 3.6.2.3.3 identifies four types of pipe whip restraint components and their associated material, inspection, and design limits.

At this time, the representative pipe whip analyses for breaks in the main steamline were not complete. However, analyses were being performed using the COMET and the ANSYS computer programs. These analyses were also intended to demonstrate that the stress limits in Branch Technical Position MEB 3-1, Revision 2, Section B.1.b.(1).(c) for the effects of pipe failure in ASME Code Class 1 piping in the containment penetration areas were satisfied. The PDA computer program is capable only of analyzing a straight uniform pipe which is fixed at one end, restrained by an intermediate pipe whip restraint and subjected to a time-dependent thrust force at the other end. Hence, it is not suitable for the intended analyses and should not be used for the ABWR pipe whip analyses.

In addition , the pipe whip restraints were not being designed in accordance with the criteria in Section 3.6.2.3.3 of the SSAR but were planned to be selected from standard GE Y-1000 restraints of U-rod type design.

Therefore, GE shall revise its SSAR to describe the computer programs it used for pipe whip analyses and the design methodology for pipe whip restraints that are applicable to the ABWR plant design.

It was also found that the referenced edition of ANSI/ANS-58.2 in Section 3.6.2.2.1 of the SSAR was not current and that the criteria on Section 3.6.2.3.1 for evaluating the effects of fluid jets on essential structures, systems and components were not in complete agreement with the guidelines of SRP Section 3.6.2 dated July 1981 and with the ANSI/ANS-58.2 standard, 1988 Edition.

Accordingly, GE should revise its SSAR in Section 3.6.2.2.1 to update its reference to the 1988 edition of the ANSI/ANS-58.2 and revise the criteria in SSAR Section 3.6.2.3.1 to be consistent with the SRP Section 3.6.2 and ANSI/ANS-58.2 (1988).

7.4 Conclusions

GE has not provided to the staff sufficient details regarding the specific analysis methods or procedures to be used for the ABWR high energy line break design. In order for the staff to complete its review, GE should include in the SSAR additional details of the high energy line break analysis methods that are intended to be used for (1) the completion of the pipe whip and jet impingement analyses and (2) the design and qualification of the pipe whip restraints and jet impingement shields.

Contingent upon GE providing an acceptable revision to its SSAR addressing the above inconsistencies and on the basis of its review of Section 3.6.2 of the SSAR, the staff concludes that the criteria for postulating pipe rupture and crack locations and the methodology for evaluating the subsequent dynamic effects resulting from these ruptures comply with SRP Section 3.6.2, meet GDC 4 and, therefore, are acceptable for ensuring that the ABWR plant design is adequately protected against the effects of postulated high energy line breaks. The staff's conclusion is based on the following.

The proposed pipe rupture locations will be adequately determined using the above staff-approved criteria and guidelines. The design methods for high energy mitigation devices and the measures to deal with the subsequent dynamic effects of pipe whip and jet impingement have been sufficiently and adequately defined by GE to provide adequate assurance that upon completion of the high energy line break analyses by the COL holder, the ability of safety-related structures, systems, and components to perform their safety functions will not be impaired by the postulated pipe ruptures.

The provisions for protection against the dynamic effects associated with pipe ruptures of the reactor coolant pressure boundary inside the containment and the resulting discharging fluid provides adequate assurance that design-basis loss-of-coolant accidents will not be aggravated by the sequential failures of safety-related piping and that the performance of the emergency core cooling system will not be degraded as a result of these dynamic effects.

The arrangement of piping and restraints and the final design considerations for high- and moderate-energy fluid systems inside and outside the containment, including the reactor coolant pressure boundary, shall be the responsibility of the COL holder to complete and shall use the above staff-approved high-energy line break criteria and guidelines to provide the assurance that the structures, systems, and components important to safety that are in close proximity to the postulated pipe ruptures will be protected. In using the above criteria and guidelines, the staff is assured that the consequences of pipe ruptures will be adequately mitigated so that the reactor can be safely shut down and be maintained in a safe-shutdown condition in the event of a postulated rupture of a high- or moderate-energy piping system inside or outside the containment.

8 LEAK-BEFORE-BREAK CRITERIA

In a letter dated February 3, 1992, GE committed to provide in a future amendment to its SSAR Section 3.6.3 and Appendix 3E, a description of the evaluation procedures for a leak-before-break (LBB) methodology. The use of the leak-before-break approach has not been pre-approved by the staff in the ABWR design certification phase; but, rather is a design option for the COL holder to consider in the COL phase in lieu of performing high energy line break analyses as discussed in the above Section 7. The staff evaluation provided herein provides guidance on the approach to be used and the material required to be submitted by the COL holder to the staff in its request for approval of the leak-before-break option.

The application of the LBB methodology to piping systems is permitted in GDC 4 of 10 CFR Part 50, Appendix A. GDC 4 states, in part, that "dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."

The analyses referred to in the GDC 4 rule (Federal Register, Vol. 52, No. 207, October 27, 1987, 41288-41295,) should be based on specific plant data, such as piping geometry, materials, piping loads, and pipe support locations. The staff must review the LBB analyses for specific piping designs before the applicant can exclude the dynamic effects from the design basis for the piping system.

GE intends to submit in its future amendment to SSAR Section 3.6.3 and Appendix 3E, an LBB methodology, not a plant-specific analysis. It is GE's intent that COL holders referencing the ABWR design will use staff-approved LBB methodology and acceptance criteria in effect at that time in their detailed plant-specific piping analyses during the COL phase. This approach is consistent with the staff's position in that GDC 4 rule calls only for LBB analyses, not methodology, to be reviewed and approved. In addition, the staff has not yet reviewed any LBB requests by currently operating BWR plants at this time, in part, because stainless steel piping materials in a BWR environment are susceptible to intergranular stress corrosion cracking (IGSCC). For the ABWR, GE stated that it would use IGSCC-resistant piping materials and implement IGSCC-mitigating measures; however, GE has yet to provide sufficient data to convince the staff about the effectiveness of these IGSCC-mitigating measures for any "IGSCC-resistant" materials that might be used in ABWRs.

Utility applicants referencing the ABWR design must submit an LBB plant-specific analysis in accordance with GDC 4. In addition, these applicants must provide information on resistance of the materials to IGSCC and proposed IGSCC-mitigating measures.

The staff reviewed the technical content of the proposed amendment to the SSAR Section 3.6.3 and Appendix E during an audit performed at the offices of GE in San Jose, California on March 24-27, 1992. The purpose of the audit was to evaluate the technical content of the LBB example analysis to assess whether the analysis approach was consistent with the current guidelines and acceptance criteria established by the staff in SRP 3.6.3 (draft dated August 1987) and NUREG-1061, Volume 3 dated November 1984. The staff found the following.

(1) In Section 3E.2.1 of Appendix 3E, GE proposed a modified tearing modulus (J/T) methodology for LBB evaluations. The modified methodology was based on the modified J-integral, J_{mod}, and tearing modulus, T_{mod}, proposed by Ernst (Reference 8 in SSAR Section 3E.1.2). Justification for the proposed (J_{mod}/T_{mod}) procedure was not provided by GE.

(2) In Section 3E.2.2.1 of Appendix 3E, a description of a carbon steel fracture toughness test program was provided. The extent of the program described might not be representative of the actual test program required for approval of a request for LBB application in order to qualify selected piping systems.

(3) In Section 3E.2.1.2 of Appendix 3E, GE proposed a linear interaction criterion for tearing instability evaluations for combinations of applied tension and bending stresses. Justification for the proposed criterion was provided.

(4) In Section 3E.4 of Appendix E, GE proposed a procedure for estimation of leak rates during blowdown of saturated steam. The validation of the proposed procedure was not provided.

(5) The criteria and procedures for bi-metallic welds were not provided in Section 3.6.3 of the SSAR.

(6) The materials specified in Section 3E.2 and 3E.6 of Appendix 3E are inconsistent. Furthermore, materials that are currently unavailable or materials that are not intended to be used in LBB candidate piping were identified. Currently, the staff has not approved LBB in any carbon steel piping that is not clad with stainless steel, in part, due to concerns with erosion-corrosion.

Accordingly, these issues need to resolved in LBB submittals by the COL holder intending to use the leak-before-break approach in lieu of postulating high energy line breaks in the ABWR.

The staff concludes that utility applicants seeking approval of the leak-before-break approach for high energy piping systems in the ABWR shall submit to the NRC staff an LBB plant-specific analysis in accordance with GDC 4. Although the staff is currently using the methodology and acceptance criteria provided in SRP Section 3.6.3 and NUREG-1061, Volume 3 and the GE example analysis was evaluated using these guidelines, the staff recognizes that the LBB technology is continually evolving. Therefore, staff evaluations of LBB requests for the ABWR plant will reviewed on a case-by-case basis using the staff's methodology and acceptance criteria in effect at the time of the submittal.

9 GENERIC PIPING DESIGN ITAAC

In Section 3.3 of the ABWR Design Document, GE provided its inspections, tests, analyses, and acceptance criteria (ITAAC) for piping design. The staff reviewed Table 3.3 which identified 12 certified design commitments for the ABWR piping design and the corresponding ITAAC.

The staff's evaluation of the 12 certified design commitments and ITAAC follows.

9.1 Fatigue

GE provided a certified design commitment that the piping shall be designed for a fatigue life of 60 years. For ASME Code Class 1 piping systems, a fatigue analysis will be performed in accordance with the applicable ASME Boiler and Pressure Vessel Code, Section III requirements. For ASME Code Class 2 and 3 piping, Section III rules will be followed using a stress reduction factor of 1.0 for those piping systems expected to experience fewer than 7000 thermal cycles in its 60 year design life.

An inspection of the certified stress report by the COL holder shall be conducted to assure that the fatigue evaluation meets the ASME Code requirements and with the 60 year design life.

The acceptance criteria for the fatigue design of ASME Code Class 1 piping shall be that the cumulative usage factor is less than 1.0.

The staff finds that the fatigue design of safety-related piping is a necessary certified design commitment to ensure the integrity of the reactor coolant pressure boundary and the ability of the piping systems to perform their safety function for their 60-year design life. The design acceptance criterion for a cumulative usage factor to be less than 1.0 is consistent with current ASME Code requirements for fatigue evaluation as stated in Subparagraph NB-3222.4. For fatigue evaluation, the environmental effects shall be considered as discussed in Section 5.7 of this safety evaluation.

The inspection to be performed by the COL holder assures that the ASME Code requirements for fatigue will be satisfied upon completion of the Code-required stress report. However, the staff finds that an additional certified design commitment is needed for any ASME Code Class 2 and 3 piping system for which it is identified in the design specification that it is expected to experience 7000 or more thermal stress cycles in its 60-year design life. For any such piping, a stress reduction factor of less than 1.0 shall be used in its stress analysis as required by Subparagraph NC/ND-3611.2 of the ASME Code, Section III. In addition, for ASME Code Class 2 and 3 piping systems for which an ASME Code Class 1 fatigue evaluation is required as discussed in Section 5.8 of this safety evaluation, a cumulative usage factor of 1.0 shall be met with environmental effects considered.

Contingent upon the completion of the above ITAAC verifying that the certified design commitment meets the above-specified design acceptance criteria, the staff concludes that there is reasonable assurance that the design of the ABWR piping systems will be adequately evaluated for fatigue effects and is, thus, acceptable.

9.2 Pipe-mounted Equipment Allowable Loads

GE provided a certified design commitment that the loads imposed by the piping system on pipe-mounted equipment and attachment interfaces shall meet the vendor allowable loads. A COL action item will require that these loads be documented for comparison with the vendor's allowable loads.

An inspection of the design documents will be performed to verify that the as-designed interface loads meet the vendor's specified allowable loads.

The staff finds that upon completion of the as-built piping analyses, it is necessary to ensure that the calculated loads imposed by the piping on the equipment nozzles and other attachment interfaces are within the vendor's recommended allowable values. This verification will ensure that the equipment and supports will function as intended under normal operating, transient, and accident conditions.

Contingent upon the completion of the ITAAC by the COL holder verifying that the calculated piping loads are within the equipment and interface allowable loads, the staff concludes that there is reasonable assurance that the ABWR pipe-mounted equipment and piping attachment interfaces will adequately satisfy the vendor interface allowable limits to ensure that the equipment can perform their intended safety functions under normal, operating, transient, and accident loading conditions.

9.3 Piping Analysis Methods

GE provided a certified design commitment that would require the analytical methods and component stress analyses be specified in a certified design specification. The analysis of the piping system will use a suitable dynamic method or an equivalent static load method.

An inspection of the certified design specification and certified stress report will verify that the piping method used is in compliance with the regulatory requirements.

The staff position is that the analytical methods to be used to complete the ABWR piping design shall ensure the pressure integrity, structural integrity, and the functional capability of the piping system under normal operating and accident loading conditions and shall use a suitable dynamic analysis or an equivalent static analysis method as approved by the staff. The analysis methods approved by the staff for the ABWR piping design are discussed in Section 3 of this safety evaluation. The key analysis input parameters approved by the staff for the ABWR piping analysis are discussed in Section 5 of this safety evaluation.

Contingent upon the ITAAC verifying that the completion of the ABWR piping analyses used the staff-approved analysis methods and input parameters, the staff concludes that there is reasonable assurance that the ABWR analysis methods are adequate to ensure the pressure integrity, structural integrity, and functional capability of the piping.

9.4 High Energy Line Break Analysis

GE provided a certified design commitment that would require an analysis demonstrating that essential piping systems are protected against the dynamic effects associated with the postulated rupture of high energy piping systems.

An inspection of the pipe rupture analysis report or a "leak-before-break" analysis report would be performed to verify that the safety of the plant will not be adversely impacted by the dynamic effects resulting from the postulated pipe breaks. For those impacted components needed to safely shutdown the plant, the ASME Code requirements for faulted plant conditions and operability limits shall be met. Pipe rupture mitigation devices (e.g., pipe whip restraints and jet impingement shields) shall be used to restrain the whipping pipe and deflect the blowdown loads.

The staff position is that a pipe rupture analysis shall be completed by the COL holder to demonstrate that safety-related systems, structures, and components will be protected against the dynamic effects of a postulated pipe break using the methods described in Section 7 of this safety evaluation. As an alternative, the COL holder may submit a request for staff approval to eliminate breaks using a "leak-before-break" approach as discussed in Section 8 of this safety evaluation.

Contingent upon the ITAAC verifying that the completion of the high energy line break analysis used the staff-approved analysis methods discussed above, the staff concludes that there is reasonable assurance that the safety-related systems, structures, and components in the ABWR are adequately protected against the dynamic effects of postulated high energy line breaks.

9.5 Functional Capability

GE identified a certified design commitment that all ASME Code Class 1, 2, and 3 piping systems which are essential for the safe shutdown of the plant shall be designed to assure that they will maintain sufficient dimensional stability to perform their required function under all loading conditions. In Section 5.11 of this safety evaluation, the staff evaluated the stress limits proposed by GE to ensure the functional capability of the essential piping systems. In no case shall the piping stress exceed the limits designated for Service Level D in the ASME Code, Section III. The Service Level D limits are 3.0 Sm (not to exceed 2.0 Sy) for ASME Code Class 1 piping and 3.0 Sh (not to exceed 2.0 Sy) for Class 2 and 3 piping.

An inspection of the certified stress report by the COL holder will be conducted in conjunction with ITAAC to assure that the functional capability limits have been satisfied.

The staff finds that the limits specified by GE to ensure functional capability of piping as documented in NEDO-21985 have been previously approved by the staff as discussed in Section 4.10 of this safety evaluation. The use of Service Level D limits (not to exceed 2.0 Sy) are consistent with the staff recommendations based on high-level dynamic tests sponsored by the EPRI and the NRC staff.

Contingent upon the ITAAC verifying that the piping stresses in the certified stress report satisfy the design acceptance criteria discussed above, the staff concludes that there is reasonable assurance that the piping is capable of performing its safety function under all normal operating, transient, and accident conditions.

9.6 Analytical Modeling of Piping

GE identified a certified design commitment to verify the piping analysis model for the computer code to be used by the COL holder to complete its piping stress analyses. The piping analysis model shall address the key parameters needed to ensure adequate static and dynamic characteristics of the piping system. The key parameters for the piping model are discussed in Section 4.2 of this safety evaluation. The computer program and the modeling techniques shall be evaluated using the NRC benchmark program discussed in Section 4.3 of this safety evaluation.

The verification of the sufficiency of the computer code and modeling techniques shall be performed in conjunction with ITAAC.

Contingent upon the ITAAC verifying that the piping benchmark results are within the acceptable range of values specified in the benchmark program, the staff concludes that there is reasonable assurance that the computer code and analytical modeling techniques to be used to complete the ABWR piping design and analyses are adequate.

9.7 ASME Code Stamp

GE identified a certified design commitment that the ABWR piping, its appurtenances, and its supports shall satisfy the ASME class, seismic category, and quality group requirements commensurate with their classification. An inspection shall be performed of the ASME Code-required design documents and installed components. The inspection shall verify the completion of a certified stress report and related Code-required documents and that the installed component has received an appropriate ASME Code Symbol Stamp.

The staff concludes that contingent upon the completion of the inspection verifying the existence of Code-required documents and Code Symbol Stamp, there is reasonable assurance that the piping and its subcomponents are adequately designed, fabricated, and examined in accordance with the applicable ASME Code requirements.

9.8 Fracture Toughness

GE provided a certified design commitment that the piping systems made of ferritic material shall not be susceptible to brittle fracture. Only intrinsically tough grades of ferritic materials shall be used. Fracture toughness tests shall be performed. The fracture toughness requirements of the ASME Code, Section III shall be satisfied.

The staff finds that the fracture toughness tests are in accordance with the ASME Code, Section III rules. Contingent upon the ferritic materials satisfying the ASME Code, Section III requirements, the staff concludes that there is reasonable assurance that the material for piping systems are adequately specified to preclude brittle fracture under pressure loadings for the expected service conditions.

9.9 Cracking in Stainless Steel Piping

GE provided a certified design commitment that piping systems made of austenitic stainless steel shall be selected to minimize the possibility of cracking during their 60-year design life. Special chemical, fabrication, handling, welding, and examination requirements that minimize the potential for cracking shall be satisfied. An inspection of the ASME Code-required documents and other related records will be performed to verify that the ASME Code and any special requirements have been satisfied.

The staff position is that the guidelines in NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," Revision 2, shall be followed.

Contingent upon an inspection of the records verifying that the material and processes satisfied the ASME Code and special requirements, the staff concludes that there is reasonable assurance that the austenitic stainless steel piping systems have been adequately fabricated to minimize the potential for cracking during service.

9.10 As-Built Piping Verification

GE identified a certified design commitment that the as-built piping system shall be consistent with the as-designed piping. An inspection shall be performed to verify the pipe routing configurations; the location, size, and orientation of piping supports, valves, and equipment; and to identify deviations from the as-designed condition. The piping configuration and component location, size, and orientation will be within the specified tolerances. Deviations (outside the tolerances) will be evaluated to ensure that the vendor allowable loads and ASME Code, Section III stress limits are satisfied.

The staff finds that the as-built piping verification is necessary to ensure that the as-constructed piping is consistent with the certified piping stress report. Contingent upon the completion of the inspection verifying that the installation tolerances have been satisfied and that all deviations have been reconciled using the staff-approved methods and design acceptance criteria discussed in this safety evaluation, the staff concludes that there is reasonable assurance that the ABWR piping systems are constructed in accordance with the design documents.

9.11 Pressure Integrity

GE identified a certified design commitment that the ASME Code Class 1, 2, and 3 piping shall retain its pressure integrity for its 60-year design life. The piping systems shall be designed to the requirements of the ASME Code, Section III as discussed in Section 2.1 of the safety evaluation.

An inspection shall be performed to verify that the ASME Code, Section III requirements have been satisfied for ASME Code Class 1, 2, and 3 piping systems. In addition, hydrostatic pressure tests of the ASME Class 1, 2, and 3 piping shall be performed in accordance with the ASME Code requirements.

The staff finds that the inspections and tests will provide assurance that the ASME Code, Section III requirements have been satisfied. Contingent upon the successful completion of the inspections and tests, the staff concludes that the ASME Code Class 1, 2, and 3 piping systems are designed and tested to ensure their pressure integrity in service under normal operating, testing, transient, and accident conditions.

9.12 Interferences

GE identified a certified design commitment that piping shall be designed with adequate clearances to preclude interferences with nearby systems, structures, and components resulting from piping displacements. An inspection shall be performed to verify that the maximum calculated pipe deflections under normal operating, transient, and accident conditions do not exceed the minimum specified clearances between the piping and nearby systems, structures, and components.

Contingent upon the completion of the inspection verifying that maximum calculated pipe deflections are within the minimum specified clearances, the staff concludes that there is reasonable assurance that the piping deflections under normal operating, transient, and accident conditions do not cause interferences with nearby systems, structures, and components.

9.13 Conclusions

On the basis of its review of Section 3.3 of the ABWR Design Document, the staff finds that GE met 10 CFR 52.47 with respect to submitting proposed inspections, tests, and analyses, and acceptance criteria for piping systems that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria are met, a plant which references the design will be built and will operate in accordance with the design certification. The referenced design for the ABWR incorporates the staff's approved piping design criteria and analysis methods for ensuring that the piping systems are adequately designed and will perform their safety-related functions for all postulated combinations of normal operating, system operating transients, and accident conditions.

10 OVERALL CONCLUSION

On the basis of (1) its review of the information provided in the ABWR SSAR and (2) its audit of the specific design criteria and representative sample analyses for the ABWR piping systems and (3) contingent upon GE supplementing its SSAR with the requested information and staff positions discussed above, the staff concludes that the applicant has satisfied 10 CFR 52.47 by providing sufficient information to reach a final safety conclusion on all safety questions associated with the ABWR piping design. The staff further concludes that the applicant has satisfied the applicable requirements of 10 CFR Parts 50, 52, and 100 by providing reasonable assurance that the piping systems will be designed and built in accordance with the certified design and will perform their safety-related functions under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events. The staff's conclusion is based on the use of pre-approved piping analysis methodology as a part of the design acceptance criteria by the COL holder to complete the design and analyses of the ABWR piping systems. The implementation of these pre-approved methods and satisfaction of the acceptance criteria will be verified through the performance of the ITAAC by the COL holder to ensure that the final as-built piping stress analyses and high energy line break analyses as well as the as-constructed piping systems are in accordance with the certified design commitments.

CHAPTER 3

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TABLES

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3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

This Section deals with the structures, systems, components and equipment in the ABWR Standard Plant.

Subsections 3.6.1 and 3.6.2 describe the design bases and protective measures which ensure that the containment; essential systems, components and equipment; and other essential structures are adequately protected from the consequences associated with a postulated rupture of high-energy piping or crack of moderate-energy piping both inside and outside the containment.

Before delineating the criteria and assumptions used to evaluate the consequences of piping failures inside and outside of containment, it is necessary to define a pipe break event and a postulated piping failure:

Pipe break event: Any single postulated piping failure occurring during normal plant operation and any subsequent piping failure and/or equipment failure that occurs as a direct consequence of the postulated piping failure.

Postulated Piping Failure: Longitudinal or circumferential break or rupture postulated in high-energy fluid system piping or throughwall leakage crack postulated in moderate-energy fluid system piping. The terms used in this definition are explained in Subsection 3.6.2.

Structures, systems, components and equipment that are required to shut down the reactor and mitigate the consequences of a postulated piping failure, without offsite power, are defined as essential and are designed to Seismic Category I requirements.

The dynamic effects that may result from a postulated rupture of high-energy piping include missile generation; pipe whipping; pipe break reaction forces; jet impingement forces; compartment, subcompartment and cavity pressurizations; decompression waves within the ruptured pipes and seven types of loads identified with loss of coolant accident (LOCA) on Table 3.9-2.

Subsection 3.6.3 and Appendix 3E describe the implementation of the leak-before-break (LBB) evaluation procedures as permitted by the broad scope amendment to General Design Criterion 4 (GDC-4) published in Reference 1. It is anticipated, as mentioned in Subsection 3.6.4.2, that a COL applicant w[illegible] apply to the NRC for approval of LBB q[illegible]cation of selected piping by submitting a technical justification report. The approved piping, referred to in this SSAR as the LBB-qualified piping, will be excluded from pipe breaks, which are required to be postulated by Subsection 3.6.1 and 3.6.2, for design against their potential dynamic effects. However, such piping are included in postulation of pipe cracks for their effects as described in Subsections 3.6.1.3.1, 3.6.2.1.5 and 3.6.2.1.6.2. It is emphasized that an LBB qualification submittal is not a mandatory requirement; a COL applicant has an option to select from none to all technically feasible piping systems for the benefits of the LBB approach. The decision may be made based upon a cost-benefit evaluation (Reference 6).

3.6.1 Postulated Piping Failures In Fluid Systems Inside and Outside of Containment

This subsection sets forth the design bases, description, and safety evaluation for determining the effects of postulated piping failures in fluid systems both inside and outside the containment, and for including necessary protective measures.

3.6.1.1 Design Bases

3.6.1.1.1 Criteria

Pipe break event protection conforms to 10CFR50 Appendix A, General Design Criterion 4, *Environmental and Missile Design Bases*. The design bases for this protection is in compliance with NRC Branch Technical Positions (BTP) ASB 3-1 and MEB 3-1 included in Subsections 3.6.1 and 3.6.2, respectively, of NUREG-0800 (Standard Review Plan).

MEB 3-1 describes an acceptable basis for selecting the design locations and orientations of postulated breaks and cracks in fluid systems piping. Standard Review Plan Sections 3.6.1 and 3.6.2 describe acceptable measures that could be taken for protection against the breaks and cracks and for restraint against pipe whip that may result from breaks.

The design of the containment structure, component arrangement, pipe runs, pipe whip restraints and compartmentalization are done in consonance with the acknowledgment of protection against dynamic effects associated with a pipe break event. Analytically sized and positioned pipe whip restraints are engineered to preclude damage based on the pipe break evaluation.

3.6.1.1.2 Objectives

Protection against pipe break event dynamic effects is provided to fulfill the following objectives:

(1) Assure that the reactor can be shut down safely and maintained in a safe cold shutdown condition and that the consequences of the postulated piping failure are mitigated to acceptable limits without offsite power.

(2) Assure that containment integrity is maintained.

(3) Assure that the radiological doses of a postulated piping failure remain below the limits of 10CFR100.

3.6.1.1.3 Assumptions

The following assumptions are used to determine the protection requirements.

(1) Pipe break events may occur during normal plant conditions (i.e., reactor startup, operation at power, normal hot standby* or reactor cooldown to a cold shutdown conditions but excluding test modes).

210.23 (2) A pipe break event may occur simultaneously with a seismic event, however, a seismic event does not initiate a pipe break event. This applies to Seismic Category I and non-Seismic Category I piping.

(3) A single active component failure (SACF) is assumed in systems used to mitigate consequences of the postulated piping failure and to shut down the reactor, except as noted in item (4) below. A SACF is malfunction or loss of function of a component of electrical or fluid systems. The failure of an active component of a fluid system is considered to be a loss of component function as a result of mechanical, hydraulic, or electrical malfunction but not the loss of component structural integrity. The direct consequences of a SACF are considered to be a part of the single active failure. The single active component failure is assumed to occur in addition to the postulated piping failure and any direct consequences of the piping failure.

(4) Where the postulated piping failure is assumed to occur in one of two or more redundant trains of a dual-purpose moderate-energy essential system (i.e., one required to operate during normal plant conditions as well as to shut down the reactor and mitigate the consequences of the piping failure), single active failure of components in the other train or trains of that system only are not assumed, provided the system is designed to Seismic Category I standards, is powered from both offsite and onsite sources, and is constructed, operated, and inspected to quality assurance, testing and inservice inspection standards appropriate for nuclear safety-related systems. Residual heat removal system is an example of such a system.

(5) If a pipe break event involves a failure of non-Seismic Category I piping, the pipe break event must not result in failure of essential systems, components and equipment to shut down the reactor and mitigate the consequences of the pipe break event considering a SACF in accordance with items (3) and (4) above.

(6) If loss of offsite power is a direct consequence of the pipe break event (e.g., trip of the turbine-generator producing a power

* *Normal hot standby is a normally attained zero power plant operating state (as opposed to a hot standby initiated by a plant upset condition) where both feedwater and main condenser are available and in use.*

surge which in turn trips the main breaker), then a loss of offsite power occurs in a mechanistic time sequence with a SACF. Otherwise, offsite power is assumed available with a SACF.

(7) A whipping pipe is not capable of rupturing impacted pipes of equal or greater nominal pipe diameter, but may develop throughwall cracks in equal or larger nominal pipe sizes with thinner wall thickness.

(8) All available systems, including those actuated by operator actions, are available to mitigate the consequences of a postulated piping failure. In judging the availability of systems, account is taken of the postulated failure and its direct consequences such as unit trip and loss of offsite power, and of the assumed SACF and its direct consequences. The feasibility of carrying out operator actions are judged on the basis of ample time and adequate access to equipment being available for the proposed actions.

Although a pipe break event outside the containment may require a cold shutdown, up to eight hours in hot standby is allowed in order for plant personnel to assess the situation and make repairs.

(10) Pipe whip occurs in the plane defined by the piping geometry and causes movement in the direction of the jet reaction. If unrestrained, a whipping pipe with a constant energy source forms a plastic hinge and rotates about the nearest rigid restraint, anchor, or wall penetration. If unrestrained, a whipping pipe without a constant energy source (i.e., a break at a closed valve with only one side subject to pressure) is not capable of forming a plastic hinge and rotating provided its movement can be defined and evaluated.

(11) The fluid internal energy associated with the pipe break reaction can take into account any line restrictions (e.g., flow limiter) between the pressure source and break location and absence of energy reservoirs, as applicable.

3.6.1.1.4 Approach

To comply with the objectives previously described, the essential systems, components, and equipment are identified. The essential systems, components, and equipment, or portions thereof, are identified in Table 3.6-1 for piping failures postulated inside the containment and in Table 3.6-2 for outside the containment.

3.6.1.2 Description

The lines identified as high-energy per Subsection 3.6.2.1.1 are listed in Table 3.6-3 for inside the containment and in Table 3.6-4 for outside the containment. Moderate-energy piping defined in Subsection 3.6.2.1.2 is listed in Table 3.6-5 for outside the containment. Pressure response analyses are performed for the subcompartments containing high-energy piping. A detailed discussion of the line breaks selected, vent paths, room volumes, analytical methods, pressure results, etc., is provided in Section 6.2 for primary containment subcompartments.

The effects of pipe whip, jet impingement, spraying, and flooding on required function of essential systems, components, and equipment, or portions thereof, inside and outside the containment are considered.

In particular, there are no high-energy lines near the control room. As such, there are no effects upon the habitability of the control room by a piping failure in the control building or elsewhere either from pipe whip, jet impingement, or transport of steam. Further discussion on control room habitability systems is provided in Section 6.4.

3.6.1.3 Safety Evaluation

3.6.1.3.1 General

An analysis of pipe break events is performed to identify those essential systems, components, and equipment that provide protective actions required to mitigate, to acceptable limits, the consequences of the pipe break event.

Pipe break events involving high-energy fluid systems are evaluated for the effects of pipe whip, jet impingement, flooding, room pressurization, and other environmental effects such as temperature. Pipe break events involving moderate-energy fluid systems are evaluated for wetting from spray, flooding, and other environmental effects.

By means of the design features such as separation, barriers, and pipe whip restraints, a discussion of which follows, adequate protection is provided against the effects of pipe break events for essential items to an extent that their ability to shut down the plant safely or mitigate the consequences of the postulated pipe failure would not be impaired.

3.6.1.3.2 Protection Methods

3.6.1.3.2.1 General

The direct effects associated with a particular postulated break or crack must be mechanistically consistent with the failure. Thus, actual pipe dimensions, piping layouts, material properties, and equipment arrangements are considered in defining the following specific measure for protection against actual pipe movement and other associated consequences of postulated failures.

(1) Protection against the dynamic effects of pipe failures is provided in the form of pipe whip restraints, equipment shields, and physical separation of piping, equipment, and instrumentation.

(2) The precise method chosen depends largely upon limitations placed on the designer such as accessibility, maintenance, and proximity to other pipes.

3.6.1.3.2.2 Separation

The plant arrangement provides physical separation to the extent practicable to maintain the independence of redundant essential systems (including their auxiliaries) in order to prevent the loss of safety function due to any single postulated event. Redundant trains (e.g., A and B trains) and divisions are located in separate compartments to the extent possible. Physical separation between redundant essential systems with their related auxiliary supporting features, therefore, is the basic protective measure incorporated in the design to protect against the dynamic effects of postulated pipe failures.

Due to the complexities of several divisions being adjacent to high-energy lines in the drywell and reactor building steam tunnel, specific break locations are determined in accordance with Subsection 3.6.2.1.4.3 for possible spatial separation. Care is taken to avoid concentrating essential equipment in the break exclusion zone allowed per Subsection 3.6.2.1.4.2. If spatial separation requirements (distance and/or arrangement to prevent damage) cannot be met based on the postulation of specific breaks, barriers, enclosures, shields, or restraints are provided. These methods of protection are discussed on Subsections 3.6.1.3.2.3 and 3.6.1.3.2.4.

For other areas where physical separation is not practical, the following high-energy line-separation analysis (HELSA) evaluation is done to determine which high-energy lines meet the spatial separation requirement and which lines require further protection:

(1) For the HELSA evaluation, no particular break points are identified. Cubicles or areas through which the high-energy lines pass are examined in total. Breaks are postulated at any point in the piping system.

(2) Essential systems, components, and equipment at a distance greater than thirty feet from any high energy piping are considered as meeting spatial separation requirements. No damage is assumed to occur due to jet impingement since the impingement force becomes negligible beyond 30 feet. Likewise, a 30-ft evaluation zone is established for pipe breaks to assure protection against potential damage from a whipping pipe. Assurance that 30 feet represents the maximum free length is made in the piping layout.

(3) Essential systems, components, and equipment at a distance less than 30 feet from any high-energy piping are evaluated to see if damage could occur to more than one essential division, preventing safe shutdown of the plant. If damage occurred to only one division of a redundant system, the

requirement for redundant separation is met. Other redundant divisions are available for safe shutdown of the plant and no further evaluation is performed.

(4) If damage could occur to more than one division of a redundant essential system within 30 ft of any high energy piping, other protection in the form of barriers, shields, or enclosures is used. These methods of protection are discussed in Subsection 3.6.1.3.2.3. Pipe whip restraints as discussed in Subsection 3.6.1.3.2.4 are used if protection from whipping pipe is not possible by barriers and shields.

3.6.1.3.2.3 Barriers, Shields, and Enclosures

Protection requirements are met through the protection afforded by the walls, floors, columns, abutments, and foundations in many cases. Where adequate protection is not already present due to spatial separation or existing plant features, additional barriers, deflectors, or shields are identified as necessary to meet the functional protection requirements.

Barriers or shields that are identified as necessity by the use of specific break locations in the drywell are designed for the specific loads associated with the particular break location.

The steam tunnel is made of reinforced concrete 2m thick. A steam tunnel subcompartment analysis was performed for the postulated rupture of a mainsteam line and for a feedwater line (see Subsection 6.2.3.3.1). The peak pressure from a mainsteam line break was found to be 11 psig. The peak pressure from a feedwater line break was found to be 3.9 psig. The steam tunnel is designed for the effects of an SSE coincident with high energy line break inside the steam tunnel. Under this conservative load combination, no failure in any portion of the steam tunnel was found to occur; therefore, a high energy line break inside the steam tunnel will not effect control room habitability.

The MSIVs and the feedwater isolation and check valves located inside the tunnel shall be designed for the effects of a line break. The details of how the MSIV and feedwater isolation and check valves functional capabilities are protected against the effects of these postulated pipe failures will be provided by the applicant referencing the ABWR design (see Subsection 3.6.4.1, item 4 and 6).

Barriers or shields that are identified as necessary by the HELSA evaluation (i.e., based on no specific break locations), are designed for worst-case loads. The closest high-energy pipe location and resultant loads are used to size the barriers.

3.6.1.3.2.4 Pipe Whip Restraints

Pipe whip restraints are used where pipe break protection requirements could not be satisfied using spatial separation, barriers, shields, or enclosures alone. Restraints are located based on the specific break locations determined in accordance with Subsections 3.6.2.1.4.3 and 3.6.2.1.4.4. After the restraints are located, the piping and essential systems are evaluated for jet impingement and pipe whip. For those cases where jet impingement damage could still occur, barriers, shields, or enclosures are utilized.

The design criteria for restraints is given in Subsection 3.6.2.3.3.

3.6.1.3.3 Specific Protection Measures

(1) Nonessential systems and system components are not required for the safe shutdown of the reactor, nor are they required for the limitation of the offsite release in the event of a pipe rupture. However, while none of this equipment is needed during or following a pipe break event, pipe whip protection is considered where a resulting failure of a nonessential system or component could initiate or escalate the pipe break event in an essential system or component, or in another nonessential system whose failure could affect an essential system.

(2) For high energy piping systems penetrating through the containment, isolation valves are located as close to the containment as possible.

(3) The pressure, water level, and flow sensor instrumentation for those essential systems,

which are required to function following a pipe rupture, are protected.

(4) High-energy fluid system pipe whip restraints and protective measures are designed so that a postulated break in one pipe could not, in turn, lead to a rupture of other nearby pipes or components if the secondary rupture could result in consequences that would be considered unacceptable for the initial postulated break.

(5) For any postulated pipe rupture, the structural integrity of the containment structure is maintained. In addition, for those postulated ruptures classified as a loss of reactor coolant, the design leak tightness of the containment fission product barrier is maintained.

(6) Safety/relief valves (SRV) and the reactor core isolation cooling (RCIC) system steamline are located and restrained so that a pipe failure would not prevent depressurization.

(7) Separation is provided to preserve the independence of the low-pressure flooder (LPFL) systems.

(8) Protection for the FMCRD scram insert lines is not required since the motor operation of the FMCRD can adequately insert the control rods even with a complete loss of insert lines (See Subsection 3.6.2.1.6.1).

(9) The escape of steam, water, combustible or corrosive fluids, gases, and heat in the event of a pipe rupture do not preclude:

(a) Accessibility to any areas required to cope with the postulated pipe rupture;

(b) Habitability of the control room; or

(c) The ability of essential instrumentation, electric power supplies, components, and controls to perform their safety-related function.

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Information concerning break and crack location criteria and methods of analysis for dynamic effects is presented in this Subsection. The location criteria and methods of analysis are needed to evaluate the dynamic effects associated with postulated breaks and cracks in high- and moderate-energy fluid system piping inside and outside of primary containment. This information provides the basis for the requirements for the protection of essential structures, systems, and components defined in introduction of Section 3.6.

3.6.2.1 Criteria Used to Define Break and Crack Location and Configuration

The following subsections establish the criteria for the location and configuration of postulated breaks and cracks.

3.6.2.1.1 Definition of High-Energy Fluid Systems

High-energy fluid systems are defined to be those systems or portions of systems that, during normal plant conditions (as defined in Subsection 3.6.1.1.3(1)),are either in operation or are maintained pressurized under conditions where either or both of the following are met:

(1) maximum operating temperature exceeds 200°F, or

(2) maximum operating pressure exceeds 275 psig.

3.6.2.1.2 Definition of Moderate-Energy Fluid Systems.

Moderate-energy fluid systems are defined to be those systems or portions of systems that, during normal plant conditions (as defined in Subsection 3.6.1.1.3.(1)), are either in operation or are maintained pressurized (above atmospheric pressure) under conditions where both of the following are met:

(1) maximum operating temperature is 200°F or less, and

(2) maximum operating pressure is 275 psig or less.

Piping systems are classified as moderate-energy systems when they operate as high-energy piping for only short operational periods in performing their system function but, for the major operational period, qualify as moderate-energy fluid systems. An operational period is considered short if the total fraction of time that the system operates within the pressure-temperature conditions specified for high-energy fluid systems is less than two percent of the total time that the system operates as a moderate-energy fluid system.

3.6.2.1.3 Postulated Pipe Breaks and Cracks

A postulated pipe break is defined as a sudden gross failure of the pressure boundary either in the form of a complete circumferential severance (guillotine break) or a sudden longitudinal split without pipe severance, and is postulated for high-energy fluid systems only. For moderate-energy fluid system, pipe failures are limited to postulation of cracks in piping and branch runs. These cracks affect the surrounding environmental conditions only and do not result in whipping of the cracked pipe. High-energy fluid systems are also postulated to have cracks for conservative environmental conditions in a confined area where high- and moderate-energy fluid systems are located.

The following high-energy piping systems (or portions of systems) are considered as potential candidates for a postulated pipe break during normal plant conditions and are analyzed for potential damage resulting from dynamic effects:

(1) All piping which is part of the reactor coolant pressure boundary and subject to reactor pressure continuously during station operation;

(2) All piping which is beyond the second isolation valve but subject to reactor pressure continuously during station operation; and

(3) All other piping systems or portions of piping systems considered high-energy systems.

Portions of piping systems that are isolated from the source of the high-energy fluid during normal plant conditions are exempted from consideration of postulated pipe breaks. This includes portions of piping systems beyond a normally closed valve. Pump and valve bodies are also exempted from consideration of pipe break because of their greater wall thickness.

3.6.2.1.4 Locations of Postulated Pipe Breaks

Postulated pipe break locations are selected as follows:

3.6.2.1.4.1 Piping Meeting Separation Requirements

Based on the HELSA evaluation described in Subsection 3.6.1.3.2.2, the high-energy lines which meet the spatial separation requirements are generally not identified with particular break points. Breaks are postulated at all possible points in such high-energy piping systems. However, in some systems break points are particularly specified per the following subsections if special protection devices such as barriers or restraints are provided.

3.6.2.1.4.2 Piping in Containment Penetration Areas

No pipe breaks or cracks are postulated in those portions of piping from containment wall to and including the inboard or outboard isolation valves which meet the following requirement in addition to the requirement of the ASME Code, Section III, Subarticle NE-1120:

(1) The following design stress and fatigue limits are not exceeded:

For ASME Code, Section III, Class 1 Piping

(a) The maximum stress range between any two loads sets (including the zero load set) does not exceed 2.4 S_m, and is calculated* by Eq. (10) in NB-3653, ASME Code, Section III.

If the calculated maximum stress range of Eq. (10) exceeds 2.4 S_m, the stress ranges calculated by both Eq. (12) and Eq. (13) in Paragraph NB-3653 meet the limit of 2.4 S_m.

(b) The cumulative usage factor is less than 0.1

(c) The maximum stress, as calculated by Eq. (9) in NB-3652 under the loadings resulting from a postulated piping failure beyond these portions of piping does not exceed the lesser of 2.25 S_m and 1.8 S_y except that following a failure outside containment, the pipe between the outboard isolation valve and

* *For those loads and conditions in which Level A and Level B stress limits have been specified in the Design Specification.*

the first restraint may be permitted higher stresses provided a plastic hinge is not formed and operability of the valves with such stresses is assured in accordance with the requirement specified in Section 3.9.3. Primary loads include those which are deflection limited by whip restraints.

For ASME Code, Section III, Class 2 Piping

(d) The maximum stress as calculated by the sum of Eqs. (9) and (10) in Paragraph NC-3652, ASME Code, Section III, considering those loads and conditions thereof for which level A and level B stress limits are specified in the system's Design Specification (i.e., sustained loads, occasional loads, and thermal expansion) including an OBE event does not exceed $0.8(1.8\ S_h + S_A)$. The S_h and S_A are allowable stresses at maximum (hot) temperature and allowable stress range for thermal expansion, respectively, as defined in Article NC-3600 of the ASME Code, Section III.

(e) The maximum stress, as calculated by Eq. (9) in NC-3653 under the loadings resulting from a postulated piping failure of fluid system piping beyond these portions of piping does not exceed the lesser of $2.25\ S_h$ and $1.8\ S_y$.

Primary loads include those which are deflection limited by whip restraints. The exceptions permitted in (c) above may also be applied provided that when the piping between the outboard isola- tion valve and the restraint is con- structed in accordance with the Power Piping Code ANSI B31.1, the piping is either of seamless construction with full radiography of all circumferential welds, or all longitudinal and circum- ferential welds are fully radiographed.

(2) Welded attachments, for pipe supports or other purposes, to these portions of piping are avoided except where detailed stress analyses, or tests, are performed to demonstrate compliance with the limits of item (1).

(3) The number of circumferential and longi- tudinal piping welds and branch connections are minimized. Where penetration sleeves are used, the enclosed portion of fluid system piping is seamless construction and without circumferential welds unless specific access provisions are made to permit inservice volumetric examination of longitudinal and circumferential welds.

(4) The length of these portions of piping are reduced to the minimum length practical.

(5) The design of pipe anchors or restraints (e.g., connections to containment penetrations and pipe whip restraints) do not require welding directly to the outer surface of the piping (e.g., flued integ- rally forged pipe fittings may be used) except where such welds are 100 percent volumetrically examinable in service and a detailed stress analysis is performed to demonstrate compliance with the limits of item (1).

(6) Sleeves provided for those portions of piping in the containment penetration areas are constructed in accordance with the rules of Class MC, Subsection NE of the ASME Code, Section III, where the sleeve is part of the containment boundary. In addition, the entire sleeve assembly is designed to meet the following requirements and tests:

(a) The design pressure and temperature are not less than the maximum operating pressure and temperature of the enclosed pipe under normal plant conditions.

(b) The Level C stress limits in NE-3220, ASME Code, Section III, are not exceeded under the loadings associated with containment design pressure and temperature in combination with the safe shutdown earthquake.

(c) The assemblies are subjected to a single pressure test at a pressure not less than its design pressure.

(d) The assemblies do not prevent the access required to conduct the inservice examination specified in item).

(7) A 100% volumetric inservice examination of all pipe welds would be conducted during each inspection interval as defined in IWA-2400, ASME Code, Section XI.

3.6.2.1.4.3 ASME Code Section III Class 1 Piping In Areas Other Than Containment Penetration

With the exception of those portions of piping identified in Subsection 3.6.2.1.4.2, breaks in ASME Code, Section III, Class 1 piping are postulated at the following locations in each piping and branch run:

(a) At terminal ends*

(b) At intermediate locations where the maximum stress range (see Subsection 3.6.2.1.4.2, Paragraph (1)(a)) as calculated by Eq. (10) in NB-3653, ASME Code, Section III.

If the calculated maximum stress range of Eq.(10) exceeds the stress range calculated by both Eq.(12) and Eq.(13) in Paragraph NB-3653 should meet the limit of 2.4 Sm.

(c) At intermediate locations where the cumulative usage factor exceeds 0.1.

* *Extremities of piping runs that connect to structures, components (e.g., vessels, pumps, valves), or pipe anchors that act as rigid constraints to piping motion and thermal expansion. A branch connection to a main piping run is a terminal end of the branch run, except where the branch run is classified as part of a main run in the stress analysis and is shown to have a significant effect on the main run behavior. In piping runs which are maintained pressurized during normal plant conditions for only a portion of the run (i.e., up to the first normally closed valve) a terminal end of such runs is the piping connection to this closed valve.*

As a result of piping re-analysis due to differences between the design configuration and the as-built configuration, the highest stress or cumulative usage factor locations may be shifted; however, the initially determined intermediate break locations need not be changed unless one of the following conditions exists:

(i) The dynamic effects from the new (as-built) intermediate break locations are not mitigated by the original pipe whip restraints and jet shields.

(ii) A change is required in pipe parameters such as major differences in pipe size, wall thickness, and routing.

3.6.2.1.4.4 ASME Code Section III Class 2 and 3 Piping in Areas Other Than Containment Penetration

With the exceptions of those portions of piping identified in Subsection 3.6.2.1.4.2, breaks in ASME Codes, Section III, Class 2 and 3 piping are postulated at the following locations in those portions of each piping and branch run:

(a) At terminal ends (see Subsection 3.6.2.1.4.3, Paragraph (a))

(b) At intermediate locations selected by one of the following criteria:

(i) At each pipe fitting (e.g., elbow, tee, cross, flange, and nonstandard fitting), welded attachment, and valve. Where the piping contains no fittings, welded attachments, or valves, at one location at each extreme of the piping run adjacent to the protective structure.

(ii) At each location where stresses calculated (see Subsection 3.6.2.1.4.2, Paragraph (1)(d)) by the sum of Eqs. (9) and (10) in NC/ND-3653, ASME Code, Section III, exceed 0.8 times the sum of the stress limits given in NC/ND-3653.

As a result of piping re-analysis due to differences between the design configuration and the as-built configuration, the highest stress locations may be shifted; however, the initially determined intermediate break locations may be used unless a redesign of the piping resulting in a change in the pipe parameters (diameter, wall thickness, routing) is required, or the dynamic effects from the new (as-built) intermediate break location are not mitigated by the original pipe whip restraints and jet shields.

3.6.2.1.4.5 Non-ASME Class Piping

Breaks in seismically analyzed non-ASME Class (not ASME Class 1, 2 or 3) piping are postulated according to the same requirements for ASME Class 2 and 3 piping above. Separation and interaction requirements between Seismically analyzed and non-seismically analyzed piping are met as described in Subsection 3.7.3.13.

3.6.2.1.4.6 Separating Structure With High-Energy Lines

If a structure separates a high energy line from an essential component, the separating structure is designed to withstand the consequences of the pipe break in the high-energy line at locations that the aforementioned criteria require to be postulated. However, as noted in Subsection 3.6.1.3.2.3, some structures that are identified as necessary by the HELSA evaluation (i.e., based on no specific break locations), are designed for worst-case loads.

3.6.2.1.5 Locations of Postulated Pipe Cracks

Postulated pipe crack locations are selected as follows:

3.6.2.1.5.1 Piping Meeting Separation Requirements

Based on the HELSA evaluation described in Subsection 3.6.1.3.2.2, the high- or moderate-energy lines which meet the separation requirements are not identified with particular crack locations. Cracks are postulated at all possible points that are necessary to demonstrate adequacy of separation or other means of protections provided for essential structures, systems and components.

3.6.2.1.5.2 High-Energy Piping

With the exception of those portions of piping identified in Subsection 3.6.2.1.4.2, leakage cracks are postulated for the most severe environmental effects as follows:

(1) For ASME Code, Section III Class 1 piping, at axial locations where the calculated stress range (see Subsection 3.6.2.1.4.2, Paragraph (1)(a)) by Eq. (10) and either Eq. (12) or Eq. (13) in NB-3653 exceeds 1.2 S_m.

(2) For ASME Code, Section III Class 2 and 3 or non-ASME class piping, at axial locations where the calculated stress (see Subsection 3.6.2.1.4.4, Paragraph (b)(ii)) by the sum of Eqs. (9) and (10) in NC/ND-3653 exceeds 0.4 times the sum of the stress limits given in NC/ND-3653.

(3) Non-ASME class piping which has not been evaluated to obtain stress information have leakage cracks postulated at axial locations that produce the most severe environmental effects.

3.6.2.1.5.3 Moderate-Energy Piping

3.6.2.1.5.3.1 Piping In Containment Penetration Areas

Leakage cracks are not postulated in those portions of piping from containment wall to and including the inboard or outboard isolation valves provided they meet the requirements of the ASME Code, Section III, NE-1120, and the stresses calculated (See Subsection 3.6.2.1.4.4, Paragraph (b)(ii)) by the sum of Eqs. (9) and (10) in ASME Code, Section III, NC-3653 do not exceed 0.4 times the sum of the stress limits given in NC-3653.

3.6.2.1.5.3.2 Piping In Areas Other Than Containment Penetration

(1) Leakage cracks are postulated in piping located adjacent to essential structures, systems or components, except:

   (a) Where exempted by Subsections 3.6.2.1.5.3.1 and 3.6.2.1.5.4,

   (b) For ASME Code, Section III, Class 1 piping the stress range calculated (see Subsection 3.6.2.1.4.2, Paragraph (1) (a)) by Eq. (10) and either Eq. (12) or Eq. (13) in NB-3653 is less than 1.2 S_m.

(c) For ASME Code, Section III, Class 2 or 3 and non-ASME class piping, the stresses calculated (see Subsection 3.6.2.1.4.4, Paragraph (b)(ii)) by the sum of Eqs. (9) and (10) in NC/ND-3653 are less than 0.4 times the sum of the stress limits given in NC/ND-3653.

(2) Leakage cracks, unless the piping system is exempted by item (1) above, are postulated at axial and circumferential locations that result in the most severe environmental consequences.

(3) Leakage cracks are postulated in fluid system piping designed to nonseismic standards as necessary to meet the environmental protection requirements of Subsection 3.6.1.1.3.

3.6.2.1.5.4 Moderate-Energy Piping in Proximity to High-Energy Piping

Moderate-energy fluid system piping or portions thereof that are located within a compartment of confined area involving considerations for a postulated break in high-energy fluid system piping are acceptable without postulation of throughwall leakage cracks except where a postulated leakage crack in the moderate-energy fluid system piping results in more severe environmental conditions than the break in the proximate high-energy fluid system piping, in which case the provisions of Subsection 3.6.2.1.5.3 are applied.

3.6.2.1.6 Types of Breaks and Cracks to be Postulated

3.6.2.1.6.1 Pipe Breaks

The following types of breaks are postulated in high-energy fluid system piping at the locations identified by the criteria specified in Subsection 3.6.2.1.4.

(1) No breaks are postulated in piping having a nominal diameter less than or equal to one inch. Instrument lines one inch and less nominal pipe or tubing size meet the provision of regulatory Guide 1.11 (See Table 3.2-1). Additionally, the 1-1/4-inch hydraulic control unit fast scram lines do not require special protection measure because of the following reasons:

(a) The piping to the control rod drives from the hydraulic control units (HCUs) are located in the containment under reactor vessel, and in the reactor building away from other safety-related equipment; therefore should a line fail, it would not affect any safety-related equipment but only impact on other HCU lines. As discussed in Subsection 3.6.1.1.3, Paragraph (7), a whipping pipe will only rupture an impacted pipe of smaller nominal pipe size or cause a through wall crack in the same nominal pipe size but with thinner wall thickness.

(b) The total amount of energy contained in the 1-1/4" piping between normally closed scram insert valve on the HCU module and the ball-check valve in the control rod housing is small. In the event of a rupture of this line, the ball-check valve will close to prevent reactor vessel flow out of the break.

(c) Even if a number of the HCU lines ruptured, the control rod insertion function would not be impaired since the electrical motor of the fine motion control drive would drive in the control rods.

(2) Longitudinal breaks are postulated only in piping having a nominal diameter equal to or greater than four inches.

(3) Circumferential breaks are only assumed at all terminal ends.

(4) At each of the intermediate postulated break locations identified to exceed the stress and usage factor limits of the criteria in Subsections 3.6.2.1.4.3 and 3.6.2.1.4.4, considerations is given to the occurrence of either a longitudinal or circumferential break. Examination of the state of stress in the vicinity of the postulated break location is used to identify the most

probably type of break. If the maximum stress range in the longitudinal direction is greater than 1.5 times the maximum stress range in the circumferential direction, only the circumferential break is postulated. Conversely, if the maximum stress range in the circumferential direction is greater than 1.5 times the stress range in the longitudinal direction, only the longitudinal break is postulated. If no significant difference between the circumferential and longitudinal stresses is determined, then both types of breaks are considered.

(5) Where breaks are postulated to occur at each intermediate pipe fitting, weld attachment, or valve without the benefit of stress calculations, only circumferential breaks are postulated.

(6) For both longitudinal and circumferential breaks, after assessing the contribution of upstream piping flexibility, pipe whip is assumed to occur in the plane defined by the piping geometry and configuration for circumferential breaks and out of plane for longitudinal breaks and to cause piping movement in the direction of the jet reactions. Structural members, piping restraints, or piping stiffness as demonstrated by inelastic limit analysis are considered in determining the piping movement limit (alternatively, circumferential breaks are assumed to result in pipe severance and separation amounting to at least a one-diameter lateral displacement of the ruptured piping sections).

(7) For a circumferential break, the dynamic force of the jet discharged at the break location is based upon the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Limited pipe displacement at the break location, line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs are used, as applicable, in the reduction of the jet discharge.

(8) Longitudinal breaks in the form of axial split without pipe severance are postulated in the center of the piping at two diametrically opposed points (but not concurrently) located so that the reaction force is perpendicular to the plane of the piping configuration and produces out-of-plane bending. Alternatively, a single split is assumed at the section of highest tensile stress as determined by detailed stress analysis (e.g., finite element analysis).

(9) The dynamic force of the fluid jet discharge is based on a circular or elliptical (2D x 1/2D) break area equal to the effective cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure modified by an analytically or experimentally determined thrust coefficient as determined for a circumferential break at the same location. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account as applicable in the reduction of jet discharge.

3.6.2.1.6.2 Pipe Cracks

The following criteria are used to postulate throughwall leakage cracks in high- or moderate-energy fluid systems or portions of systems.

(1) Cracks are postulated in moderate-energy fluid system piping and branch runs exceeding a nominal pipe size of one inch.

(2) At axial locations determined per Subsection 3.6.2.1.5, the postulated cracks are oriented circumferentially to result in the most severe environmental consequences.

(3) Crack openings are assumed as a circular orifice of area equal to that of a rectangle having dimensions one-half-pipe-diameter in length and one-half-pipe-wall thickness in width.

(4) The flow from the crack opening is assumed to result in an environment that wets all unprotected components within the compartment, with consequent flooding in the compartment and communicating compartments, based on a conservatively estimated time period to effect corrective actions.

3.6.2.2 Analytic Methods to Define Blowdown Forcing Functions and Response Models.

3.6.2.2.1 Analytic Methods to Define Blowdown Forcing Functions.

The rupture of a pressurized pipe causes the flow characteristics of the system to change creating reaction forces which can dynamically excite the piping system. The reaction forces are a function of time and space and depend upon fluid state within the pipe prior to rupture, break flow area, frictional losses, plant system characteristics, piping system, and other factors. The methods used to calculate the reaction forces for various piping systems are presented in the following subsections.

The criteria that are used for calculation of fluid blowdown forcing functions include:

(1) Circumferential breaks are assumed to result in pipe severance and separation amounting to at least a one-diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints, structural members, or piping stiffness as may be demonstrated by inelastic limit analysis (e.g., a plastic hinge in the piping is not developed under loading).

(2) The dynamic force of the jet discharge at the break location is based on the cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by analytically- or experimentally-determined thrust coefficient. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs are taken into accounts, as applicable, in the reduction of jet discharge.

(3) All breaks are assumed to attain full size within one millisecond after break initiation.

The forcing functions due to the postulated pipe breaks near the reactor ot at a branch connection are calculated by the solution of one-dimensional, compressible unsteady steam flow in the gas system. The numerical analysis is performed by the method of characteristics. The flow starts with steady flow from the RPV to the turbine. A pipe break causes the steam flow to reverse its direction and to flow from the turbine to the break location. The pipe segment force time histories are determined by calculating the momentum change in the pipe segments of a closed system. The broken pipe segment force time history is calculated in accordance with Appendix B of ANSI/ANS-58.2.

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(5) Piping within the broken loop is no longer considered part of the RCPB. Plastic deformation in the pipe is considered as a potential energy absorber. Limits of strain are imposed which are similar to strain levels allowed in restraint plastic members. Piping systems are designed so that plastic instability does not occur in the pipe at the design dynamic and static loads unless damage studies are performed which show the consequences do not result in direct damage to any essential system or component.

(6) Components such as vessel safe ends and valves which are attached to the broken piping system, do not serve a safety-related function, or failure of which would not further escalate the consequences of the accident are not designed to meet ASME Code-imposed limits for essential components under faulted loading. However, if these components are required for safe shutdown or serve to protect the structural integrity of an essential component, limits to meet the Code requirements for faulted conditions and limits to ensure required operability will be met.

(7) The piping stresses in the containment penetration areas due to loads resulting from a postulated piping failure can not exceed the limits specified in Subsection 3.6.2.1.4.2(1)(c).

3.6.2.2.2 Pipe Whip Dynamic Response Analyses

The prediction of time-dependent and steady-thrust reaction loads caused by blowdown of subcooled, saturated, and two-phase fluid from ruptured pipe is used in design and evaluation of dynamic effects of pipe breaks. A discussion of the analytical methods employed to compute these blowdown loads is given in Subsection 3.6.2.2.1. Following is a discussion of analytical methods used to account for this loading.

The criteria used for performing the pipe whip dynamic response analyses include:

(1) A pipe whip analysis is performed for each postulated pipe break. However, a given analysis can be used for more than one postulated break location if the blowdown forcing function, piping and restraint system geometry, and piping and restraint system properties are conservative for other break locations.

(2) The analysis includes the dynamic response of the pipe in question and the pipe whip restraints which transmit loading to the support structures.

(3) The analytical model adequately represents the mass/inertia and stiffness properties of the system.

(4) Pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration and to cause pipe movement in the direction of the jet reaction.

An analysis for pipewhip restraint selection PDA computer program; and a pipe break modeling program ANSYS are performed as described in Appendix 3D, which predicts the response of a pipe subjected to the thrust force occurring after a pipe break. The program treats the situation in terms of generic pipe break configuration which involves a straight, uniform pipe fixed at one end and subjected to a time-dependent thrust force at the other end. A typical restraint used to reduce the resulting deformation is also included at a location between the two ends. Nonlinear and time-independent stress- strain relationships are used to model the pipe and the restraint. Using a plastic-hinge concept, bending of the pipe is assumed to occur only at the fixed end and at the location supported by the restraint.

Effects of pipe shear deflection are considered negligible. The pipe-bending moment-deflection (or rotation) relation used for these locations is obtained from a static nonlinear cantilever-beam analysis. Using the moment-rotation relation, nonlinear equations of motion of the pipe are formulated using energy considerations and the equations are numerically integrated in small time steps to yield time-history of the pipe motion.

The piping stresses in the containment penetration areas are calculated by the ANSYS computer program, a program as described in Appendix 3D. The program is used to perform the non-linear analysis of a piping system for time varying displacements and forces due to postulated pipe breaks.

3.6.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

3.6.2.3.1 Jet Impingement Analyses and Effects on Safety-Related Components

The methods used to evaluate the jet effects resulting from the postulated breaks of high-energy piping are described in Appendices C and D of ANSI/ANS 58.2 and presented in this subsection.

The criteria used for evaluating the effects of fluid jets on essential structures, systems, and components are as follows:

(1) Essential structures, systems, and components are not impaired so as to preclude essential functions. For any given postulated pipe break and consequent jet, those essential structures, systems, and components need to safely shut down the plant are identified.

(2) Essential structures, systems, and components which are not necessary to safely shut down the plant for a given break are not protected from the consequences of the fluid jet.

(3) Safe shutdown of the plant due to postulated pipe ruptures within the RCPB is not aggravated by sequential failures of safety-related piping and the required emergency cooling system performance is maintained.

(4) Offsite dose limits specified in 10CFR100 are complied with.

(5) Postulated breaks resulting in jet impingement loads are assumed to occur in high-energy lines at full (102%) power operation of the plant.

(6) Throughwall leakage cracks are postulated in moderate energy lines and are assumed to

result in wetting and spraying of essential structures, systems, and components.

(7) Reflected jets are considered only when there is an obvious reflecting surface (such as a flat plate) which directs the jet onto an essential equipment. Only the first reflection is considered in evaluating potential targets.

(8) Potential targets in the jet path are considered at the calculated final position of the broken end of the ruptured pipe. This selection of potential targets is considered adequate due to the large number of breaks analyzed and the protection provided from the effects of these postulated breaks.

The analytical methods used to determine which targets will be impinged upon by a fluid jet and the corresponding jet impingement load include:

(1) The direction of the fluid jet is based on the arrested position of the pipe during steady-state blowdown.

(2) The impinging jet proceeds along a straight path.

(3) The total impingement force acting on any cross sectional area of the jet is time and distance invariant with a total magnitude equivalent to the steady-state fluid blowdown force given in Subsection 3.6.2.2.1 and with jet characteristics shown in Figure 3.6-3.

(4) The jet impingement force is uniformly distributed across the cross-sectional area of the jet and only the portion intercepted by the target is considered.

(5) The break opening is assumed to be a circular orifice of cross-sectional flow area equal to the effective flow area of the break.

(6) The jet impingement force is equal to the steady-state value of the fluid blowdown force calculated by the methods described in Subsection 3.6.2.2.1.

(7) The distance of jet travel is divided into two or three regions. Region 1 (Figure 3.6-3) extends from the break to the asymptotic area. Within this region the discharging fluid flashes and undergoes expansion from the break area pressure to the atmospheric pressure. In Region 2 the jet expands further. For partial-separation circumferential breaks, the area increases as the jet expands. In Region 3 jet expands at a half angle of 10°. (Figures 3.6-3a and c.)

(8) The analytical model for estimating the asymptotic jet area for subcooled water and saturated water assumes a constant jet area. For fluids discharging from a break which are below the saturation temperature at the corresponding room pressure or have a pressure at the break area equal to the room pressure, the free expansion does not occur.

(9) The distance downstream from the break where the asymptotic area is reached (Region 2) is calculted for circumferential and longitudinal breaks.

(10) Both longitudinal and fully separated circumferential breaks are treated similarly. The value of fL/D used in the blowdown calculation is used for jet impingement also.

(11) Circumferential breaks with partial (i.e., $h < D/2$) separation between the two ends of the broken pipe not significantly offset (i.e., no more than one pipe wall thickness lateral displacement) are more difficult to

quantify. For these cases, the following assumptions are made.

(a) The jet is uniformly distributed around the periphery.

(b) The jet cross section at any cut through the pipe axis has the configuration depicted in Figure 3.6-3b and the jet regions are as therein delineated.

(c) The jet force F_j = total blowdown F.

(d) The pressure at any point intersected by the jet is:

$$P_j = \frac{F_s}{A_R}$$

where

A_R = the total 360° area of the jet at a radius equal to the distance from the pipe centerline to the target.

(e) The pressure of the jet is then multiplied by the area of the target submerged within the jet.

(12) Target loads are determined using the following procedures.

(a) For both the fully separated circumferential break and the longitudinal break, the jet is studied by determining target locations vs. asymptomatic distance and applying ANSI/ANS-58.2, Appendices C and D.

(b) For circumferential break limited separation, the jet is analyzed by using different equations of ANSI/ANS 58.2, Appendices C and D and determing respective target and asymptomatic locations

c) After determination of the total area of the jet at the target, the jet pressure is calculated by:

$$P_1 = \frac{F_j}{A_x}$$

where

P_1 = incident pressure

A_x = area of the expanded jet at the target intersection.

If the effective target area (A_{te}) is less than expanded jet area ($A_{te} \leq A_x$), the target is fully submerged in the jet and the impingement load is equal to (P_1) (A_{te}). If the effective target area is greater than expanded jet area ($A_{te} > A_x$), the target intercepts the entire jet and the impingement load is equal to (P_1) (A_x) = F_j. The effective target area (A_{te}) for various geometries follows:

(1) Flat surface - For a case where a target with physical area A_t is oriented at angle d with respect to the jet axis and with no flow reversal, the effective target area A_{te} is:

$$A_{te} = (A_t)\ (\sin\phi).$$

(2) Pipe Surface - As the jet hits the convex surface of the pipe, its forward momentum is decreased rather than stopped; therefore, the jet impingement load on the impacted area is expected to be reduced. For conservatism, no credit is taken for this reduction and the pipe is assumed to be impacted with the full impingement load. However, where shape factors are justifiable, they may be used. The effective target area A_{te} is:

$$A_{te} = (D_A)\,(D)$$

where

D_A = diameter of the jet at the target interface, and

D = pipe OD of target pipe for a fully submerged pipe.

When the target (pipe) is larger than the area of the jet, the effective target area equals the expanded jet area

$$A_{te} = A_x$$

(3) For all cases, the jet area (A_x) is assumed to be uniform and the load is uniformly distributed on the impinged target area A_{te}.

3.6.2.3.2 Pipe Whip Effects on Essential Components

This subsection provides the criteria and methods used to evaluate the effects of pipe displacements on essential structures, systems, and components following a postulated pipe rupture.

Pipe whip (displacement) effects on essential structures, systems, and components can be placed in two categories: (1) pipe displacement effects on components (nozzles, valves, tees, etc.) which are in the same piping run that the break occurs in; and (2) pipe whip or controlled displacements onto external components such as building structure, other piping systems, cable trays, and conduits, etc.

3.6.2.3.2.1 Pipe Displacement Effects on Components in the Same Piping Run

The criteria for determining the effects of pipe displacements on inline components are as follows:

(1) Components such as vessel safe ends and valves which are attached to the broken piping system and do not serve a safety function or failure of which would not further escalate the consequences of the accident need not be designed to meet ASME

Code Section III-imposed limits for essential components under faulted loading.

(2) If these components are required for safe shutdown or serve to protect the structural integrity of an essential component, limits to meet the ASME Code requirements for faulted conditions and limits to ensure required operability are met.

The methods used to calculate the pipe whip loads on piping components in the same run as the postulated break are described in Section 3.6.2.2.2.

3.6.2.3.2.2 Pipe Displacement Effects on Essential Structures, Other Systems, and Components

The criteria and methods used to calculate the effects of pipe whip on external components consists of the following:

(1) The effects on essential structures and barriers are evaluated in accordance with the barrier design procedures given in Subsection 3.5.3

(2) If the whipping pipe impacts a pipe of equal or greater nominal pipe diameter and equal or greater wall thickness, the whipping pipe does not rupture the impacted pipe. Otherwise, the impacted pipe is assumed to be ruptured.

(3) If the whipping pipe impacts other components (valve actuators, cable trays, conduits, etc.), it is assumed that the impacted component is unavailable to mitigate the consequences of the pipe break event.

(4) Damage of unrestrained whipping pipe on essential structures, components, and systems other than the ruptured one is prevented by either separating high energy systems from the essential systems or providing pipe whip restraints.

3.6.2.3.3 Loading Combinations and Design Criteria for Pipe Whip Restraint

Pipe whip restraints, as differentiated from piping supports, are designed to function and carry load for an extremely low-probability gross failure in a piping system carrying high- energy fluid. In the ABWR plant, the piping integrity does not depend on the pipe whip restraints for any piping design loading combination including earthquake but shall remain functional following an earthquake up to and including the SSE (See Subsection 3.2.1). When the piping integrity is lost because of a postulated break, the pipe whip restraint acts to limit the movement of the broken pipe to an acceptable distance. The pipe whip restraints (i.e., those devices which serve only to control the movement of a ruptured pipe following gross failure) will be subjected to once-in-a-lifetime loading. For the purpose of the pipe whip restraint design, the pipe break is considered to be a faulted condition (See Subsection 3.9.3.1.1.4) and the structure to which the restraint is attached is also analyzed and designed accordingly. The pipe whip restraints are non-ASME Code components; however, the ASME Code requirements may be used in the design selectively to assure its safety-related function if ever needed. Other methods, i.e. testing, with reliable data base for design and sizing of pipe whip restraints can also be used.

The pipe whip restraints utilize energy absorbing U-rods to attenuate the kinetic energy of a ruptured pipe. A typical pipe whip restraint is shown in Figure 3.6-6. The principal feature of these restraints is that they are installed with several inches of annular clearance between them and the process pipe. This allows for installation of normal piping insulation and for unrestricted pipe thermal movements during plant operation. Select critical locations inside primary containment are also monitored during hot functional testing to provide verification of adequate clearances prior to plant operation. The specific design objectives for the restraints are:

(1) The restraints shall in no way increase the reactor coolant pressure boundary stresses by their presence during any normal mode of reactor operation or condition;

(2) The restraint system shall function to stop the movement of a pipe failure (gross loss of piping integrity) without allowing damage to critical components or missile development; and

(3) The restraints should provide minimum hindrance to inservice inspection of the process piping.

For the purpose of design, the pipe whip restraints are designed for the following dynamic loads:

(1) Blowdown thrust of the pipe section that impacts the restraint;

(2) Dynamic inertia loads of the moving pipe section which is accelerated by the blowdown thrust and subsequent impact on the restraint;

(3) Design characteristics of the pipe whip restraints are included and verified by the pipe whip dynamic analysis described in Subsection 3.6.2.2.2; and

(4) Since the pipe whip restraints are not contacted during normal plant operation, the postulated pipe rupture event is the only design loading condition.

Strain rate effects and other material property variations have been considered in the design of the pipe whip restraints. The material properties utilized in the design have included one or more of the following methods:

(1) Code minimum or specification yield and ultimate strength values for the affected components and structures are used for both the dynamic and steady-state events;

(2) Not more than a 10% increase in minimum code or specification strength values is used when designing components or structures for the dynamic event, and code minimum or specification yield and ultimate strength values are used for the steady-state loads;

(3) Representative or actual test data values are used in the design of components and structures including justifiably elevated strain rate-affected stress limits in excess of 10%; or

(4) Representative or actual test data are used for any affected component(s) and the minimum code or specification values are used for the structures for the dynamic and the steady-state events

3.6.2.4 Guard Pipe Assembly Design

The ABWR primary containment does not require guard pipes.

3.6.2.5 Material to be Supplied for the Operating License Review

See Subsection 3.6.4.1

3.6.3 Leak-Before-Break Evaluation Procedures

Per Regulatory Guide 1.70, Revision 3, November 1978, the safety analysis Section 3.6 has traditionally addressed the protection measures against dynamic effects associated with the non- mechanistic or postulated ruptures of piping. The dynamic effects are defined in introduction to Section 3.6. Three forms of piping failure (full flow area circumferential and longitudinal breaks, and throughwall leakage crack) are postulated in accordance with Subsection 3.6.2 and Branch Technical Position MEB 3-1 of NUREG - 0800 (Standard Review Plan) for their dynamic as well as environmental effects.

However, in accordance with the modified General Electric Criterion 4 (GDC-4), effective November 27, 1987, (Reference 1), the mechanistic leak-before-break (LBB) approach, justified by appropriate fracture mechanics techniques, is recognized as an acceptable procedure under certain conditions to exclude design against the dynamic effects from postulation of breaks in high energy piping. The LBB approach is not used to exclude postulation of cracks and associated effects as required in Subsection 3.6.2.1.5 and 3.6.2.1.6.2. It is anticipated, as mentioned in Subsection 3.6.4.2, that a COL applicant will apply to the NRC for approval of LBB qualific[illegible] of selected piping. These approved pi[illegible], referred to in this SSAR as the LBB-accordance with Subsections 3.6.2.1.5 and 3.6.2.1.6.2.

The LBB approach is not applicable to piping systems where operating experience has indicated particular susceptibility to failure from the effects of intergranular stress corrosion cracking (IGSCC), water hammer, thermal fatigues, or erosion.

The LBB approach is not a replacement for existing regulations or criteria pertaining to the design bases of emergency core cooling system (Subsection 6.3), containment system (Subsection 6.2) or equipment qualification (Subsection 3.11). However, benefits of the LBB procedures to these areas will be taken and the subsections will be revised as the regulations will be relaxed by the NRC. For clarity, it is noted that the LBB approach is not used to relax the design requirements of the primary containment system that includes the primary containment vessel (PCV), vent systems (vertical flow channels and horizontal vent discharges), drywell zones, suppression chamber (wetwell), vacuum breakers, PCV penetrations, and drywell head. However, in designing for loads per Table 3.9-2, which does not apply to these PCV subsystems, the seven types of design loads identified with LOCA-induced dynamics of suppression pool or shield wall annulus pressurization are excluded if they are a result of LOCA postulated in those piping that meet the LBB criteria.

Appendix 3E characterizes fracture mechanics properties of piping materials and analysis methods including leakage calculation methods, as required by the criteria of this subsection. Following NRC's review and approval, this appendix will become approved LBB methodology for application to ABWR Standard Plant piping. Appendix 3F applies these properties and methods to specific piping to demonstrate their eligibility for exclusion under the LBB approach. See Subsection 3.6.4.2 for interface requirements.

3.6.3.1 General Evaluation

The high-energy piping system (or analyzable portion thereof) is evaluated with the following considerations in addition to the deterministic LBB evaluation procedure of Subsection 3.6.3.2

(1) Degradation by erosion, erosion/corrosion and erosion/cavitation due to unfavorable flow conditions and water chemistry is examined. The evaluation is based on the industry experience and guidelines. Additionally, fabrication wall thinning of elbows and other fittings is considered in the purchase specification to assure that the code minimum wall requirements are met. These evaluations demonstrate that these mechanisms are not potential sources of pipe rupture

(2) The ABWR plant design involves operation below 700°F in ferritic steel piping and below 800°F in austenitic steel piping. This assures that creep and creep-fatigue are not potential sources of pipe rupture.

(3) The design also assures that the piping material is not susceptible to brittle cleavage-type failure over the full range of system operating temperatures (that is, the material is on the upper shelf).

(4) The ABWR plant design specifies use of austenitic stainless steel piping made of material (e.g., nuclear grade or low carbon type) that is recognized as resistant to IGSCC. The material of piping in reactor coolant pressure boundary is ferritic steel.

(5) A systems evaluation of potential water hammer is made to assure that pipe rupture due to this mechanism is unlikely. Water hammer is a generic term including various unanticipated high frequency hydrodynamic events such as steam hammer and water slugging. To demonstrate that water hammer is not a significant contributor to pipe rupture, reliance on historical frequency of water hammer events in specific piping systems coupled with a review of operating procedures and conditions is used for this evaluation. The ABWR design includes features such as vacuum breakers and jockey pumps coupled with improved operational procedures to reduce or eliminate the potential for water hammer identified by past

experience. Certain anticipated water hammer events, such as a closure of a valve, are accounted for in the Code design and analysis of the piping.

(6) The systems evaluation also addresses a potential for fatigue cracking or failure from thermal and mechanical induced fatigue. Based on past experience, the piping design avoids potential for significant mixing of high- and low- temperature fluids or mechanical vibration. The startup and preoperational monitoring assures avoidance of detrimental mechanical vibration.

(7) Based on experience and studies by Lawrence Livermore Laboratory, potential indirect sources of indirect pipe rupture are remote causes of pipe rupture. Compliance with the snubber surveillance requirements of the technical specifications assures that snubber failure rates are acceptably low.

(8) Initial LBB evaluation is based on the design configuration and stress levels that are acceptably higher than those identified by the initial analysis. This evaluation is reconciled when the as-built configuration is documented and the Code stress evaluation is reconciled. It is assured that the as-built configuration does not deviate significantly from the design configuration to invalidate the initial LBB evaluation, or a new evaluation coupled with necessary configuration modifications is made to assure applicability of the LBB procedure.

(9) Sufficiently reliable, redundant, diverse and sensitive leak detection systems are provided for monitoring of leak. The system that is relied upon to predict the throughwall flaw used in the deterministic fracture mechanics evaluation is sufficiently reliable and sensitive to justify a margin of 2 on the leakage prediction.

3.6.3.2 Deterministic Evaluation Procedure

The following deterministic analysis and evaluation are performed as an NRC-approved method for the ABWR Standard Nuclear Island to justify applicability of the LBB concept.

(1) Use the fracture mechanics and the leak rate computational methods that are accepted by the NRC staff, or are demonstrated accurate with respect to other acceptable computational procedures or with experimental data.

(2) Identify the types of materials and materials specifications used for base metal, weldments and safe ends, and provide the materials properties including toughness and tensile data, long-term effects such as thermal aging, and other limitations.

(3) Specify the type and magnitude of the loads applied (forces, bending and torsional moments), their source(s) and method of combination. For each pipe size in the functional system, identify the location(s) which have the least favorable combination of stress and material properties for base metal, weldments and safe ends.

(4) Postulate a throughwall flaw at the location(s) specified in (3) above. The size of the flaw should be large enough so that the leakage is assured detection with sufficient margin using the installed leak detection capability when the pipes are subjected to normal operating loads. If auxiliary leak detection systems are relied on, they should be described. For the estimation of leakage, the normal operating loads (i.e., deadweight, thermal expansion, and pressure) are to be combined based on the algebraic sum of individual values.

Using fracture mechanics stability analysis or limit load analysis based on (11) below, and normal plus SSE loads, determine the critical crack size for the postulated throughwall crack. Determine crack size margin by comparing the selected leakage size crack to the critical crack size. Demonstrate that there is a margin of 2 between the leakage and critical crack sizes. The same load combination method selected in (5) below is used to determine the critical crack size.

(5) Determine margin in terms of applied loads by a crack stability analysis. Demonstrate

that the leakage size cracks will not experience unstable crack growth if 1.4 times the normal plus SSE loads are applied. Demonstrate that crack growth is stable and the final crack is limited such that a double-ended pipe break will not occur. The dead-weight, thermal expansion, pressure, SSE (inertial), and seismic anchor motion (SAM) loads are combined based on the same method used for the primary stress evaluation by the ASME Code. The SSE (inertial) and SAM loads are combined by square-root-of-the-sum-of-the- squares (SRSS) method.

(6) The piping material toughness (J-R curves) and tensile (stress-strain curves) properties are determined at temperatures near the upper range of normal plant operation.

(7) The specimen used to generate J-R curves is assured large enough to provide crack extensions up to an amount consistent with J/T condition determined by analysis for the application. Because practical specimen size limitations exist, the ability to obtain the desired amount of experimental crack extension may be restricted. In this case, extrapolation techniques is used as described in NUREG-1061, Volume 3, or in NUREG/CR-4575. Other techniques can be used if adequately justified.

(8) The stress-strain curves are obtained over the range from the proportional limit to maximum load.

(9) Preferably, the materials tests should be conducted using archival materials for the pipe being evaluated. If archival material is not available, plant specific or industry wide generic material data bases are assembled and used to define the required material tensile and toughness properties. Test material includes base and weld metals.

(10) To provide an acceptable level of reliability, generic data bases are reasonable lower bounds for compatible sets of material tensile and toughness properties associated with materials at the plant. To assure that the plant specific generic data base is adequate, a determination is made to demonstrate that the generic data base represents the range of plant materials to be evaluated. This determination is based on a comparison of the plant material properties identified in (2) above with those of the materials used to develop the generic data base. The number of material heats and weld procedures tested are adequate to cover the strength and toughness range of the actual plant materials. Reasonable lower bound tensile and toughness properties from the plant specific generic data base are to be used for the stability analysis of individual materials, unless otherwise justified.

Industry generic data bases are reviewed to provide a reasonable lower bound for the population of material tensile and toughness properties associated with any individual specification (e.g., A106, Grade B), material type (e.g., austenitic steel) or welding procedures.

The number of material heats and weld procedures tested should be adequate to cover the range of the strength and tensile properties expected for specific material specifications or types. Reasonable lower bound tensile and toughness properties from the industry generic data base are used for the stability analysis of individual materials.

If the data are being developed from an archival heat of material, three stress-strain curves and three J-resistance curves from that one heat of material is sufficient. The tests should be conducted at temperatures near the upper range of normal plant operation. Tests should also be conducted at a lower temperature, which may represent a plant condition (e.g., hot standby) where pipe break would present safety concerns similar to normal operation. These tests are intended only to determine if there is any significant dependence of toughness on temperature over the temperature range of interest. The lower toughness should be used in the fracture mechanics evaluation. One J-R curve and one stress-strain curve for one base metal and weld metal are considered adequate to determine temperature dependence.

(11) There are certain limitations that currently preclude generic use of limit load analyses to evaluate leak-before-break conditions deterministically. However, a modified limit-load analysis can be used for austenitic steel piping to demonstrate acceptable margins as indicated below:

Construct a master Curve where a stress index, SI, given by

$$SI = S + M\ P_m \qquad (1)$$

is plotted as a function of postulated total circumferential throughwall flaw length, L, defined by

$$L = 2\ \theta\ R \qquad (2)$$

where

$$S = \frac{2\sigma_f}{\pi}\ [\,2 \sin\beta - \sin\theta\,], \qquad (3)$$

$$\beta = 0.5\ [(\pi - \theta) - \pi\ (P_m/\sigma_f)] \qquad (4)$$

θ = half angle in radians of the postulated throughwall circumferential flaw.

R = pipe mean radius, that is, the average between the inner and outer radius,

P_m = the combined membrane stress, including pressure, deadweight, and seismic components,

M = 1.4, the margin associated with the load combination method selected for the analysis, per item (5).

σ_f = flow stress for austenitic steel pipe material categories.

If $\theta + \beta$ from Eqs. (2) and (4) is greater than π, then

$$S = \frac{2\sigma_f}{\pi}\ [\,\sin\beta\,] \qquad (5)$$

where

$$\beta = \ - \pi(P_m/\sigma_f). \qquad (6)$$

When the master curve is constructed using Eqs. (1), (2), and (3) or (5), the allowable circumferential throughwall flaw length can be determined by entering the master curve at a stress index (SI) value determined from the loads and austenitic steel piping material of interest. The allowable flaw size determined from the master curve at the appropriate SI value can then be used to determine if the required margins are met. Allowable values of θ are those that result in S being greater than zero from Eqs. (3) and (5). The flow stress used to construct the master curve and the definition of SI used to enter the master curve are defined for each material category as follows:

Base Metal and TIG Welds:

The flow stress used to construct the master curve is

$$\sigma_f = 0.5\ (\sigma_y + \sigma_u)$$

when the yield strength, σ_y, and the ultimate strength, σ_u, at temperature are known.

If the yield and ultimate strengths at temperature are not known, then Code minimum values at temperature can be used, or alternatively if

$$\frac{(SI)}{17M} < 2.5, \text{ then}$$

$$\sigma_f = 51 \text{ ksi, or}$$

if

$$\frac{(SI)}{17M} \geq 2.5, \text{ then}$$

$$\sigma_f = 45 \text{ ksi.}$$

The value of SI used to enter the master curve for base metal and TIG welds is

$$SI = M\ (P_m + P_b) \qquad (7)$$

where

P_b = the combined primary bending stress,

3.6.4 COL License Information

3.6.4.1 Details of Pipe Break Analysis Results and Protection Methods

The following shall be provided by the COL applicant (See Subsection 3.6.2.5):

(1) A summary of the dynamic analyses applicable to high-energy piping systems in accordance with Subsection 3.6.2.5 of Regulatory Guide 1.70. This shall include:

(a) Sketches of applicable piping systems showing the location, size and orientation of postulated pipe breaks and the location of pipe whip restraints and jet impingement barriers.

(b) A summary of the data developed to select postulated break locations including calculated stress intensities, cumulative usage factors and stress ranges as delineated in BTP MEB 3-1.

(2) For failure in the moderate-energy piping systems listed in Table 3.6-5, descriptions showing how safety-related systems are protected from the resulting jets, flooding and other adverse environmental effects. 410.21

(3) Identification of protective measures provided against the effects of postulated pipe failures for protection of each of the systems listed in Tables 3.6-1 and 3.6-2. 410.22

(4) The details of how the MSIV functional capability is protected against the effects of postulated pipe failures. 410.26

(5) Typical examples, if any, where protection for safety-related systems and components against the dynamic effects of pipe failures include their enclosure in suitably designed structures or compartments (including any additional drainage system or equipment environmental qualification needs). 410.28

(6) The details of how the feedwater line check and feedwater isolation valves functional capabilities are protected against the effects of postulated pipe failures.

3.6.4.2 Leak-Before-Break Analysis Report

As required by Reference 1, and LBB analysis report shall be prepared for the piping systems proposed for exclusion from analysis for the dynamic effects due to failure of piping failure. The report shall be parpared in accrodance with the guidelines presented in Appendix 3E and Submitted by the COL applicant to the NRC for approvai

3.6.5 References

1. *Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Rupture,* Federal Register, Volume 52, No. 207, Rules and Regulations, Pages 41288 to 41295, October 27, 1987

2. *RELAP 3, A Computer Program for Reactor Blowdown Analysis,* IN-1321, issued June 1970, Reactor Technology TID-4500.

3. ANSI/ANS-58.2, *Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture.*

4. *Standard Review Plan; Public Comments Solicited,* Federal Register, Volume 52, No. 167, Notices, Pages 32626 to 32633, August 28, 1987.

5. NUREG-1061, Volume3, *Evaluation of Potential for Pipe Breaks, Report of the U.S. NRC Piping Review Committee*, November 1984.

6. Mehta, H. S., Patel, N.T. and Ranganath, S., *Application of the Leak-Before-Break Approach to BWR Piping*, Report NP-4991, Electric Power Research Institute, Palo Alto, CA, December 1986.

Table 3.6-1

ESSENTIAL SYSTEMS, COMPONENTS, AND EQUIPMENT* FOR POSTULATED PIPE FAILURES INSIDE CONTAINMENT

1. Reactor Coolant Pressure Boundary (up to and including the outboard isolation valves)

2. Containment Isolation system and Containment Boundary (including liner plate)

3. Reactor Protection system (SCRAM SIGNALS)

4. Emergency Core Cooling Systems** (For LOCA events only)

   One of the following combinations is available (see Table 6.3-3):

   (a) HPCF (B and C) + RCIC + RHR-LPFL (B and C) + ADS
   (b) HPCF (B and C) + RHR-LPFL (A and B and C) + ADS
   (c) HPCF (B or C) + RCIC + RHR-LPFL (A and either of B or C) + ADS

5. Core Cooling Systems (other than LOCA events)

   (a) HPCF (B or C) or RCIC

   (b) RHR-LPFL (A or B or C) + ADS

   (c) RHR shutdown Cooling Mode (two loops)

   (d) RHR Suppression Pool Cooling Mode (two loops)

6. Control rod drive (scram/rod insertion)

7. Flow restrictors (passive)

8. Atmospheric control (for LOCA event only)

9. Standby gas treatment*** (for LOCA event only)

10. Control Room Environmental***

11. The following equipment/systems or portions thereof required to assure the proper operation of those essential items listed in items 1 through 10.

    (a) Class 1E electrical systems, ac and dc (including diesel generator system***, 6900, 480 and 120V ac, and 125V dc emergency buses***, motor control centers***, switchgear***, batteries*** and distribution systems)

Table 3.6-1

ESSENTIAL SYSTEMS, COMPONENTS, AND EQUIPMENT* FOR POSTULATED PIPE FAILURES INSIDE CONTAINMENT (Continued)

(b) Reactor Building Cooling Water*** to the following:

1. Room coolers
2. Pump coolers
3. Diesel generator jacket coolers
4. Electrical switchgear coolers

(c) Environmental Systems*** (HVAC)

(d) Instrumentation (including post-LOCA monitoring)

(e) Fire Protection System ***

410.22 (f) HVAC Emergency Cooling Water System ***

(g) Process Sampling System ***

NOTE

* The essential items listed in this table are protected in accordance with Subsection 3.6.1 consistent with the particular pipe break evaluated.

** Reference Section 6.3 for detailed discussion of emergency core cooling capabilities.

*** Located outside containment but listed for completeness of essential shutdown requirements.

Table 3.6-2

ESSENTIAL SYSTEMS, COMPONENTS, AND EQUIPMENT* FOR POSTULATED PIPE FAILURES OUTSIDE CONTAINMENT

1. Containment Isolation System and containment boundary.

2. Reactor Protection System (SCRAM signals)

3. Core Cooling systems

   (a) HPCF (B or C) or RCIC

   (b) RHR-LPFL (A or B or C) + ADS

   (c) RHR shutdown cooling mode (two loops)

   (d) RHR suppression pool cooling mode (two loops)

4. Flow restrictors

5. Control room habitability

6. Spent fuel pool cooling

7. Standby gas treatment

8. The following equipment/systems or portions thereof required to assure the proper operation of those essential items listed in items 1 through 7.

   (a) Class 1E electrical systems, ac and dc (including diesel generator system, 6900, 480 and 120V ac, and 125V dc emergency buses, motor control centers, switchgear, batteries, auxiliary shutdown control panel, and distribution systems).

   (b) Reactor Building Cooling water to the following:

      (1) Room coolers

      (2) Pump coolers (motors and seals)

      (3) Diesel generator auxiliary system coolers

      (4) Electrical switchgear coolers

      (5) RHR heat exchangers

* The essential items listed in this table are protected in accordance with Subsection 3.6.1 consistent with the particular pipe break evaluated.

Table 3.6-2

ESSENTIAL SYSTEMS, COMPONENTS, AND EQUIPMENT* FOR POSTULATED PIPE FAILURES OUTSIDE CONTAINMENT (Continued)

(6) FPC heat exchangers

(7) HECW refrigerators

(c) HVAC

(d) Instrumentation (including post accident monitoring)

(e) Fire Water System

(f) HVAC Emergency Cooling Water System

(g) Process Sampling System

Table 3.6-3

HIGH-ENERGY PIPING INSIDE CONTAINMENT

Piping System	
Main steam	
Main steam drains	
Steam supply to RCIC	
Feedwater	
Recirculation motor cooling	
HPCF	(RPV to first check valve)
RHR-LPFL	(RPV to first check valve)
RHR	(Suction from RPV to first normally closed gate valve)
Reactor Water Cleanup (from RHR and RPV drain)	
RPV head spray (RPV to first check valve)	
RPV vent (RPV to first closed valve)	
Standby Liquid Control (from HPCF to first check valve)	
CRD	(Scram/rod insertion)
RPV bottom head drain lines (RPV to first closed valves)	
Miscellaneous 3-inch and smaller piping	

Table 3.6-4

HIGH ENERGY PIPING OUTSIDE CONTAINMENT

Piping System*

Main Steam

Main Steam Drains

Steam supply to RCIC Turbine

CRD(to and from HCU)

RHR(injection to feedwater from nearest check valves in the RHR lines)

Reactor Water Cleanup (to Feedwater via RHR and to first inlet valve to RPV head spray)

Reactor Water Cleanup (pumps suction and discharge)

* *Fluid systems operating at high-energy levels less than 2 percent of the total time are not included. These systems are classified moderate-energy systems, (i.e., HPCF, RCIC, SAM and SLCS).*

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Table 3.6-6

MODERATE-ENERGY PIPING OUTSIDE CONTAINMENT

Residual Heat Removal System
(Piping beyond outermost isolation valve)

High Pressure Core Flooder System
(Piping beyond outermost isolation valve)

Reactor Core Isolation Cooling System
(Suction line from condensate storage pool beyond second shutoff valve; vacuum pump discharge line from vacuum pump to containment isolation valve)

Control Rod Drive System
(Piping up to pump suction)

Standby Liquid Control System
(Piping beyond injection valves)

Suppression Pool Cleanup System
(Beyond containment isolation valve)

Fuel Pool Cooling and Cleanup System

Radioactive Waste System
(Beyond isolation valve)

Instrument/Service Air System
(Beyond isolation valve)

HVAC Cooling Water System

Makeup Water System (Condensate)

Reactor Building Cooling Water System

Turbine Building Cooling Water System

Atmospheric Control System
(Beyond shutoff valve)

Table 3.6-7

ADDITIONAL CRITERIA FOR INTEGRATED LEAKAGE RATE TEST

(1) Those portions of fluids systems that are part of the reactor coolant pressure boundary, that are open directly to the primary reactor containment atmosphere under post-accident conditions and become an extension of the boundary of the primary reactor containment, shall be opened or vented to the containment atmosphere prior to or during the Type A test. Portions of closed systems inside containment that penetrate primary containment and are not relied upon for containment isolation purposes following a LOCA shall be vented to the containment atmosphere.

(2) All vented systems shall be drained of water to the extent necessary to ensure exposure of the system primary containment isolation valves to the containment air test pressure. 430.48b

(3) Those portions of fluid systems that penetrate primary containment, that are external to containment and are not designed to provide a containment isolation barrier, shall be vented to the outside atmosphere as applicable, to assure that full post-accident differential pressure is maintained across the containment isolation barrier.

(4) Systems that are required to maintain the plant in a safe condition during the Type A test shall be operable in their normal mode and are not vented.

(5) Systems that are normally filled with water and operating under post-LOCA conditions need not be vented.

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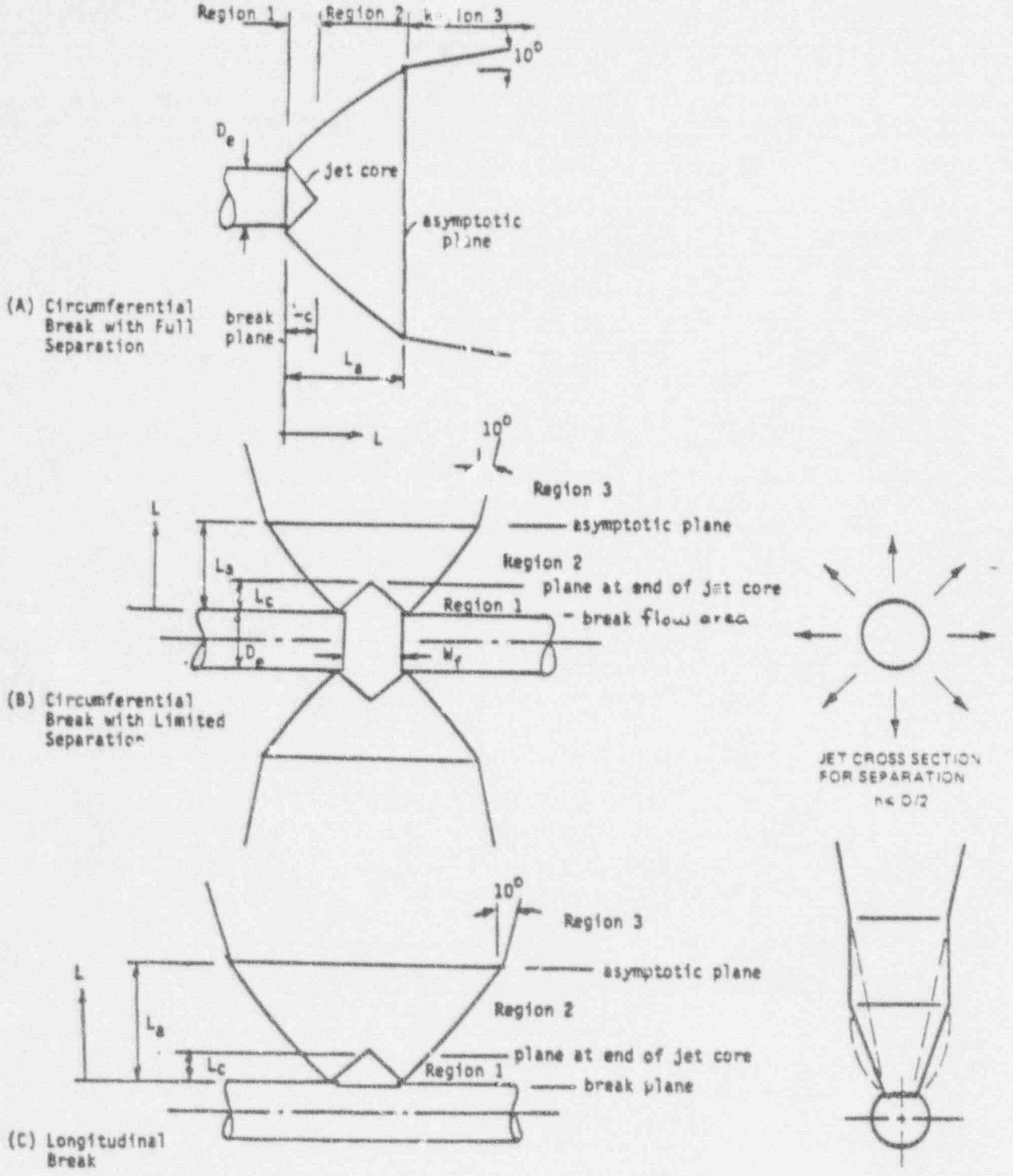

Figure 3.6-8 JET CHARACTERISTICS

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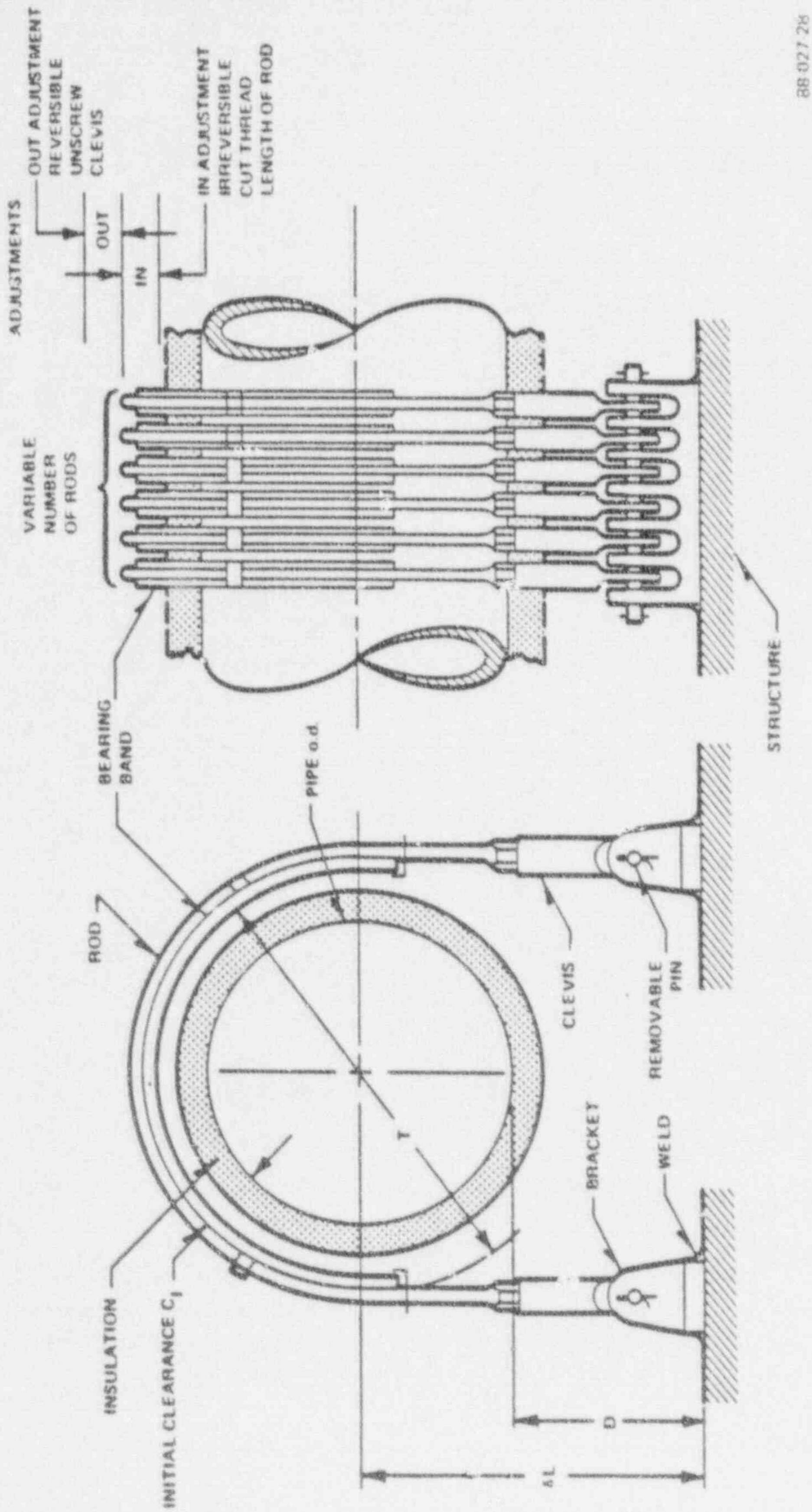

Figure 3.6-6 TYPICAL PIPE WHIP RESTRAINT CONFIGURATION

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ILLUSTRATIONS (Continued)

SECTION 3.7

ILLUSTRATIONS (Continued)

3.7 SEISMIC DESIGN

All structures, systems, and equipment of the facility are defined as either Seismic Category I or non-Seismic Category I. The requirements for Seismic Category I identification are given in Section 3.2 along with a list of systems, components, and equipment which are so identified.

All structures, systems, components, and equipment that are safety-related, as defined in Section 3.2, are designed to withstand earthquakes as defined herein and other dynamic loads including those due to reactor building vibration (RBV) caused by suppression pool dynamics. Although this section addresses seismic aspects of design and analysis in accordance with Regulatory Guide 1.70, the methods of this section are also applicable to other dynamic loading aspects, except for the range of frequencies considered. The cutoff frequency for dynamic analysis is 33 Hz for seismic loads and 60 ZHz for suppression pool dynamic loads. The definition of rigid system used in this section is applicable to seismic design only.

The safe shutdown earthquake (SSE) is that earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology, seismology, and specific characteristics of local subsurface material. It is that earthquake which produces the maximum vibratory ground motion for which Seismic Category I systems and components are designed to remain functional. These systems and components are those necessary to ensure:

(1) the integrity of the reactor coolant pressure boundary;

(2) the capability to shut down the reactor and maintain it in a safe shutdown condition; and

(3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10CFR100.

The operating basis earthquake (OBE) is that earthquake which, considering the regional and local geology, seismology, and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant. It is that earthquake which produce vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional. During the OBE loading condition, the safety-related systems are designed to be capable of continued safe operation. Therefore, for this loading condition, safety-related structures, and equipment are required to operate within design limits.

The seismic design for the SSE is intended to provide a margin in design that assures capability to shut down and maintain the nuclear facility in a safe condition. In this case, it is only necessary to ensure that the required systems and components do not lose their capability to perform their safety-related function. This is referred to as the no-loss-of-function criterion and the loading condition as the SSE loading condition.

Not all safety-related components have the same functional requirements. For example, the reactor containment must retain capability to restrict leakage to an acceptable level. Therefore, based on present practice, elastic behavior of this structure under the SSE loading condition is ensured. On the other hand, there are certain structures, components, and systems that can suffer permanent deformation without loss of function. Piping and vessels are examples of the latter where the principal requirement is that they retain contents and allow fluid flow.

Table 3.2-1 identifies the equipment in various systems as Seismic Category I or non-Seismic Category I.

3.7.1 Seismic Input

3.7.1.1 Design Response Spectra

The design earthquake loading is specified in terms of a set of idealized, smooth curves called the design response spectra in accordance with Regulatory Guide 1.60.

Figure 3.7-1 shows the standard ABWR design values of the horizontal SSE spectra applied at the ground surface in the free field for damping ratios of 2.0, 5.0, 7.0 and 10.0% of critical values of the vertical SSE spectra applied at the ground surface in the free field for damping ratios of 2.0, 5.0, 7.0, and 10.0% of critical damping where the maximum vertical ground acceleration is 0.30 g at 33Hz, same as the maximum horizontal ground acceleration.

The design values of the OBE response spectra are one-half* of the spectra shown in Figures 3.7-1 and 3.7-2. These spectra are shown in Figures 3.7-3 through 3.7-20.

The design spectra are constructed in accordance with Regulatory Guide 1.60. The normalization factors for the maximum values in two horizontal directions are 1.0 and 1.0 as applied to Figure 3.7-1. For vertical direction, the normalization factor is 1.0 as applied to Figure 3.7-2.

3.7.1.2 Design Time History

The design time histories are synthetic acceleration time histories generated to match the design response spectra defined in Subsection 3.7.1.1.

The design time histories considered in GESSAR (Reference 1) are used. They are developed based on the method proposed by Vanmarcke and Cornell (Reference 2) because of its intrinsic capability of imposing statistical independence among the synthesized acceleration time history components. The earthquake acceleration time history components are identified as H1, H2, and V. The H1 and H2 are the two horizontal components mutually perpendicular to each other. Both H1 and H2 are based on the design horizontal ground spectra shown in Figure 3.7-1. The V is the vertical component and it is based on the design vertical ground spectra shown in Figure 3.7-2.

* *The OBE given in Chapter 2 is one-third of the SSE, i.e., 0.10 g, for the ABWR Standard Nuclear Island design. However, as discussed in Chapter 2, a more conservative value of one-half of the SSE, i.e., 0.15 g, was employed to evaluate the structural and component response.*

The magnitude of the SSE design time history is equal to twice the magnitude of the design OBE time history. The OBE time histories and response spectra are used for dynamic analysis and evaluation of the structural Seismic System; the OBE results are doubled for evaluating the structural adequacy for SSE. For development of floor response spectra for Seismic Subsystem analysis and evaluation, see Subsection 3.7.2.5.

The response spectra produced from the OBE design time histories are shown in Figures 3.7-3 through 3.7-20 along with the design OBE response spectra. The closeness of the two spectra in all cases indicates that the synthetic time histories are acceptable.

The response spectra from the synthetic time histories for the damping values of 1, 2, 3 and 4 percent conform to the requirement for an enveloping procedure provided in Item II.1.b of Section 3.7.1 of NUREG-0800 (Standard Review Plan, SRP). However, the response spectra for the higher damping values of 7 and 10 percent show that there are some deviations from the SRP requirement. This deviation is considered inconsequential, because (1) generating an artificial time history whose response spectra would envelop design spectra for five different damping values would result in very conservative time histories for use as design basis input, and (2) the response spectra from the synthetic time histories do envelop the design spectra for the lower damping values. This is very important because the loads due to SSE on structures should use 7 percent damping for concrete components, but are obtained by ratioing up the response from the OBE analysis involving the lower damping. The OBE analysis uses only the lower damping values (up to 4%), which are consistent with the SRP requirements (See Subsection 3.7.1.3).

The frequency range used in generating the response spectra from synthetic histories is 0.2 to 33 Hz. The frequency range intervals used in generating those spectra is the same as given in Table 3.7.1-1 of SRP Section 3.7.1.

The coherence function for the three earthquake acceleration time history components H1, H2, and V are gener[illegible]d to check the statistical independence amo[illegible] them. The coherence function for H1 and H2 is given in Figure 3.7-21; for H1 and V in Figure 3.7-22; and for H2 and V in Figure 3.7-23. All values within the frequency range between 0 to 50 Hz are calculated at a frequency increment of 0.1 Hz. The small values of these coherence functions indicate that the three components are sufficiently statistically independent.

To assess the energy content of the synthetic time history, the power spectral density functions (PSDFs) are generated from the two horizontal components H1 and H2. The PSDFs are computed at a frequency increment of 0.024 Hz, and are smoothed using the average method as recommended in Revision 2 of Reference 3.

The stationary duration used in the calculation is taken to be 22 seconds which is the total duration of the synthetic time history. The calculated PSDFs for the H1 and H2 time histories normalized to 0.15g peak ground acceleration are shown in Figures 3.7-24 and 3.7-25, respectively, for frequencies ranging from 0.3 to 24 Hz.

The target PSDFs and 80% of target PSDFs specified on revision 2 of Reference 3 are also plotted on these figures for comparison. As shown, PSDF of H1 and H2 time histories envelope the target PSDF with a wide margin in the specified frequency range of 0.3 to 24 Hz. This demonstrates that the two synthetic time kectonies have sufficient energy content.

3.7.1.3 Critical Damping Values

The damping values for OBE and SSE analyses are presented in Table 3.7-1 for various structures and components. They are in compliance with Regulatory Guides 1.61 and 1.84

For seismic system evaluation of the SSE, the larger SSE damping values shown in Table 3.7-1 are not used. The SSE loads are obtained by doubling the OBE loads that result from the OBE Seismic System analysis based on the lower OBE damping values (see Subsection 3.7.1.2).

For analysis and evaluation of seismic subsystems (piping, components and equipment), the floor response spectra are obtained from the OBE time-history response of the seismic system, that supports the subsystems. The floor response spectra are computed (see Subsection 3.7.2.5) for damping values that are applicable to the subsystems under OBE as well as SSE; and further the OBE spectra are doubled to obtain the SSE floor response spectra for input to the SSE analysis in design of the subsystems.

3.7.1.4 Supporting Media for Seismic Category I Structures

The following ABWR Standard Plant Seismic Category I structures have concrete mat foundations supported on soil, rock or compacted backfill. The maximum value of the embedment depth below plant grade to the bottom of the base mat is given below for each structure.

(1) Reactor Building (including the enclosed primary containment vessel and reactor pedestal) - 25.7 m (84 ft, 4 in.).

(2) Control Building - 12.2 m (40 ft).

(3) Service Building - Surface founded.

All of the above buildings have independent foundations. In all cases the maximum value of embedment is used for the dynamic analysis to determine seismic soil-structure interaction effects. The foundation support materials withstand the pressures imposed by appropriate loading combinations without failure. The total structural height of each building is described in Subsection 3.8.2 through 3.8.4. For details of the structural foundations refer to Subsection 3.8.5. The ABWR Standard Plant is designed for a range of soil conditions given in Appendix 3A.

3.7.1.4.1 Soil-Structure Interaction

When a structure is supported on a flexible foundation, the soil-structure interaction is taken into account by coupling the structural model with the soil medium. The finite-element representation is used for a broad range of supporting medium conditions. A different representation based on the continuum impedance approach is also used for selected site conditions. Detailed methodology and results of the soil-structure interaction analysis are provided in Appendices 3A and 3G, respectively.

3.7.2 Seismic System Analysis

This subsection applies to the design of Seismic Category I structures and the reactor pressure vessel (RPV). Subsection 3.7.3 applies to all Seismic Category I piping systems and equipment.

3.7.2.1 Seismic Analysis Methods

Analysis of Seismic Category I structures and the RPV is accomplished using the response spectrum or time-history approach. The time-history approach is made either in the time domain or in the frequency domain.

Either approach utilizes the natural period, mode shapes, and appropriate damping factors of the particular system toward the solution of the equations of dynamic equilibrium. The time-history approach may alternately utilize the direct integration method of solution. When the structural response is computed directly from the coupled structure-soil system, the time-history approach solved in the frequency domain is used. The frequency domain analysis method is described in Appendix 3A.

3.7.2.1.1 The Equations of Dynamic Equilibrium for Base Support Excitation

Assuming velocity proportional damping, the dynamic equilibrium equations for a lumped-mass, distributed-stiffness system are expressed in a matrix form as:

$$[M]\{\ddot{u}(t)\} + [c]\{\dot{u}(t)\} + [K]\{u(t)\} = \{P(t)\} \tag{3.7-2}$$

where

$\{u(t)\}$	=	time-dependent displacement vector of non-support points relative to the supports $(u_t(t) = u(t) + u_s(t))$
$\{\dot{u}(t)\}$	=	time-dependent velocity vector of non-support points relative to the supports
$\{\ddot{u}(t)\}$	=	time-dependent acceleration vector of non-support points relative to the supports
[M]	=	mass matrix
[C]	=	damping matrix
[K]	=	stiffness matrix
$\{P(t)\}$	=	time-dependent inertia force vector $(-[M]\{u_s(t)\}$ acting at non-support points

The manner in which a distributed-mass, distributed-stiffness system is idealized into a lumped-mass, distributed-stiffness system of Seismic Category I structures and the RPV is shown in Figure 3.7-28 along with a schematic representation of relative acceleration; $\ddot{u}$ (t), support acceleration; $\ddot{u}_s(t)$ and total acceleration; $\ddot{u}_t(t)$.

3.7.2.1.2 Solution of the Equations of Motion by Modal Superposition

The technique used for the solution of the equations of motion is the method of modal superposition.

The set of homogeneous equations represented by the undamped free vibration of the system is:

$$[M]\{\ddot{u}(t)\} + [K]\{u(t)\} = \{0\}. \qquad (3.7\text{-}3)$$

Since the free oscillations are assumed to be harmonic, the displacements can be written as:

$$\{u(t)\} = \{\phi\}\, e^{i\omega t}, \qquad (3.7\text{-}4)$$

where

$\{\phi\}$ = column matrix of the amplitude of displacements $\{u\}$

ω = circular frequency of oscillation

t = time.

Substituting Equation 3.7-4 and its derivatives in Equation 3.7-3 and noting that $e^{i\omega t}$ is not necessarily zero for all values of ωt yields:

$$[-\omega^2 [M] + [K]]\,\{\phi\} = \{0\}. \qquad (3.7\text{-}5)$$

Equation 3.7-5 is the classic dynamic characteristic equation, with solution involving the eigenvalues of the frequencies of vibrations ω_i and the eigenvalues mode shapes, $\{\phi\}_i$, $(i = 1, 2, ..., n)$.

For each frequency ω_i, there is a corresponding solution vector $\{\phi\}_i$ determined to within arbitrary scalar factor Y_i known as the normal coordinate. It can be shown that the mode shape vectors are orthogonal with respect to the weighting matrix [K] in the n-dimensional vector space.

The mode shape vectors are also orthogonal with respect to the mass matrix [M].

The orthogonality of the mode shapes can be used to effect a coordinate transformation of the displacements, velocities and accelerations such that the response in each mode is independent of the response of the system in any other mode. Thus, the problem becomes one of solving n independent differential equations rather than n simultaneous differential equations; and, since the system is linear, the principle of superposition holds and the total response of the system oscillating simultaneously in n modes may be determined by direct addition of the responses in the individual modes.

3.7.2.1.3 Analysis by Response Spectrum Method

The response spectrum method is based on the fact that the modal response can be expressed as a set of convolution integrals which satisfy the governing differential equations. The advantage of this form of solution is that, for a given ground motion, the only variables under the integral are the damping factor and the frequency. Thus, for a specified damping factor it is possible to construct a curve which gives a maximum value of the integral as a function of frequency.

Using the calculated natural frequencies of vibration of the system, the maximum values of the modal responses are determined directly from the appropriate response spectrum. The modal maxima are then combined as discussed in Subsection 3.7.2.7.

When the equipment is supported at more than two points located at different elevations in the building, the response spectrum analysis is performed using the envelope response spectrum of all attachment points. Alternatively, the multiple support excitation analysis methods may be used where acceleration time histories or response spectra are applied to all the equipment attachment points. In some cases, the worst single floor response spectrum selected from a set of floor response spectra obtained at various floors may be applied identically to all floors provided there is no significant shift in frequencies of the spectra peaks.

3.7.2.1.4 Support Displacements in Multi-Supported Structures

In the preceding sections, analysis procedures for forces and displacements induced by time-dependent support displacement were discussed. In a multi-supported structure there are, in addition, time-dependent support displacements which produce additional displacements at nonsupport points and pseudo-static forces at both support and nonsupport points.

The governing equation of motion of a structural system which is supported at more than one point and has different excitations applied at each may be expressed in the following concise matrix form:

$$\begin{bmatrix} M_a & O \\ O & M_s \end{bmatrix} \begin{Bmatrix} \ddot{U}_a \\ \ddot{\bar{U}}_s \end{Bmatrix} + \begin{bmatrix} C_{aa} & C_{as} \\ C_{as} & C_{ss} \end{bmatrix} \begin{Bmatrix} \dot{U}_a \\ \dot{\bar{U}}_s \end{Bmatrix} + \begin{bmatrix} K_{aa} & K_{as} \\ K_{as} & K_{ss} \end{bmatrix} \begin{Bmatrix} U_a \\ \bar{U}_s \end{Bmatrix} = \begin{Bmatrix} \bar{F}_a \\ F_s \end{Bmatrix} \quad (3.7\text{-}6)$$

where

U_a = displacement of the active (unsupported) degrees of freedom;

$\bar{U}_s$ = Specified displacements of support points;

M_a and M_s = lumped diagonal mass matrices associated with the active degrees of freedom and the support points;

C_{aa} and K_{aa} = damping matrix and elastic stiffness matrix, respectively, expressing the forces developed in the active degrees of freedom due to the motion of the active degrees of freedom;

C_{ss} and K_{ss} = support forces due to unit velocities and displacement of the supports;

C_{as} and K_{as} = damping and stiffness matrices denoting the coupling forces developed in the active degrees of freedom by the motion of the supports and vice versa;

$\bar{F}_a$ = prescribed external time-dependent forces applied on the active degrees of freedom; and

F_s = reaction forces at the system support points.

Total differentiation with respect to time is denoted by ($\cdot$) in Equation 3.7-6. Also, the contributions of the fixed degrees of freedom have been removed in the equation. The procedure utilized to construct the damping matrix is discussed in Subsection 3.7.2.15. The mass and elastic stiffness matrices are formulated by using standard procedures.

Equation 3.7-6 can be separated into two sets of equations. The first set of equations can be written as:

$$[M_s]\{\ddot{\bar{U}}_s\} + [C_{ss}]\{\dot{\bar{U}}_s\} + [K_{ss}]\{\bar{U}_s\} + [C_{as}]\{\dot{U}_a\} + [K_{as}]\{U_a\} = \{F_s\}; \quad (3.7\text{-}7a)$$

and the second set as:

$$[M_a]\{\ddot{U}_a\} + [C_{aa}]\{\dot{U}_a\} + [K_{aa}]\{U_a\} + [C_{as}]\{\dot{\bar{U}}_s\} + [K_{as}]\{\bar{U}_s\} = \{\bar{F}_a\}; \quad (3.7\text{-}7b)$$

The timewise solution of Equation 3.7-7b can be obtained easily by using the standard normal mode solution technique. After obtaining the displacement response of the active degrees of freedom (U_a), Equation 3.7-7a can then be used to solve the support point reaction forces (F_s).

Modal superposition is used to determine the solutions of the uncoupled form of Equation 3.7-7a. The procedure is identical to that described in Subsection 3.7.2.1.2.

3.7.2.1.5 Dynamic Analysis of Buildings

The time-history method either in the time domain or in the frequency domain is used in the dynamic analysis of buildings. As for the modeling, both finite-element and lumped-mass methods are used.

3.7.2.1.5.1 Description of Mathematical Models

A mathematical model reflects the stiffness, mass, and damping characteristics of the actual structural systems. One important consideration is the information required from the analysis. Consideration of maximum relative displacements among supports of Seismic Category I structures, systems, and components require that enough points on the structure be used. Locations of Seismic Category I equipment are taken into consideration. Buildings are mathematically modeled as a system of lumped masses located at elevations of mass concentrations such as floors.

In general three-dimensional models are used for seismic analysis. In all structures, six degrees of freedom exist for all mass points (i.e., three translational and three rotational). However, in most structures, some of the dynamic degrees of freedom can be neglected or can be uncoupled form each other so that separate analyses can be performed for different types of motions.

Coupling between the two horizontal motions occurs when the center of mass, the centroid, and the center of rigidity do not coincide. The degree of coupling depends on the amount of eccentricity and the ratio of the uncoupled torsional frequency to the uncoupled lateral frequency. Since lateral/torsional coupling and torsional response can significantly influence floor accelerations, structures are in general designed to keep minimum eccentricities. However, for analysis of structures that possess unusual eccentricities a model of the support building is developed to include the effect of lateral/torsional coupling.

3.7.2.1.5.1.1 Reactor Building and Reactor Pressure Vessel

The reactor building (RB) complex includes: (a) the reinforced concrete containment vessel (RCCV) that includes the reactor shield wall (RSW), the reactor pedestal, and the reactor pressure vessel (RPV) and its internal components (b) the secondary containment zone having many equipment compartments, and (c) the clean zone. The building basemat is assumed to be rigid. Building elevations along the 0°-180° and 90°-270° sections are shown in Figures 3.7-29 and 3.7-30, respectively. The mathematical model is shown in Figure 3.7-31. Model elevations are with respect to the RPV bottom head. The model X and Y axes correspond to the RB 0°-180° and 90°-270° directions, respectively. The Z axis is along the vertical direction. The combined RB model as shown in Figure 3.7-31 basically consists of two uncoupled 2-D models in the X-Z and Y-Z planes since the building is essentially of a symmetric design with respect to its two principal directions in the horizontal plane. The coupling effects of the lateral and torsional motions on the building natural frequencies in the horizontal directions are found to be negligible. Therefore, the uncoupled 2-D models which omit the torsional degrees of freedom are used for seismic dynamic analysis. The methods used to account for torsional effects to define design loads are given in Subsection 3.7.2.11.

The model shown in Figure 3.7-31 corresponds to the X-Z plane. The only differences in terms of schematic representation between the X-Z and Y-Z plane models are that (1) the two building walls represented above EL. 18.5 m (60.7ft) in the X-Z plane by two sticks combine into one stick in the Y-Z plane, and (2) the rotational spring between the RCCV top slab (node 90) and the basemat top (node 88) is presented only in the X-Z plane.

Each structure in the reactor building complex is idealized by a center-lined stick model of a series of massless beam elements. Axial, flexural, and shear deformation effects are included in formulating beam stiffness terms. Coupling between individual structures is modeled by linear spring elements. Masses including dead weights of structural elements, equipment weights and piping weights are lumped to nodal points. The weights of water in the spent fuel storage pool and the suppression pool are also considered and lumped to appropriate locations.

The portions of the reactor building outside the RCCV are box-type shear wall systems of reinforced concrete construction. The major walls between floor slabs are represented by beam elements of a box cross section. The shear rigidity in the direction of excitation is provided by the parallel walls. The bending rigidity includes the cross walls contribution. The reactor building is fully integrated with the RCCV through floor slabs at various elevations. Spring elements are used to represent the slab in-plane shear stiffness in the horizontal direction. The outer and inner walls between EL. 44.7 m (146.6ft) and 18.5 m (60.7ft) along the X direction are also coupled rigidly in rotation about the Y axis at the connecting slab locations. In the vertical direction a single mass point is used for each slab and it is connected to the walls and RCCV by spring elements. The spring stiffness is determined so that the fundamental frequency of the slab in the vertical direction is maintained.

The RCCV is a cylindrical structure with a flat top slab with the drywell opening, which, along with upper pool girders and reactor building walls, form the upper pool. Mass points are selected at the RB floor slab locations. Stiffnesses are represented by a series of beam elements. In the X-Z plane, a rotational spring element connecting the top slab and the basemat is used to account for the additional rotational rigidity provided by the integrated RCCV-pool girder-building walls system. The RCCV is also coupled to the RPV through the refueling bellows, to the RSW through the RSW stabilizers, and to the reactor pedestal through the diaphragm floor. Spring elements are used to account for these interactions. The lower drywell access tunnels spanning between the RCCV and the reactor pedestal are not modeled since flexible rings are provided which are designed to reduce the coupling effects.

The RSW consists of two steel ring plates with concrete fill in between for shielding purposes. Concrete in the RSW does not contribute to stiffness; but its weight is included. The reactor pedestal is a cylindrical structure of a composite steel-concrete design. The total stiffness of the pedestal includes the full strength of the concrete core. Mass points are selected at equipment interface locations and geometrical discontinuities. In addition, intermediate mass points are chosen to result in more uniform mass distribution. The pedestal supports the reactor pressure vessel and it also provides lateral restraint to the reactor control rod drive housings below the vessel. The top of the RSW is connected to the RPV by the RPV stabilizers which are modeled as spring elements.

The model of the RPV and its internal components is described in Subsection 3.7.2.3.2. This model as shown in Figure 3.7-32 is coupled with the above-described RB model for the seismic analysis.

3.7.2.1.5.1.2 Control Building

The control building dynamic model is shown in Figure 3.7-33. The control building is box type shear wall system reinforced concrete. The major walls between floor slabs are represented by beam elements of a box cross section. The shear rigidity in the direction of excitation is provied by the parallel walls. The bending rigidity includes the cross walls contribution. In the vertical direction a single mass point is used for each slab and it is connected to the walls by spring elements. The spring element stiffness is determined so that the fundamental frequency of the slab in the vertical direction is maintained.

3.7.2.1.5.1.3 Radwaste Building

The radwaste building dynamic model is shown in Figure 3.7-34. The radwaste building is box type shear wall system of reinforced concrete. The major walls between floor slabs are represented by beam elements of a box cross section. The shear rigidity in the direction of excitation is provided by the parallel walls. The bending rigidity includes the cross walls contribution. In the vertical direction a single mass point is used for each slab and it is connected to the walls by spring elements. The spring element stiffness is determined so that the fundamental frequency of the slab in the vertical direction is maintained.

3.7.2.1.5.2 Rocking and Torsional Effects

Rocking effects due to horizontal ground movement are considered in the soil-structure interaction analysis as described in Appendix 3A. Whenever building response is calculated from a second step structural analysis, rocking effects are included as input simultaneously applied with the horizontal translational motion at the basemat. The torsional effect considered is described in Subsection 3.7.2.11.

3.7.2.1.5.3 Hydrodynamic Effects

For a dynamic system in which a liquid such as water is involved, the hydrodynamic effects on adjacent structures due to horizontal excitation are taken into consideration by including hydrodynamic mass coupling terms in the mass matrix. The basic formulas used for computing these terms are in Reference 4. In the vertical excitation, the hydrodynamic coupling effect are assumed to be negligible and the water mass is lumped to appropriate structural locations.

3.7.2.2 Natural Frequencies and Response Loads

The natural frequencies up to 33 Hz for the reactor-control buildings and radwaste are presented in Tables 3.7-2 through 3.7-5 and 3.7-10 for the fixed base condition.

Enveloped response loads at key locations in the reactor building complex due to OBE for the range of site conditions considered in Appendix 3A are presented in Appendix 3G . Response spectra at the major equipment elevations and support points are also given in Appendix 3G.

The SSE loads are two times the OBE loads as explained in Subsection 3.7.1.2.

3.7.2.3 Procedure Used for Modeling

3.7.2.3.1 Modeling Techniques for Systems Other Than Reactor Pressure Vessel

An important step in the seismic analysis of systems other than the reactor pressure vessel is the procedure used for modeling. The techniques center around two methods. The first method, the system is represented by lumped masses and a set of spring dashpots idealizing both the inertial and stiffness properties of the system. The details of the mathematical models are determined by the complexity of the actual structures and the information required for the analysis. For the decoupling of the subsystem and the supporting system, the following criteria equivalent to the SRP requirements are used:

(1) If $R_m \leq 0.01$, decoupling can be done for any R_f.

(2) If $0.01 \leq R_m \leq 0.1$, decoupling can be done if $R_f \leq 0.8$ or $R_f \geq 1.25$.

(3) If $R_m > 0.1$, an approximate model of the subsystem should be included in the primary system model.

Where R_m and R_f are defined as:

R_m = Total mass of the supported system/ Mass that supports the subsystem

R_f = Fundamental frequency of the supported subsystem/frequency of the dominant support motion

If the subsystem is comparatively rigid in relation to the supporting system, and also is rigidly connected to the supporting system, it is sufficient to include only the mass of the subsystem at the support point in the primary system model. On the other hand, in case of a subsystem supported by very flexible connections, e.g., pipe supported by hangers, the subsystem need not be included in the primary model. In most cases the equipment and components, which come under the definition of subsystems, are analyzed (or tested) as a decoupled system from the primary structure and the seismic input for the former is obtained by the analysis of the latter. One important exception to this procedure is the reactor coolant system, which is considered a subsystem but is usually analyzed using a coupled model of the reactor coolant system and primary structure.

In the second method of modeling, the structure of the system is represented as a two- or three-dimensional finite-element model using combinations of beam, plate, shell, and solid elements. The details of the mathematical models are determined by the complexity of the actual structures and the information required for the analysis.

3.7.2.3.2 Modeling of Reactor Pressure Vessel and Internals

The seismic loads on the RPV and reactor internals are based on coupled dynamic analysis with the reactor building. The mathematical model of the RPV and internals is shown in Figure 3.7-32. This model is coupled with the reactor building model for this analysis.

The RPV and internals mathematical model consists of lumped masses connected by elastic beam element members. Using the elastic properties of the structural components, the stiffness properties of the model are determined and the effects of axial bending and shear are included.

Mass points are located at all points of critical interest such as anchors, supports, points of discontinuity, etc. In addition, mass points are chosen so that the mass distribution in various zones is uniform as practicable and the full range of frequency of response of interest is adequately represented. Further, in order to facilitate hydrodynamic mass calculations, several mass points (fuel, shroud, vessel) are selected at the same elevation. The RPV and internals are quite stiff in the vertical direction. Vertical modes in the frequency range of interest are adequately obtained with few dynamic degrees of freedom. Therefore, vertical masses are distributed to a few key nodal points. The various length of control rod drive housing are grouped in to the two representative lengths shown in Figure 3.7-32. These lengths represent the longest and shortest housing in order to adequately represent the full range of frequency response of the housings.

Not included in the mathematical model are the stiffness properties of light components, such as in-core guide tubes and housings, sparger, and their supply headers. This is done to reduce the complexity of the dynamic model. For the seismic responses of these components, floor response spectra generated from system analysis is used.

The presence of a fluid and other structural components (e.g., fuel within the RPV) introduces a dynamic coupling effect. Dynamic effects of water enclosed by the RPV are accounted for by introduction of a hydrodynamic mass matrix which will serve to link the acceleration terms of the equations of motion of points at the same elevation in concentric cylinders with a fluid entrapped in the annulus. The details of the hydrodynamic mass derivation are given in Reference 4.

3.7.2.4 Soil-Structure Interaction

The soil model and soil-structure interaction analysis are described in Appendix 3A.

3.7.2.5 Development of Floor Response Spectra

In order to predict the seismic effects on equipment located at various elevations within a structure, floor response spectra are developed using a time-history analysis technique.

The procedure entails first developing the mathematical model assuming a linear system and then obtaining its natural frequencies and mode shapes. The dynamic response at the mass points is subsequently obtained by using a time-history approach.

Using the acceleration time-history response of a particular mass point, a spectrum response curve is developed and incorporated into a design acceleration spectrum to be utilized for the seismic analysis of equipment located at the mass point. Horizontal and vertical response spectra are computed for various damping values applicable for OBE and SSE evaluation of equipment. Two orthogonal horizontal and one vertical earthquake component are input separately. Response spectra at selected locations are then generated for each earthquake component separately. They are combined using the square-root-of-the-sum-of-the-squares (SRSS) method to predict the total co-directional floor response spectrum for that particular frequency. This procedure is carried out for each site-soil case used in the soil-structure interaction analysis. Response spectra for all site-soil cases are finally combined to arrive at one set of final response spectra.

An alternate approach to obtain co-directional floor response spectra is to perform dynamic analysis with simultaneous input of various earthquake components if those components are statistically independent to each other.

The SSE floor response spectra are obtained by doubling the OBE response spectra as explained in Subsection 3.7.1.3.

The response spectra values are computed as a minimum either at frequency intervals as specified in Table 3.7.1-1 of SRP 3.7.1 or at a set of frequencies in which each frequency is within 10% of the previous one.

3.7.2.6 Three Components of Earthquake Motion

The three components of earthquake motion are considered in the building seismic analyses. To properly account for the responses of systems subjected to the three-directional excitation, a statistical combination is used to obtain the net response according to the SRSS criterion of Regulatory Guide 1.92. The SRSS method accounts for the randomness of magnitude and direction of earthquake motion. The SRSS criterion, applied to the responses associated with the three components of ground earthquake motion, is used for seismic stress computation for steel structural design as well as for resultant seismic member force computations for reinforced concrete structural design

3.7.2.7 Combination of Modal Response

Since only the time-history method is used for seismic system analysis, the response spectrum combination of modal responses is not applied.

3.7.2.8 Interaction of Non-Category I Structures with Seismic Category I Structures

The interfaces between Seismic Category I and non-Seismic Category I structures and plant equipment are designed for the dynamic loads and displacements produced by both the Category I and non-Category I structures and plant equipment. All non-Category I structures will meet any one of the following requirements:

(1) The collapse of any non-Category I structure will not cause the non-Category I structure to strike a Seismic Category I structure component.

(2) The collapse of any non-Category I structure will not impair the integrity of Seismic Category I structures or components

(3) The non-Category I structures will be analyzed and designed to prevent their failure under SSE conditions in manner such that the margin of safety of these structures is equivalent to that of Seismic Category I structures.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

The following conservative assumptions are included in the calculation of the floor response spectra:

(1) The expected actual earthquake time histories are enveloped by a smooth ground response spectrum for design use. The smooth curve leads to conservative effects on modal analysis because it treats all the modes in the maximum acceleration range having the same amplification factor as the most strongly amplified.

(2) The time history used to calculate the floor response spectra produces a ground response which envelopes the design ground response spectra. In order to do this, it has spectral peaks which are substantially higher than the design spectra.

(3) The building and soil damping values used in the analysis are near the lower bound of the available damping data. The actual values of damping are expected to be much higher than the values used in the analysis.

(4) The yield strengths used in the analysis are based on the minimum values and are considerably lower than expected values.

(5) The additional strength and damping that is available when materials are stressed beyond yield are neglected when using linear elastic analytical methods.

(6) The working stresses for most equipment are usually considerably below the yield stresses.

(7) The calculated natural frequencies of equipment are usually lower than actual because of conservative modeling assumptions.

These elements of conservatism are in series (i.e., they are compounded), which results in an extremely conservative design. The only reason for broadening the spectra at all is to account for the unlikely possibility that a particular piece of equipment might have a natural frequency which is not on the calculated spectral peak but is on the real peak.

Since the peaks characteristic of the low damping response are narrow, such an occurrence is extremely improbable. Even if this eventuality does occur, the extreme conservatism described above ensures seismic adequacy of equipment design. Further, the floor response spectra obtained from the time-history analysis of the building are broadened plus and minus 10% in frequency. Alternatively, peak shifting method of ASME Code Case N-397, as permitted by Regulatory Guide 1.84, Revision 24, is used.

The broadening method of accounting for variations causes modes having frequencies near the spectral peaks to be calculated as though they experience the peak acceleration. This is quite conservative because the spectra for the actual structure have only one narrow peak somewhere in the 20% broadened range.

3.7.2.10 Use of Constant Vertical Static Factors

Since all Seismic Category I structures and the RPV are subjected to a vertical dynamic analysis with a time-history defining the input, no constant vertical static factors are utilized.

3.7.2.11 Methods Used to Account for Torsional Effects

Torsional effects for two-dimensional analytical models are accounted for in the following manner. The locations of the center of mass are calculated for each floor. The centers of rigidity and rotational stiffness are determined for each story. Torsion effects are introduced in each story by applying a rotational moment about its center of rigidity. The rotational moment is calculated as the sum of the products of the inertial force applied at the center of mass of each floor above and a moment arm equal to the distance from the center of mass of the floor to the center of rigidity of the story plus five percent of the maximum building dimension at the level under consideration. To be conservative, the absolute values of the moments are used in the sum. The torsional moment and story shear are distributed to the resisting structural elements in proportion to each individual stiffness.

The RPV model is axisymmetric with no built-in eccentricity. Hence, the torsional effects for the RPV are only those associated with the reactor building model.

3.7.2.12 Comparison of Responses

Since only the time-history method is used for structural analysis, the responses obtained from response spectrum and time-history methods are not compared.

3.7.2.13 Methods for Seismic Analysis of Category I Dams

The analysis of all Category I dams, if applicable for the site, taking into consideration the dynamic nature of forces (due to both horizontal and vertical earthquake loadings), the behavior of the dam material under earthquake loadings, soil structure interaction effects, and nonlinear stress-strain relations for the soil, will be used. Analysis of earth-filled dams, if applicable, includes an evaluation of deformations.

3.7.2.14 Determination of Seismic Category I Structure Overturning Moments

Seismic loads are dynamic in nature. The method of calculating seismic loads with dynamic analysis and then treating them as static loads to evaluate the overturning of structures and foundation failures while treating the foundation materials as linear elastic is conservative. Overturning of the structure, assuming no soil slip failure occurs, can be caused only by the center of gravity of the structure moving far enough horizontally to cause instability.

Furthermore, when the combined effect of earthquake ground motion and structural response is strong enough, the structure undergoes a rocking motion pivoting about either edge of the base. When the amplitude of rocking motion becomes so large that the center of structural mass reaches a position right above either edge of the base, the structure becomes unstable and may tip over. The mechanism of the rocking motion is like an inverted pendulum and its natural period is long compared with the linear, elastic struc- tural response. Thus with regard to overturning, the structure is treated as a rigid body.

The maximum kinetic energy can be conservatively estimated to be:

$$E_s = \frac{1}{2} \sum_i m_i \left[(v_H)_i^2 + (v_V)_i^2 \right] \qquad (3.7\text{-}8)$$

where (v_H) and (v_V) are the maximum values of the total lateral velocity and total vertical velocity, respectively, of mass m_i.

Values for $(v_H)_i$ and $(v_V)_i$ are computed as follows:

$$(v_H)_i^2 = (v_x)_i^2 + (v_H)_g^2 \qquad (3.7\text{-}9)$$

$$(v_V)_i^2 = (v_z)_i^2 + (v_V)_g^2 \qquad (3.7\text{-}10)$$

where $(v_H)_g$ and $(v_V)_g$ are the peak horizontal and vertical ground velocity, respectively, and $(v_x)_i$ and $(v_z)_i$ are the maximum values of the relative lateral and vertical velocity of mass m_i.

Letting m_o be total mass of the structure and base mat, the energy required to overturn the structure is equal to

$$E_o = m_o g h \qquad (3.7\text{-}11)$$

where h is the height to which the center of mass of the structure must be lifted to reach the overturning position. Because the structure may not be a symmetrical one, the value of h is computed with respect to the edge that is nearer to the center of mass. The structure is defined as stable against overturning when the ratio E_o to E_s exceeds 1.5.

These calculations assume the structure rests on the ground surface, hence, are conservative because the structure is actually embedded to a considerable depth. The embedded effect is considered only when the ratio E_o to E_s is less than 1.5.

3.7.2.15 Analysis Procedure for Damping

In a linear dynamic analysis using a modal superposition approach, the procedure to be used to properly account for damping in different elements of a coupled system model is as follows:

(1) The structural percent critical damping of the various structural elements of the model is first specified. Each value is referred to as the damping ratio (C_j) of a particular component which contributes to the complete stiffness of the system.

(2) An eigenvalue analysis of the linear system model is performed. This results in the eigenvector matrices (ϕ_i) which are normalized and satisfy the orthogonality conditions:

$$\phi_i^T K \phi_i = \omega_i^2, \text{ and } \phi_i^T K \phi_j = 0 \text{ for } i \neq j \qquad (3.7\text{-}12)$$

where

K = stiffness matrix;

ω_i = circular natural frequency associated with mode i; and

ϕ_i^T = transpose of i^{th} mode eigenvector ϕ_i

Matrix ϕ contains all translational and rotational coordinates.

(3) Using the strain energy of the individual components as a weighting function, the following equation is derived to obtain a suitable damping ratio (β_i) for mode i.

$$\beta_i = \frac{1}{\omega_i^2} \sum_{j=1}^{N} \left[C_j \left(\phi_i^T K \phi_i \right)_j \right] \qquad (3.7\text{-}13)$$

where

β_i = modal damping coefficient for i^{th} mode;

N = total number of structural elements;

ϕi = component of i^{th} mode eigenvector corresponding to j^{th} element;

ϕ_i^T = Transpose of ϕ_i defined above;

C_j = percent critical damping associated with element j;

K = stiffness matrix of element j; and

ω_i = circular natural frequency of mode i.

3.7.3 Seismic Subsystem Analysis

3.7.3.1 Seismic Analysis Methods

This subsection discusses the methods by which Seismic Category I subsystems and components are qualified to ensure the functional integrity of the specific operating requirements which characterize their Seismic Category I designation.

In general, one of the following five methods of seismically qualifying the equipment is chosen based upon the characteristics and complexities of the subsystem:

(1) dynamic analysis;

(2) testing procedures;

(3) equivalent static load method of analysis;

(4) a combination of (1) and (2); or

(5) a combination of (2) and (3).

Equivalent static load method of subsystem analysis is described in Subsection 3.7.3.5.

Appropriate design response spectra (OBE and SSE) are furnished to the manufacturer of the equipment for seismic qualification purposes. Additional information such as input time history is also supplied only when necessary.

When analysis is used to qualify Seismic Category I subsystems and components, the analytical techniques must conservatively account for the dynamic nature of the subsystems or components. Both the SSE and OBE, with their difference in damping values, are considered in the dynamic analysis as explained in Subsection 3.7.1.3.

The general approach employed in the dynamic analysis of Seismic Category I equipment and component design is based on the response spectrum technique. The time-history technique described in Subsection 3.7.2.1.1 generates timehistories at various support elevations for use in the analysis of subsystems and equipment. The structural response spectra curves are subsequently generated from the time history accelerations.

At each level of the structure where vital components are located, three orthogonal components of floor response spectra, two horizontal and one vertical, are developed. The floor response spectrum is smoothed and envelopes all calculated response spectra from different site soil conditions. The response spectra are peak broadened plus or minus 10%. When components are supported at two or more elevations, the response spectra of each elevation are superimposed and the resulting spectrum is the upper bound envelope of all the individual spectrum curves considered.

For vibrating systems and their supports, multi-degree-of-freedom models are used in accordance with the lumped-parameter modeling techniques and normal mode theory described in Subsection 3.7.2.1.1. Piping analysis is described in Subsection 3.7.3.3.1.

When testing is used to qualify Seismic Category I subsystems and components, all the loads normally acting on the equipment are simulated during the test. The actual mounting of the equipment is also simulated or duplicated. Tests are performed by supplying input accelerations to the shake table to such an extent that generated test response spectra (TRS) envelope the required response spectra.

For certain Seismic Category I equipment and components where dynamic testing is necessary to ensure functional integrity, test performance data and results reflect the following:

(1) performance data of equipment which has been subjected to dynamic loads equal to or greater than those experienced under the specified seismic conditions;

(2) test data from previously tested comparable equipment which has been subjected under similar conditions to dynamic loads equal to or greater than those specified; and

(3) actual testing of equipment in accordance with one of the methods described in Subsection 3.9.2.2 and Section 3.10.

3.7.3.2 Determination of Number of Earthquake Cycles

3.7.3.2.1 Piping

Fifty (50) peak OBE cycles are postulated for fatigue evaluation.

3.7.3.2.2 Other Equipment and Components

Criterion II.2.b of SRP Section 3.7.3 recommends that at least one safe shutdown earthquake (SSE) and five operating basis earthquakes (OBEs) should be assumed during the plant life. It also recommends that a minimum of 10 maximum stress cycles per earthquake should be assumed (i.e., 10 cycles for SSE and [illegible] cycles for OBE). For equipment and components other than piping, 10 peak OBE stress cycles are postulated for fatigue evaluation based on the following justification.

To evaluate the number of cycles engendered by a given earthquake, a typical Boiling Water Reactor Building reactor dynamic model was excited by three different recorded time histories: May 18, 1940, El Centro NS component, 29.4 sec; 1952, Taft N69° W component, 30 sec; and March 1957, Golden Gates 89°E component, 13.2 sec. The modal response was truncated so that the response of three different frequency bandwidths could be studied, 0^{+}-to-10 Hz, 10-to-20 Hz, and 20-to-50 Hz. This was done to give a good approximation to the cyclic behavior expected from structures with different frequency content.

Enveloping the results from the three earthquakes and averaging the results from several different points of the dynamic model, the cyclic behavior given in Table 3.7-6 was formed.

Independent of earthquake or component frequency, 99.5% of the stress reversals occur below 75% of the maximum stress level, and 95% of the reversals lie below 50% of the maximum stress level.

In summary, the cyclic behavior number of fatigue cycles of a component during a earthquake is found in the following manner:

(1) the fundamental frequency and peak seismic loads are found by a standard seismic analysis (i.e., from eigen extraction and forced response analysis);

(2) the number of cycles which the component experiences are found from Table 3.7-6 according to the frequency range within which the fundamental frequency lies; and

(3) for fatigue evaluation, one-half percent (0.005) of these cycles is conservatively assumed to be at the peak load, and 4.5% (0.045) at the three-quarter peak. The remainder of the cycles have negligible contribution to fatigue usage.

The SSE has the highest level of response. However, the encounter probability of the SSE is so small that it is not necessary to postulate the possibility of more than one SSE during the 60-year life of a plant. Fatigue evaluation due to the SSE is not necessary since it is a faulted condition and thus not required by ASME Code Section III.

The OBE is an upset condition and is included in fatigue evaluations according to ASME Code Section III. Investigation of seismic histories for many plants show that during a 60-year life it is probable that five earthquakes with intensities one-tenth of the SSE intensity, and one earthquake approximately 20% of the proposed SSE intensity, will occur. The 60-year life corresponds to 40 years of actual plant operation divided by a 67% usage factor. To cover the combined effects of these earthquakes and the cumulative effects of even lesser earthquakes, 10 peak OBE stress cycles are postulated for fatigue evaluation.

3.7.3.3 Procedure Used for Modeling

3.7.3.3.1 Modeling of Piping Systems

3.7.3.3.1.1 Summary

To predict the dynamic response of a piping system to the specified forcing function, the dynamic model must adequately account for all significant modes. Careful selection must be made of the proper response spectrum curves and proper location of anchors in order to separate Seismic Category I from non-Category I piping systems.

3.7.3.3.1.2 Selection of Mass Points

When performing a dynamic analysis, a piping system is idealized either as a mathematical model consisting of lumped masses connected by weightless elastic members or as a consistent mass model. The elastic members are given the properties of the piping system being analyzed. The mass points are carefully located to adequately represent the dynamic properties of the piping system. A mass point is located at the beginning and end of every elbow or valve, at the extended valve operator, and at the intersection of every tee. On straight runs, mass points are located at spacings no greater than the span length corresponding to 33 Hz. A mass point is located at every extended mass to account for torsional effects on the piping system. In addition, the increased stiffness and mass of valves are considered in the modeling of a piping system.

3.7.3.3.1.3 Selection of Spectrum Curves

In selecting the spectrum curve to be used for dynamic analysis of a particular piping system, a curve is chosen which most closely describes the accelerations existing at the end points and restraints of the system. The procedure for decoupling small branch lines from the main run of Seismic Category I piping systems when establishing the analytical models to perform seismic analysis are as follows:

(1) The small branch lines are decoupled from the main runs if they have a diameter less than one-third the diameter of the main run.

(2) The stiffness of all the anchors and its supporting steel is large enough to effectively decouple the piping on either side of the anchor for analytic and code jurisdictional boundary purposes. The RPV is very stiff compared to the piping system and therefore, it is modeled as an anchor. Penetration assemblies (head fittings and penetration sleeve pipe) are very stiff compared to the piping system and are modeled as anchors.

The stiffness matrix at the attachment location of the process pipe (i.e., main steam, RHR supply and return, RCIC, etc.) head fitting is sufficiently high to decouple the penetration assembly from the process pipe. Previous analysis indicates that a satisfactory minimum stiffness for this attachment point is equal to the stiffness in bending and torsion of a cantilevered pipe section of the same size as the process pipe and equal in length to three times the process pipe outer diameter.

For a piping system supported at more than two points located at different elevations in the building, the response spectrum analysis is performed using the envelope response spectrum of all attachment points. Alternatively, the multiple support excitation analysis methods may be used where acceleration time histories or response spectra are applied at all the piping attachment points. Finally, the worst single floor response spectrum selected from a set of floor response spectra obtained at various floors may be applied identically to all floors provided it envelops the other floor response spectra in the set.

3.7.3.3.2 Modeling of Equipment

For dynamic analysis, Seismic Category I equipment is represented by lumped-mass systems which consist of discrete masses connected by weightless springs. The criteria used to lump masses are:

(1) The number of modes of a dynamic system is controlled by the number of masses used; therefore, the number of masses is chosen so that all significant modes are included. The modes are considered as significant if the corresponding natural frequencies are less than 33 Hz and the stresses calculated from these modes are greater than 10% of the total stresses obtained from lower modes. This approach is acceptable provided at least 90% of the loading/inertia is contained in the modes used. Alternately,

the number of degrees of freedom are taken more than twice the number of modes with frequencies less than 33 Hz.

(2) Mass is lumped at any point where a significant concentrated weight is located (e.g., the motor in the analysis of pump motor stand, the impeller in the analysis of pump shaft, etc).

(3) If the equipment has free-end overhang span with flexibility significant compared to the center span, a mass lumped at the overhang span.

(4) When a mass is lumped between two supports, it is located at a point where the maximum displacement is expected to occur. This tends to lower the natural frequencies of the equipment because the equipment frequencies are in the higher spectral range of the response spectra. Similarly, in the case of live loads (mobile) and a variable support stiffness, the location of the load and the magnitude of support stiffness are chosen to yield the lowest frequency content for the system. This ensures conservative dynamic loads since the equipment frequencies are such that the floor spectra peak is in the lower frequency range. If not, the model is adjusted to give more conservative results.

3.7.3.3.3 Field Location of Supports and Restraints

The field location of seismic supports and restraints for Seismic Category I piping and piping systems components is selected to satisfy the following two conditions:

(1) the location selected must furnish the required response to control strain within allowable limits; and

(2) adequate building strength and stiffness for attachment of the component supports must be available.

The final location of seismic supports and restraints for Seismic Category I piping, piping system components, and equipment, including the placement of snubbers, is checked against the drawings and instructions issued by the engineer. An additional examination of these supports and restraining devices is made to assure that their location and characteristics are consistent with the dynamic and static analyses of the system.

3.7.3.4 Basis of Selection of Frequencies

Where practical, in order to avoid adverse resonance effects, equipment and components are designed/selected such that their fundamental frequencies are outside the range of 1/2 to twice the dominant frequency of the associated support structures. Moreover, in any case, the equipment is analyzed and/or tested to demonstrate that it is adequately designed for the applicable loads considering both its fundamental frequency and the forcing frequency of the applicable support structure. 210.27

All frequencies in the range of 0.25 to 33 Hz are considered in the analysis and testing of structures, systems, and components. These frequencies are excited under the seismic excitation.

If the fundamental frequency of a component is greater than or equal to 33 Hz, it is treated as seismically rigid and analyzed accordingly. Frequencies less than 0.25 Hz are not considered as they represent very flexible structures and are not encountered in this plant.

The frequency range between 0.25 Hz and 33 Hz covers the range of the broad band response spectrum used in the design.

3.7.3.5 Use of Equivalent Static Load Methods of Analysis

3.7.3.5.1 Subsystems Other Than NSSS

See Subsection 3.7.3.8.1.5 for equivalent static load analysis method.

3.7.3.5.2 NSSS Subsystems

When the natural frequency of a structure of component is unknown, it may be analyzed by applying a static force at the center of mass. In order to conservatively account for the possibility of more than one significant dynamic mode, the static force is calculated as 1.5

times the mass times the maximum spectral acceleration from the floor response spectra of the point of attachments of multispan structures. The factor of 1.5 is adequate for simple beam type structures. For other more complicated structures, the factor used is justified.

3.7.3.6 Three Components of Earthquake Motion

The total seismic response is predicted by combining the response calculated from the two horizontal and the vertical analysis.

When the response spectrum method is used, the method for combining the responses due to the three orthogonal components of seismic excitation is given as follows:

$$R_i = \left[\sum_{j=1}^{3} R_{ij}^2 \right]^{1/2} \qquad (3.7\text{-}14)$$

where

R_{ij} = maximum, coaxial seismic response of interest (e.g., displacement, moment, shear, stress, strain) in directions i due to earthquake excitation in direction j, (j = 1, 2, 3).

R_i = seismic response of interest in i direction for design (e.g., displacement, moment, shear, stress, strain) obtained by the SRSS rule to account for the nonsimultaneous occurrence of the R_{ij}'s.

3.7.3.7 Combination of Modal Response

3.7.3.7.1 Subsystems Other Than NSSS

When the response spectrum method of modal analysis is used, contributions from all modes, except the closely spaced modes (i.e., the difference between any two natural frequencies is equal to or less than 10%) are combined by the square-root-of-the-sum-of-the-squares (SRSS) combination of modal responses. This is defined mathematically as:

$$R = \sqrt{\sum_{i=1}^{N} (R_i)^2} \qquad (3.7\text{-}15)$$

where

R = combined response;

R_i = response to the i^{th} mode; and

N = number of modes considered in the analysis.

Closely spaced modes are combined by taking the absolute sum of the such modes.

An alternate to the absolute sum method presented in Regulatory Guide 1.92 is the following:

$$R = \left[\sum_{i=1}^{N} R_i^2 + 2\Sigma \left| R_\ell R_m \right| \right]^{1/2} \qquad (3.7\text{-}16)$$

where the second summation is to be done on all ℓ and m modes whose frequencies are closely spaced to each other.

3.7.3.7.2 NSSS Subsystems

In a response spectrum modal dynamic analysis, if the modes are not closely spaced (i.e., if the frequencies differ from each other by more than 10% of the lower frequency), the modal responses are combined by the square-root-of-the-sum-of-the-squares (SRSS) method as described in Subsection 3.7.3.7.1 and Regulatory Guide 1.92.

If some or all of the modes are closely spaced, a double sum method, as described in Subsection 3.7.3.7.2.2, is used to evaluate the combined response. In a time-history method of dynamic analysis, the vector sum of every step is used to calculate the combined response. The use of the time-history analysis method precludes the need to consider closely spaced modes.

3.7.3.7.2.1 Square-Root-of-the-Sum-of-the-Squares Method

Mathematically, this SRSS method is expressed as follows:

$$R = \left(\sum_{i=i}^{N} (R_i)^2 \right)^{1/2} \qquad (3.7\text{-}17)$$

where

R = combined response,

R_i = response to the i^{th} mode; and

N = number of modes considered in the analysis.

3.7.3.7.2.2 Double Sum Method

This method, as defined in Regulatory Guide 1.92, is mathematically:

$$R = \left(\sum_{k=1}^{N} \sum_{s=1}^{N} | R_k R_s | \epsilon_{ks} \right)^{1/2} \quad (3.7\text{-}18)$$

where

R = representative maximum value of a particular response of a given element to a given component of excitation;

R_k = peak value of the response of the element due to the k^{th} mode;

N = number of significant modes considered in the modal response combination; and

R_s = peak value of the response of the element attributed to s^{th} mode

where

$$\epsilon_{ks} = \left[1 + \left\{ \frac{(\omega'_k - \omega'_s)}{(\beta'_k \; \omega_k + \beta'_s \; \omega_s)} \right\}^2 \right]^{-1} \quad (3.7\text{-}19)$$

in which

$$\omega'_k = \omega_k \left[1 - \beta_k^2 \right]^{1/2}$$

$$\beta'_k = \beta_k + \frac{2}{t_d \, \omega_k}$$

where ω_k and β_k are the modal frequency and the damping ratio in the kth mode, respectively, and t_d is the duration of the earthquake.

3.7.3.8 Analytical Procedure for Piping

3.7.3.8.1 Piping Subsystems Other Than NSSS

3.7.3.8.1.1 Qualification by Analysis

The methods used in seismic analysis vary according to the type of subsystems and supporting structure involved. The following possible cases are defined along with the associated analytical methods used.

3.7.3.8.1.2 Rigid Subsystems with Rigid Supports

If all natural frequencies of the subsystem are greater than 33 Hz, the subsystem is considered rigid and analyzed statically as such. In the static analysis, the seismic forces on each component of the subsystem are obtained by concentrating the mass at the center of gravity and multiplying the mass by the appropriate maximum floor acceleration.

3.7.3.8.1.3 Rigid Subsystems with Flexible Supports

If it can be shown that the subsystem itself is a rigid body (e.g., piping supported at only two points) while its supports are flexible, the overall subsystem is modeled as a single-degree-of-freedom subsystem consisting of an effective mass and spring.

The natural frequency of the subsystem is computed and the acceleration determined from the floor response spectrum curve using the appropriate damping value. A static analysis is performed using 1.5 times the acceleration value. In lieu of calculating the natural frequency, the peak acceleration from the spectrum curve may be used.

If the subsystem has no definite orientation, the excitation along each of three mutually perpendicular axes is aligned with respect to the system to produce maximum loading. The excitation in each of the three axes is considered to act simultaneously. The excitations are combined by the SRSS method.

3.7.3.8.1.4 Flexible Subsystems

If the piping subsystem has more than two supports, it cannot be considered a rigid body and must be modeled as a multi-degree-of-freedom subsystem.

The subsystem is modeled as discussed in Subsection 3.7.3.3.1 in sufficient detail (i.e., number of mass points) to ensure that the lowest natural frequency between mass points is greater than 33 Hz. The mathematical model is analyzed using a time-history analysis technique or a response spectrum analysis approach. After the natural frequencies of the subsystem are obtained, a stress analysis is performed using the inertia forces and equivalent static loads obtained from the dynamic analysis for each mode.

For a response spectrum analysis based on a modal superposition method, the modal response accelerations are taken directly from the spectrum. The total seismic stress is normally obtained by combining the modal stress using the SRSS method. The seismic stress of closely spaced modes (i.e., within 10% of the adjacent mode) are combined by absolute summation. The resulting total is treated as a pseudomode and is then combined with the remaining modal stresses by the SRSS method.

The approach is simple and straightforward in all cases where the group of modes with closely spaced frequencies is tightly bundled (i.e., the lowest and the highest modes of the group are within 10% of each other). However, when the group of closely spaced modes is spaced widely over the frequency range of interest while the frequencies of the adjacent modes are closely spaced, the absolute sum method of combining response tends to yield over-conservative results. To prevent this problem, a general approach applicable to all modes is considered appropriate. The following equation is merely a mathematical representation of this approach.

The most probable system response, R, is given by:

$$R = \left(\sum_{i=1}^{N} R_i^2 + 2\Sigma \left| R_\ell R_m \right| \right)^{1/2} \qquad (3.7\text{-}20)$$

where the second summation is to be done on all ℓ and m modes whose frequencies are closely spaced to each other,

and where

R_i = response to the i^{th} mode

N = number of significant modes considered in the modal response combinations.

The excitation in each of the three major orthogonal directions is considered to act simultaneously with their effect combined by the SRSS method.

3.7.3.8.1.5 Static Analysis

A static analysis is performed in lieu of a dynamic analysis by applying the following forces at the concentrated mass locations (nodes) of the analytical model of the piping system:

(1) horizontal static load, $F_h = C_h W$, in one of the horizontal principal directions;

(2) equal static load, F_h, in the other horizontal principal direction; and

(3) vertical static load, $F_v = C_v W$;

where

C_h, C_v = multipliers of the gravity acceleration, g, determined from the horizontal and vertical floor response spectrum curves, respectively. (They are functions of the period and the appropriate damping of the piping system); and

W = weight at node points of the analytical model.

For special case analyses, C_h and C_v may be taken as:

(1) 1.0 times the zero-period acceleration of the response spectrum of subsystems described in Subsection 3.7.3.8.1.2;

(2) 1.5 times the value of the response spectrum at the determined frequency for subsystems described in Subsection 3.7.3.8.1.3 and 3.7.3.8.1.4; and

(3) 1.5 times the peak of the response spectrum for subsystems described in Subsections 3.7.3.8.1.3 and 3.7.3.8.1.4.

An alternate method of static analysis which allows for simpler technique with added conservatism is acceptable . No determination of natural frequencies is made, but rather the response of the subsystem is assumed to be the peak of the appropriate response spectrum at a conservative and justifiable value of damping. The response is then multiplied by a static coefficient of 1.5 to take into account the effects of both multifrequency excitation and multimodal response.

3.7.3.8.1.6 Dynamic Analysis

The dynamic analysis procedure using the response spectrum method is provided as follows:

(1) The number of node points and members is indicated. If a computer program is utilized, use the same order of number in the computer program input. The mass at each node point, the length of each member, elastic constants, and geometric properties are determined.

(2) The dynamic degrees of freedom according to the boundary conditions are determined.

(3) The dynamic properties of the subsystem (i.e., natural frequencies and mode shapes) are computed.

(4) Using a given direction of earthquake motion, the modal participation factors, s_j, for each mode are calculated:

$$s_j = \frac{\sum_{i=1}^{N} M_i \phi_{ij}}{\sum_{i=1}^{N} M_i \phi_{ij}^2} \qquad (3.7\text{-}21)$$

where

M_i = i^{th} mass

ϕij = component of ϕ_{ij} in the earthquake direction

$\phi' ij$ = i^{th} characteristic displacement in the j^{th} mode

s_j = modal participation factor for the j^{th} mode

N = number of masses.

(5) Using the appropriate response spectrum curve, the spectral acceleration, r_a, for the j^{th} mode as a function of the j^{th} mode natural frequency and the damping of the system is determined.

(6) The maximum modal acceleration at each mass point, i, in the model is computed as follows:

$$a_{ij} = s_j \, r_{aj} \phi_{ij} \qquad (3.7\text{-}22)$$

where

a_{ij} = acceleration of the i^{th} mass point in the j^{th} mode.

(7) The maximum modal inertia force at the i^{th} mass point for the j^{th} mode is calculated from the equation:

$$F_{ij} = M_i \; a_{ij} \qquad (3.7\text{-}23)$$

(8) For each mode, the maximum inertia forces

are applied to the subsystem model, and the modal forces, shears, moments, stresses, and deflections are determined.

(9) The modal forces, shears, moments, stresses, and deflections for a given direction are combined in accordance with Subsection 3.7.3.8.1.4.

(10) Steps (5) through (9) are performed for each of the three earthquake directions.

(11) The seismic force, shear, moment, and stress resulting from the simultaneous application of the three components of earthquake loading are obtained in the following manner:

$$R = \sqrt{R_x^2 + R_y^2 + R_z^2} \qquad (3.7\text{-}24)$$

R = equivalent seismic response quantity (force, shear, moment, stress, etc.)

$R_x\ R_y\ R_z$ = colinear response quantities due to earthquake motion in the x, y, and z directions, respectively.

3.7.3.8.1.7 Damping Ratio

The damping ratio percentage of critical damping of piping subsystems corresponds to Regulatory Guide 1.61 or 1.84 (ASME Code Case N-411-1). The damping ratio is specified in Table 3.7-1.

3.7.3.8.1.8 Effect of Differential Building Movements

In most cases, piping subsystems are anchored and restrained to floors and walls of buildings that may have differential movements during a seismic event. The movements may range from insignificant differential displacements between rigid walls of a common building at low elevations to relatively large displacements between separate buildings at a high seismicity site.

Differential endpoint or restraint deflections cause forces and moments to be induced into the piping system. The stress thus produced is a secondary stress. It is justifiable to place this stress, which results from restraint of free-end displacement of the piping system, in the secondary stress category because the stresses are self-limiting and, when the stresses exceed yield strength, minor distortions or deformations within the piping system satisfy the condition which caused the stress to occur.

The earthquake thus produces a stress-exhibiting property much like a thermal expansion stress and a static analysis can be used to obtain actual stresses. The differential displacements are obtained from the dynamic analysis of the building. The displacements are applied to the piping anchors and restraints corresponding to the maximum differential displacements which could occur. The static analysis is made three times: once for one of the horizontal differential displacements, once for the other horizontal differential displacement, and once for the vertical.

3.7.3.8.2 NSSS Piping Subsystems

3.7.3.8.2.1 Dynamic Analysis

As described in Subsection 3.7.3.3.1, pipe line is idealized as a mathematical model consisting of lumped masses connected by elastic members. The stiffness matrix for the piping subsystem is determined using the elastic properties of the pipe. This includes the effects of torsional, bending, shear, and axial deformations as well as changes in stiffness due to curved members.

Next, the mode shapes and the undamped natural frequencies are obtained. The dynamic response of the subsystem is usually calculated by using the response spectrum method of analysis. When the connected equipment is supported at more than two points located at different elevations in the building, the response spectrum analysis is performed using the envelope response spectrum of all attachment points. Alternatively, the multiple excitation analysis methods may be used where acceleration time histories or response spectra are applied at all the equipment and piping attachment points.

3.7.3.8.2.2 Effect of Differential Building Movements

The relative displacement between anchors is determined from the dynamic analysis of the structures. The results of the relative anchor-point displacement are used in a static analysis to determine the additional stresses due to relative anchor-point displacements. Further details are given in Subsection 3.7.3.8.1.8.

3.7.3.9 Multiple Supported Equipment Components With Distinct Inputs

The procedure and criteria for analysis are described in Subsections 3.7.2.1.3 and 3.7.3.3.1.3.

3.7.3.10 Use of Constant Vertical Static Factors

All Seismic Category I subsystems and components are subjected to a vertical dynamic analysis with the vertical floor spectra or time histories defining the input. A static analysis is performed in lieu of dynamic analysis if the peak value of the applicable floor spectra times a factor of 1.5 is used in the analysis. A factor of 1.0 instead of 1.5 can be used if the equipment is simple enough such that it behaves essentially as a single degree of freedom system. If the fundamental frequency of a component in the vertical direction is greater than or equal to 33 Hz, it is treated as seismically rigid and analyzed statically using the zero-pe-sponse spectrum.

210.28

3.7.3.11 Torsional Effects of Eccentric Masses

Torsional effects of eccentric masses are included for Seismic Category I subsystems similar to that for the piping systems discussed in Subsection 3.7.3.3.1.2.

3.7.3.12 Buried Seismic Category I Piping and Tunnels

For buried Category I buried piping systems and tunnels the following items are considered in the analysis:

(1) The inertial effects due to an earthquake upon buried systems and tunnels will be adequately accounted for in the analysis. In case of buried systems sufficiently flexible relative to the surrounding or underlying soil, it is assumed that the systems will follow essentially the displacements and deformations that the soil would have if the systems were absent. When applicable, procedures, which take into account the phenomena of wave travel and wave reflection in compacting soil displacements from the ground displacements, are employed.

(2) The effects of static resistance of the surrounding soil on piping deformations or displacements, differential movements of piping anchors, bent geometry and curvature changes, etc., are considered. When applicable, procedures utilizing the principles of the theory of structures on elastic foundations are used.

(3) When applicable, the effects due to local soil settlements, soil arching, etc., are also considered in the analysis.

3.7.3.13 Interaction of Other Piping with Seismic Category I Piping

In certain instances, non-Seismic Category I piping may be connected to Seismic Category I piping at locations other than a piece of equipment which, for purposes of analysis, could be represented as an anchor. The transition points typically occur at Seismic Category I valves which may or may not be physically anchored. Since a dynamic analysis must be modeled from pipe anchor point to anchor point, two options exis

(1) specify and design a structural anchor at the Seismic Category I valve and analyze the Seismic Category I subsystem; or, if impractical to design an anchor,

(2) analyze the subsystem from the anchor point in the Seismic Category I subsystem through the valve to either the first anchor point in the non-Seismic Category I subsystem; or to sufficient distance in the non-Seismic Category I Subsystem so as not to significantly degrade the accuracy of analysis of the Seismic Category I piping.

Where small, non-Seismic Category piping is directly attached to Seismic Category I piping, its effect on the Seismic Category I piping is accounted for by lumping a portion of its mass with the Seismic Category I piping at the point of attachment.

Furthermore, non-Seismic Category I piping (particularly high energy piping as defined in Section 3.6) is designed to withstand the SSE to avoid jeopardizing adjacent Seismic Category I piping if it is not feasible or practical to isolate these two piping systems.

3.7.3.14 Seismic Analysis for Reactor Internals

The modeling of RPV internals is discussed in Subsection 3.7.2.3.2. The damping values are given in Table 3.7-1. The seismic model of the RPV and internal in shown in Figure 3.7-32.

3.7.3.15 Analysis Procedures for Damping

The modeling of RPV internals is discussed in Subsection 3.7.2.3.2. The damping values are given in Table 3.7-1. The seismic model of the RPV and internals is shown in Figure 3.7-32.

3.7.3.16 Analysis Procedure for NonSeismic Structures in Lieu of Dynamic Analysis

The method described here can be used for non-seismic structures in lieu of a dynamic analysis.

Structures designed to this method should be able to do the following:

(1) Resist minor levels of earthquake ground motion without damage.

(2) Resist moderate levels of earthquake ground motion without structural damage, but possibly experience some nonstructural damage.

(3) Resist major levels of earthquake ground motion having an intensity equal to the strongest either experienced or forecast at the building site, without collapse, but possibly with some structural as well as nonstructural damage.

3.7.3.16.1 Lateral Forces

Seismic loads are characterized as a force profile that varies with the height of the structure. These forces are applied at each floor of the structure and the resulting forces and moments are calculated from static equilibrium.

The buildings total base shear is characterized by the following equation:

$V = Z^{*}I^{*}C^{*}W/R_w$; where,

V	=	Total lateral force or shear at the base.
$F_x F_i F_n$	=	Lateral force applied to level i, n, or x respectively.
F_t	=	That portion of V considered to be concentrated at the top of the structure in addition to F_n
Z	=	Seismic zone factor
I	=	Importance factor
C	=	Numerical Coefficient
R_w	=	Numerical Coefficient
S	=	Coefficient for site soil characteristics
T	=	Fundamental period of vibration of the structure in the direction under consideration, as determined by using the properties and deformation characteristics of the resisting elements in a properly substantiated analysis.
W	=	Total dead load of building including the partition load where applicable.
$w_i w_x$	=	That portion of W which is located at or is assigned to level i or x, respectively
$h_i h_x$	=	Height in feet above the base to level i or x, respectively

The ABWR design will fix Z and I and leave R and C as variables for each building and site.

The value of I has been selected for power generating facilities.

$I = 1.0$

The site coefficient Z will be selected to provide enveloping coverage for most of the U.S. east of rocky mountains.

$Z = 0.15$

The value of C is calculated based upon the following formula:

$C = 1.25*ST^{2/3}$

Where: C need not exceed 2.75

The value of S is dependent on the site soil characteristics. The value of S shall be selected from Table 3.7-11.

The value of R_w shall be selected from Table 3.7-12 according to the type of construction material and framing system under consideration.

3.7.3.16.2 Lateral Force Distribution

The concentrated force at the top of the structure shall be determined according to the following formula:

$F_t = 0.07*T*V$ where,

F_t need not exceed 0.25V and may be considered as 0 where T is 0.7 seconds or less. The remaining portion of the total base shear V shall be distributed over the rest of the structure including level n according to the following formula:

$$F_x = \frac{(V - F_t)\, w_x h_x}{\sum_{i=1}^{n} w_i h_i}$$

At each level designated x, the force F_x shall be applied over the area of the building in accordance with the mass distribution on that level.

3.7.3.16.3 Accident Torsion

In addition, the vertical resisting elements depend on diaphragm action for shear distribution at any level, the shear resisting elements shall be capable of resisting torsional moment assumed to be equivalent to the story shear acting with an eccentricity of not less than 5 percent of the maximum building dimension at that level.

3.7.3.16.4 Lateral Displacement Limits

Lateral deflections or drift of a story relative to its adjacent stories shall not exceed 0.005 times the story height nor $0.04/R_w$ for buildings less than 65 feet in height. For buildings greater in height, the calculated story drift shall not exceed 0.004 times the story height nor $0.04/R_w$. These drift limits may be exceeded when it is demonstrated that greater drift can be tolerated by both structural elements and nonstructural elements that could effect life or safety. For designs using working stress methods, this capacity may be determined using an allowable stress increase of 1.7. The rigidity of other elements shall also be considered.

3.7.3.16.5 Ductility Requirements

All framing not required by design to be part of the lateral force-resisting system shall be investigated and shown to be adequate for vertical load-carrying capacity and induced moment due to $3R_w/8$ times the distortions resulting from the code required lateral forces.

Connections shall be designed to develop the full capacity of the members or shall be based upon the above forces without the one-third increase usually permitted for stresses resulting from earthquake forces.

3.7.4 Seismic Instrumentation

3.7.4.1 Comparison with NRC Regulatory Guide 1.12

The seismic instrumentation program is consistent with Regulatory Guide 1.12.

3.7.4.2 Location and Description of Instrumentation

The following instrumentation and associated equipment are used to measure plant response to earthquake motion:

(1) three triaxial time-history accelerographs (THA);

(2) three peak-recording accelerographs (PRA);

(3) two triaxial seismic triggers;

(4) one seismic switch (SS);

(5) four response spectrum recorders;

(6) recording and playback equipment; and

(7) annuciators.

The location of seismic instrumentation is outlined in Table 3.7-7.

3.7.4.2.1 Time-History Accelerographs

Time-history accelerographs produce a record of the time-varying acceleration at the sensor location. This data is used directly for analysis and comparison with reference information and may be, by calculational methods, converted to response spectra form for spectra comparisons with design parameters.

Each triaxial acceleration sensor unit contains three accelerometers mounted in an orthogonal array (two horizontal and one vertical). All acceleration units have their principal axes oriented identically. The mounted units are oriented so that their axes are aligned with the building major axes used in development of the mathematical models for seismic analysis.

One THA is located on the reactor building (RB) foundation mat, El (-) 13.2 M, at the base of an RB clean zone for the purpose of measuring the input vibratory motion of the foundation mat. A second THA is located in an RB clean zone at El (+) 26.7 M on the same azimuth as the foundation mat THA. They provide data on the frequency, amplitude, and phase relationship of the seismic response of the reactor building structure. A third THA is located in the free field at the finished grade approximately 160 M from any station structures with axes oriented in the same direction as the reactor building accelerometers.

Two seismic triggers, connected to form redundant triggering, are provided to start the THA recording system. They are located in the free field at the finished grade 160 M from the reactor building. The trigger unit consists of orthogonally mounted acceleration sensors that actuate relays whenever a threshold acceleration is exceeded for any of the three axes. The trigger in engineered to discriminate against false starts from other operating inputs such as traffic, elevators, people, and rotating equipment.

Magnetic tape recording and playback units are provided for multiple channel recording and playback of the THA accelerometer signals. The data recordings include an additional recorded channel for the timing reference signal generated in the control unit. The recording and playback systems have a special cabinet furnished for those instruments and devices necessary for system testing, annunciating, calibration, and control. This cabinet is located in the control equipment room.

3.7.4.2.2 Peak Recording Accelerographs

Each sensor unit contains three peak-recording accelerographs mounted in a mutually orthogonal array. The units are unpowered and record peak accelerations triaxially by proportional scratches on record plates. The PRAs that are mounted directly on equipment have one axis coincident with the principal equipment axis. All other PRAs have their principal axes oriented identically with one horizontal axis parallel to the major horizontal axis assumed in the seismic analysis.

One PRA is located on a reactor water cleanup unit (RWCU) regenerative heat exchanger support. A second PRA is located on an RHR pipe support. A third PRA is located on a diesel generator support.

Data from PRAs must be manually retrieved following an earthquake and is used in the detailed investigations for particular structures, systems, and equipment.

3.7.4.2.3 Seismic Switches

One triaxial seismic switch (SS) is installed on the reactor building foundation. This device actuates a visual and audible annunciator in the main control room when the OBE acceleration on at least one of the axes has been exceeded. When the threshold acceleration is sensed, the relay closes and remains closed for an adjustable period after the threshold is no longer exceeded.

3.7.4.2.4 Response Spectrum Recorders

The response spectrum recorders measure both horizontal and vertical peak acceleration for a series of frequencies pertinent to specific structures and equipment. Response spectra are recorded for three mutually orthogonal directions at the sensor location by inscribing steel reed deflections upon record plates. One recorder is located on the reactor building foundation in a clean zone. Another recorder is located on the control building foundation. If the OBE design response spectra values for specific frequencies are exceeded during an earthquake, specific switches mounted in the recorders annunciate the specific frequencies in the control equipment room.

Two other recorders do not contain alarm contacts. One is mounted in the reactor building pipe tunnel on a 20-inch RHR line and another is on a FMCRD control panel support.

3.7.4.2.5 Recording and Playback Equipment

A cabinet located in the control equipment room houses the recording, playback, and calibration units that are used in conjunction with the THA sensors to produce a time-history record of the earthquake. It also contains audible and visual annunciators wired to display initiation of the THA recorder and the power supply components for all equipment contained within the cabinet.

3.7.4.3 Control Room Operator Notification

Activation of the seismic triggers causes an audible and visual annunciation in the main control room to alert the plant operator that an earthquake has occurred. The annunciation is set to occur at 0.01g vertical acceleration on the free field.

The triggers cause initiation of the THA recording system at horizontal or vertical acceleration levels slightly higher than the expected background level including induced vibrations from sources such as traffic, elevators, people, and machinery. The initial set points may be changed once significant plant operating data have been obtained which indicate that a different setpoint would provide better THA system operation.

Audible and visual annunciators are provided in the main control room to indicate whether the OBE floor accelerations have been exceeded for the seismic switch location.

The peak acceleration level experienced by the reactor building basemat is available immediately following the earthquake. This is obtained by playing back the recorded THA data from the basemat location and reading the peak value from a strip chart recorder.

Significant response spectra from the reactor building basemat are available immediately following an earthquake for comparison with the OBE and SSE response spectra.

3.7.4.4 Comparison of Measured and Predicted Responses

Initial determination of the earthquake level is performed immediately after the earthquake by comparing the measured response spectra from the reactor building basemat with the OBE and SSE response spectra for the corresponding location. If the measured spectra exceed the OBE response spectra, the plant is shut down and a detailed analysis of the earthquake motion is undertaken.

After any earthquake, the data from all seismic recorders and recording instruments are retrieved. When the OBE has been exceeded, the data from these instruments are analyzed to obtain the seismic accelerations experienced at the location of major Seismic Category I structures and equipment. The measured response from the time-history accelerographs, peak-recording accelerographs, and response spectrum recorders are used to determine the response spectra at the location of each Seismic Category I structure and system. These spectra are compared with those used in the design to determine whether the structure or system is still adequate for future use. Peak-recording accelerographs mounted on equipment are used to determine whether the design limitation of that specific equipment has been exceeded.

The theoretical structural response and measured structural responses are compared to assess the degree of conservatism in the analytical predictions. Seismic levels are established to determine whether the plant can be brought back on line. The criteria consider system design and dynamic analysis in establishing the acceptable levels for continued operation.

3.7.4.5 In-Service Surveillance

Each of the seismic instruments will be demonstrated operable by the performance of the channel check, channel calibration, and channel functional test operations at the intervals specified in Table 3.7-9.

3.7.5 COL License Information

3.7.5.1 Seismic Parameters

The design basis horizontal g value is 0.3g for SSE and 0.15g for OBE. These are maximum free-field ground accelerations at the site as measured at the existing grade level near the ABWR. The response spectra are presented in Subsection 3.7.1. The range of site parameters used to establish the design basis seismic parameters is presented in Appendix 3A.

3.7.6 References

1. General Electric Company BWR/6-238 *Standard Safety Analysis Report (GESSAR)*, Docket No. STN 50-447, November 7, 1975.

2. E. H. Vanmarcke and C. A. Cornell, *Seismic Risk and Design Response Spectra*, ASCE Specialty Conference on Safety and Reliability of Metal Structures, Pittsburgh, Pennsylvania, November 1972.

3. NUREG-0800, *Standard Review Plan*, Section 3.7.1.

4. L. K. Liu, *Seismic Analysis of the Boiling Water Reactor*, symposium on seismic analysis of pressure vessel and piping components, First National Congress on Pressure Vessel and Piping, San Francisco, California, May 1971.

Table 3.7-1

DAMPING FOR DIFFERENT MATERIALS

Item	Percent Critical Damping OBE	Percent Critical Damping SSE
Reinforced concrete structures	4	7
Welded structural assemblies	2	4
Steel frame structures	2	4
Bolted or riveted structural assemblies	4	7
Equipment	2	3
piping systems		
- diameter greater than 12 in.	2*	3
- diameter less than or equal to 12 in.	1*	2
Reactor pressure vessel, support skirt, shroud head and separator	2	4
Guide tubes and CRD housings	1	2
Fuel	6	6

* *Damping values of ASME Code Case N-411-1, alternative damping Values for Response Spectra Analysis of Class 1, 2, and 3 Piping, Section III, Division 1, may be used as permitted by Regulatory Guide 1.84. These damping values are applicable in analyzing piping response for Seismic and other dynamic loads filtering through building structures in high frequencies range beyond 33 Hz.*

Table 3.7-2

NATURAL FREQUENCIES OF THE REACTOR BUILDING COMPLEX IN X DIRECTION (0^0-180^0 AXIS) - FIXED BASE CONDITION

Mode No.	Frequency (HZ)
1	3.97
2	4.53
3	7.70
4	8.11
5	9.17
6	11.57
7	13.64
8	13.89
9	15.02
10	15.31
11	15.79
12	16.26
13	16.82
14	18.00
15	19.73
16	20.42
17	21.08
18	22.05
19	23.11
20	24.61
21	26.27
22	27.29
23	28.17
24	28.51
25	29.38
26	31.10
27	32.04
28	32.22
29	32.58

Table 3.7-3

NATURAL FREQUENCIES OF THE REACTOR BUILDING COMPLEX IN Y DIRECTION (90°-270° AXIS) - FIXED BASE CONDITION

Mode No.	Frequency (HZ)
1	3.81
2	4.52
3	7.03
4	7.65
5	7.73
6	8.65
7	11.57
8	13.02
9	13.67
10	14.17
11	15.32
12	15.91
13	16.68
14	16.82
15	18.00
16	19.25
17	19.74
18	21.24
19	22.14
20	23.75
21	24.58
22	26.15
23	26.66
24	27.83
25	29.59
26	29.90
27	31.10
28	31.63
29	32.22

Table 3.7-4

NATURAL FREQUENCIES OF THE REACTOR BUILDING COMPLEX IN Z DIRECTION (VERTICAL) - FIXED BASE CONDITION

Mode No.	Frequency (HZ)
1	5.07
2	5.176
3	5.183
4	8.44
5	9.20
6	9.23
7	12.80
8	13.37
9	19.60
10	27.54
11	31.36

Table 3.7-5

NATURAL FREQUENCIES OF THE CONTROL BUILDING - FIXED BASE CONDITION

Mode No.	Frequency (HZ)	Direction
1	5.42	X HORIZ
2	6.72	Y HORIZ
3	13.30	Z VERT
4	18.55	X HORIZ
5	24.81	Y HORIZ
6	31.59	Y HORIZ
7	33.61	X HORIZ

Table 3.7-6

NUMBER OF DYNAMIC RESPONSE CYCLES EXPECTED DURING A SEISMIC EVENT FOR SYSTEMS & COMPONENTS

	FREQUENCY BANDWIDTH (Hz)		
	0^{+}-10	10-20	20-50
Total number of seismic cycles	168	359	643
No. of seismic cycles (0.5% of total) between 75% and 100% of peak loads	0.8	1.8	3.2
No. of seismic cycles (4.5% of total) between 50% and 75% of peak loads	7.5	16.2	28.9

Table 3.7-7

DESCRIPTION OF SEISMIC INSTRUMENTATION

Component	Location	Elevation*	Setpoint (g)	Operating Range
Time-history accelerometer sensor	Free field, 160 M from Reactor Building RB	N/A	-	0.01 to 1.0g
Time-history accelerometer sensor	Reactor building foundation mat at base an RB clean zone	(-) 13.2 M	-	0.01 to 1.0g
Time-history accelerometer sensor	At RB clean zone	(+) 26.7 M	-	0.01 to 1.0g
Seismic trigger	Free field, 160 M from Reactor Building	N/A	0.01	0.005 to 0.02g
Seismic trigger	Free field, 160 M from Reactor Building	N/A	0.01	0.005 to 0.02g
Peak recording accelerograph	Reactor Building, RWCU regenerative heat exchanger support	(-) 6.7 M	-	1 to 20 Hz
Peak recording accelerograph	Reactor Building, RHR line	-	-	1 to 20 Hz
Peak recording accelerograph	Reactor Building, Diesel generator A support	(+) 7.3 M	-	1 to 20 Hz
Seismic switch	Reactor Building foundation	(-) 13.2 M	0.10	0.1 to 30.0 Hz
Response spectrum recorder, (active)	Reactor Building foundation mat at the base of an RB clean zone	(-) 13.2 M	Table 3.7-8	Table 3.7-8
Response spectrum recorder, (active)	Control Building foundtion mat	(-) 3.5 M	Table 3.7-8	Table 3.7-8
Response spectrum recorder, (passive)	Reactor Building pipe tunnel RHR hanger	-	-	1.0 to 32 Hz
Response spectrum recorder, (passive)	Reactor Building FMCRD control panel support	(+) 18.7 M	-	1.0 to 32 Hz
Seismic event - recording alarm, playback panel	Control equipment room	-	-	-

* *Elevations are with respect to the RPV bottom head.*

Table 3.7-8

SET POINTS FOR ACTIVE RESPONSE SPECTRUM RECORDERS

Setpoint (g)		Operating Range
Horizontal	Vertical	Frequen (Hz)
0.19	0.13	1.00
0.23	0.16	1.26
0.29	0.20	1.59
0.35	0.25	2.00
0.48	0.30	2.52
0.41	0.37	3.17
0.40	0.39	4.00
0.39	0.38	5.04
0.37	0.37	6.35
0.36	0.36	8.00
0.32	0.32	10.01
0.26	0.26	12.07
0.21	0.21	20.00
0.17	0.17	20.20
0.13	0.13	24.40
0.10	0.10	32.00

Table 3.7-9

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK[a]	CHANNEL CALIBRATION[a]	CHANNEL FUNCTIONAL TEST[a]
1. Triaxial Time-History Accelerographs	M	R	SA
2. Triaxial Peak Accelerographs	NA	R	NA
3. Triaxial Seismic Switches	M	R	SA
4. Triaxial Response-Spectrum Recorders	M	R	SA

[a]M = Monthly
R = Refueling
SA = One per 18 months
NA = Not Applicable

Table 3.7-10

NATURAL FREQUENCIES OF THE RADWASTE BUILDING - FIXED BASE CONDITION

Mode No.	Frequency (HZ)	Direction
1	4.66	Y HORIZ
2	5.61	X HORIZ
3	11.47	Z VERT
4	11.74	Y HORIZ
5	14.29	X HORIZ
6	18.42	Y HORIZ
7	22.39	X HORIZ
8	23.01	Y HORIZ
9	28.13	X HORIZ
10	28.61	Z VERT

Table 3.7-11

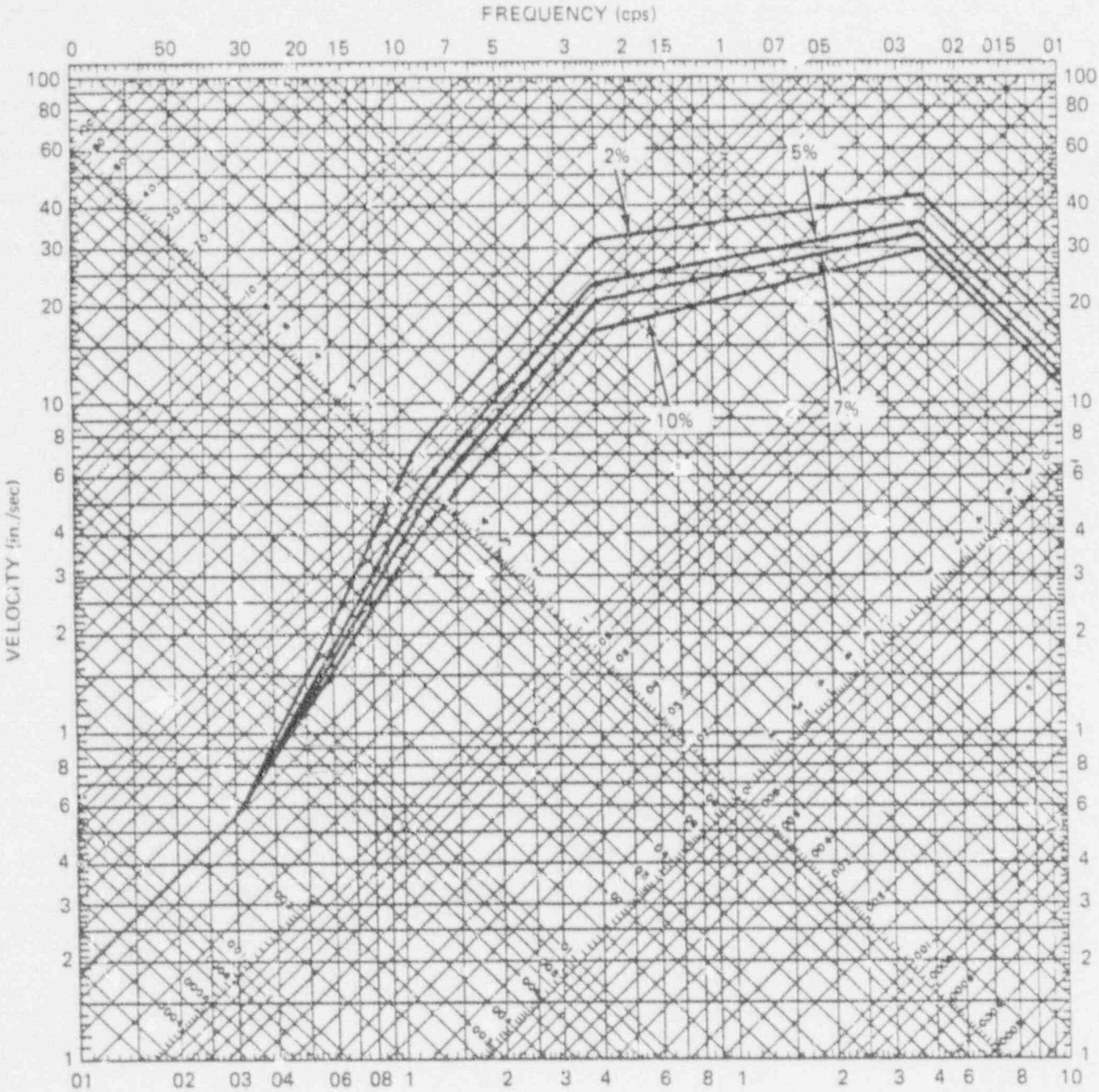
SITE COEFFICIENTS

Type	Description	S Factor
S_1	A soil profile with either; (a) A rock like material characterized by a shear wave velocity greater than 2,500 fps or by other suitable means of classification. or (b) Stiff or dense soil condition where soil depth is less than 200 ft.	1.0
S_2	A soil profile with dense or stiff soil conditions, where the soil depth exceeds 200 feet.	1.2
S_3	A soil profile 40 feet or more in depth and containing more than 20 feet of soft to medium stiff clay but not more than 40 feet of soft clay.	1.5
S_4	A soil profile containing more than 40 feet of soft clay.	2.0

Table 3.7-12

STRUCTURAL SYSTEMS

Basic Structural	Lateral Load Resisting System Description	R_w
A Bearing wall	1. Shear walls - concrete	6
	2a. Braced frames where bracing carries gravity loads - steel	6
	2b. Braced frames where bracing carries gravity loads - concrete	4
B Building frame	1. Steel eccentric braced frame	10
	Shear walls - concrete	8
	Concentric braced frames - steel	8
	Concentric braced frames - concrete	8
C Moment resisting frame	Special moment resisting space frames	12
	Concrete intermediate moment-resisting space frames (OMRSF)	7
	Ordinary moment resisting space frames (OMRSF) - steel	6
	Ordinary moment resisting space frames (OMRSF) - concrete	5
D Dual		
1. Shear walls	a. Concrete with SMRSF	12
	b. Concrete with concrete IMRSF	9
2. Steel EBF with steel SMRSF		12
3. Concentric braced frames	a. Steel with steel SMRSF	10
	b Concrete with concrete SMRSF	9
	c. Concrete with concrete IMRSF	6



87-592-27

Figure 3.7-1 HORIZONTAL SAFE SHUTDOWN EARTHQUAKE DESIGN SPECTRA

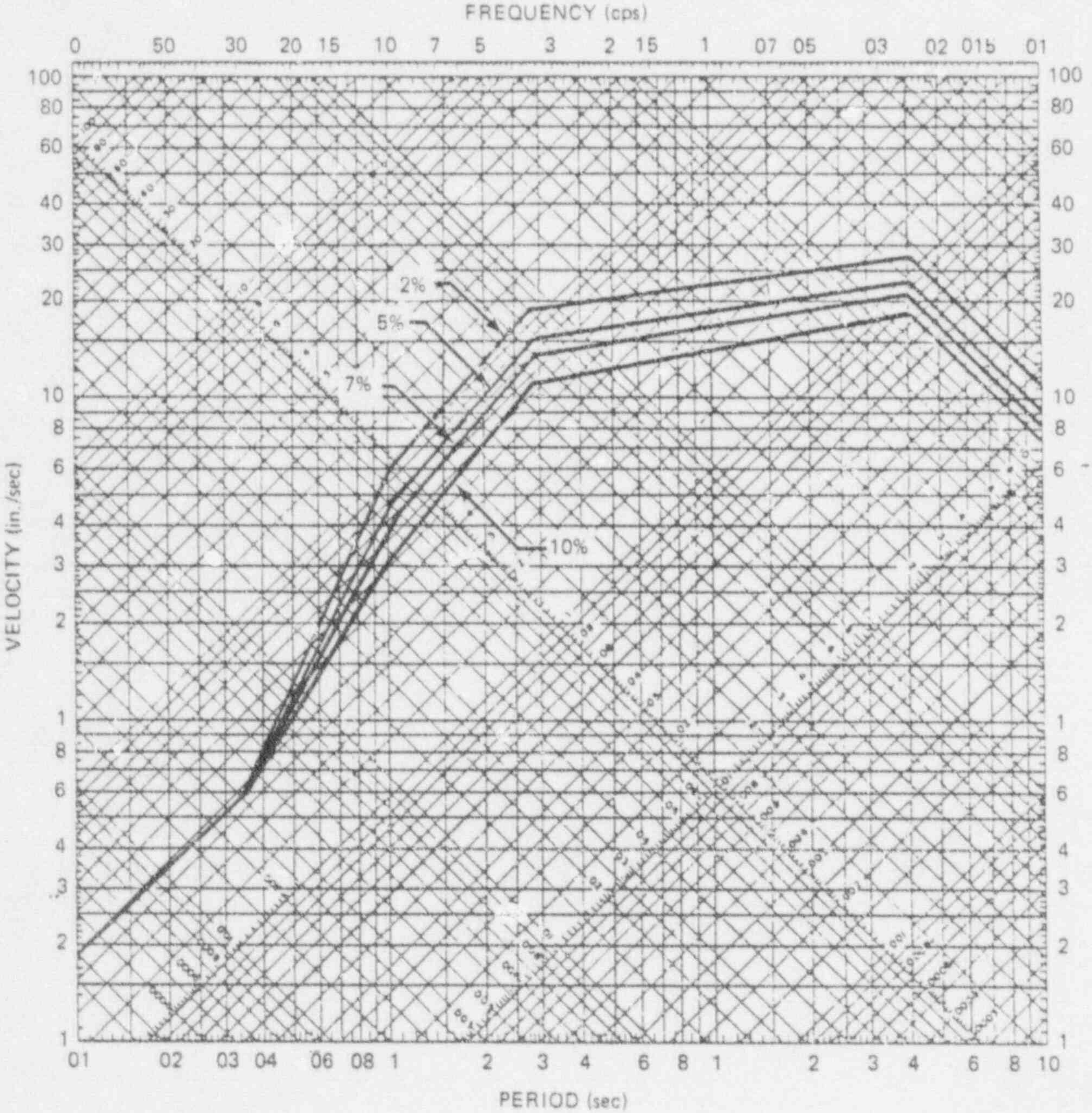
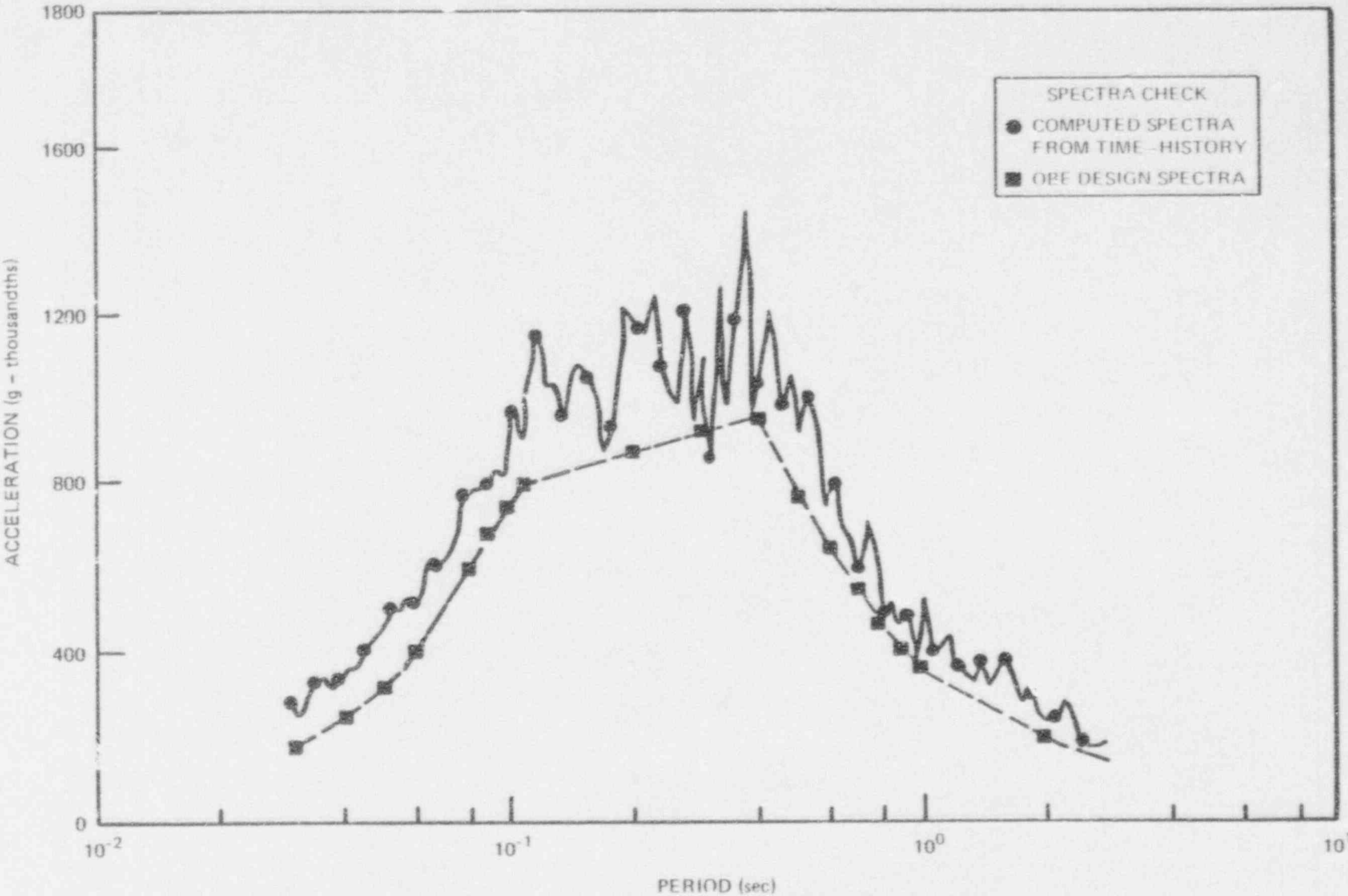


Figure 3.7-2 VERTICAL SAFE SHUTDOWN EARTHQUAKE DESIGN SPECTRA



87-592-29

Figure 3.7-3 SYNTHETIC TIME HISTORY, FIRST HORIZONTAL DIRECTION, DAMPING RATIO 0.01

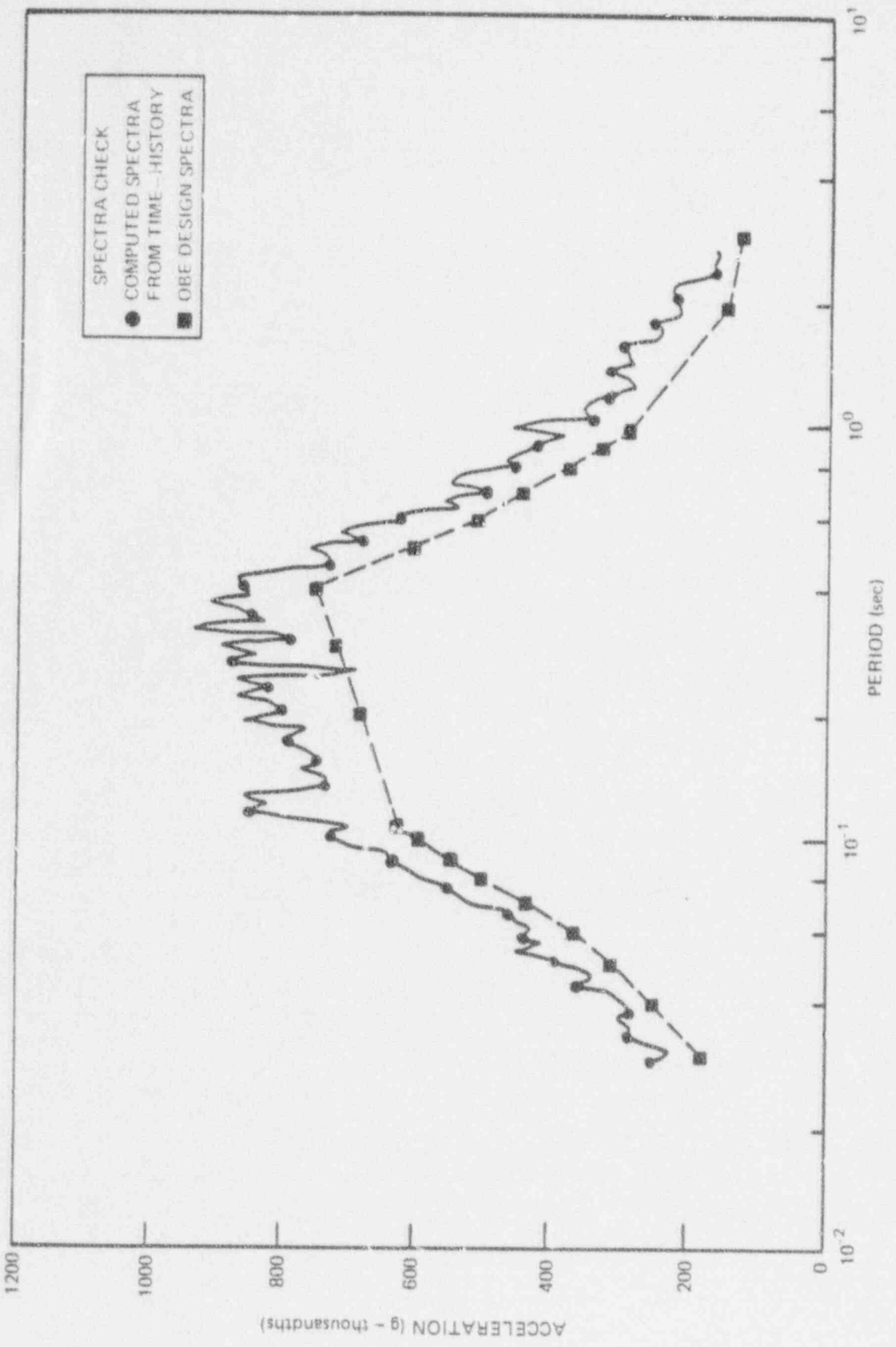


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Figure 3.7–4 SYNTHETIC TIME HISTORY, FIRST HORIZONTAL DIRECTION, DAMPING RATIO 0.02

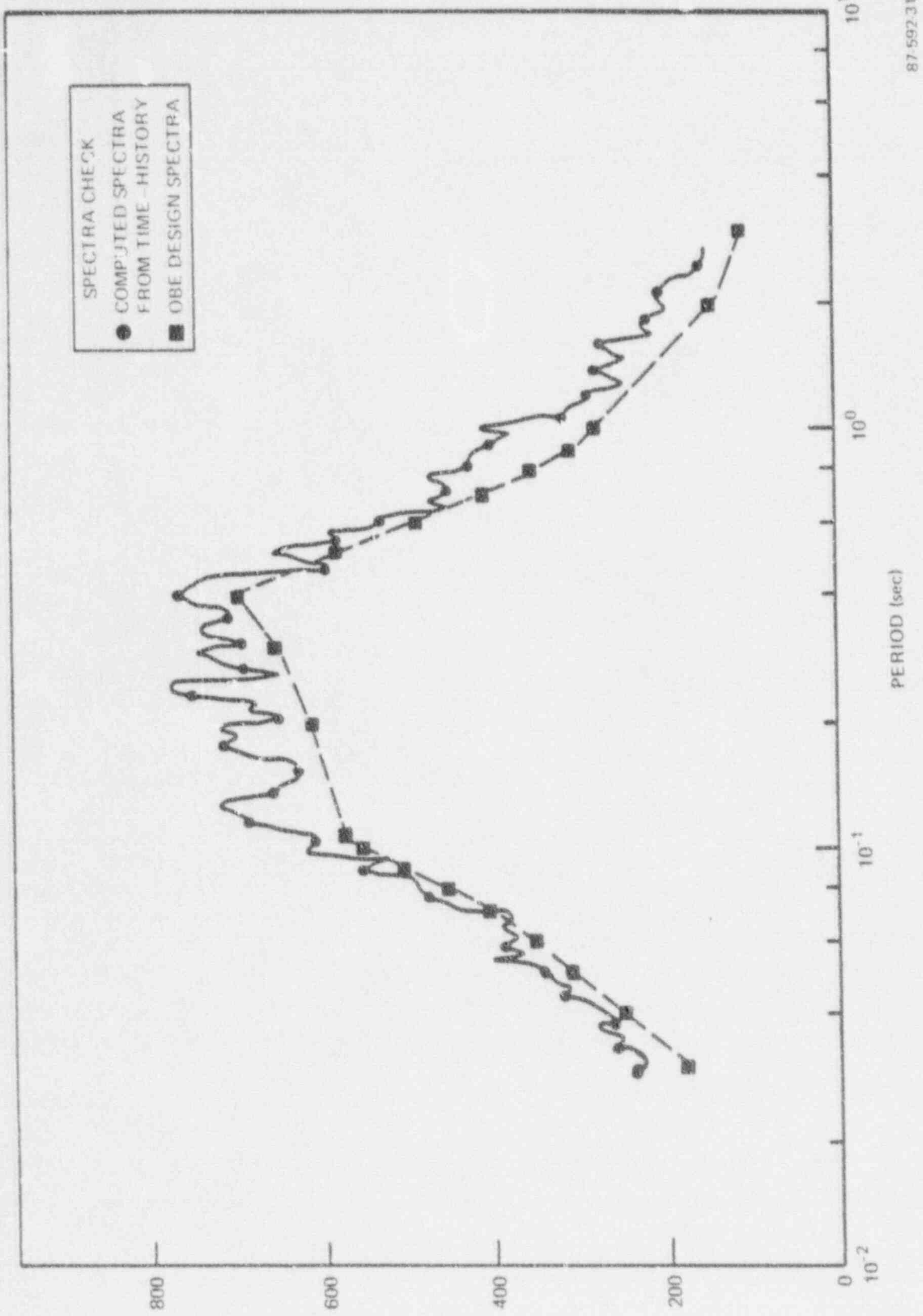

Figure 3.7-5 SYNTHETIC TIME HISTORY, FIRST HORIZONTAL DIRECTION, DAMPING RATIO 0.03

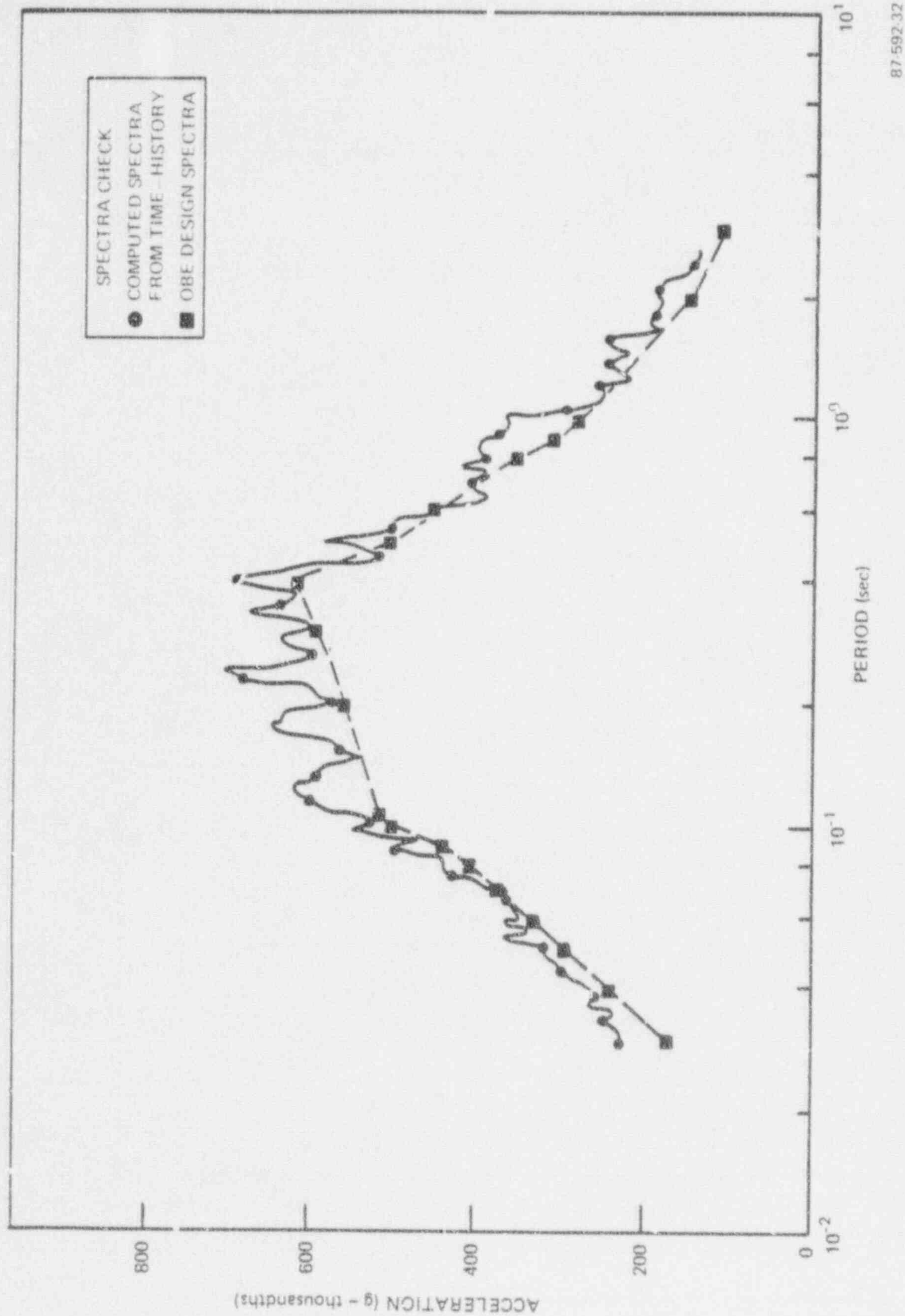

Figure 3.7-6 SYNTHETIC TIME HISTORY, FIRST HORIZONTAL DIRECTION, DAMPING RATIO 0.04

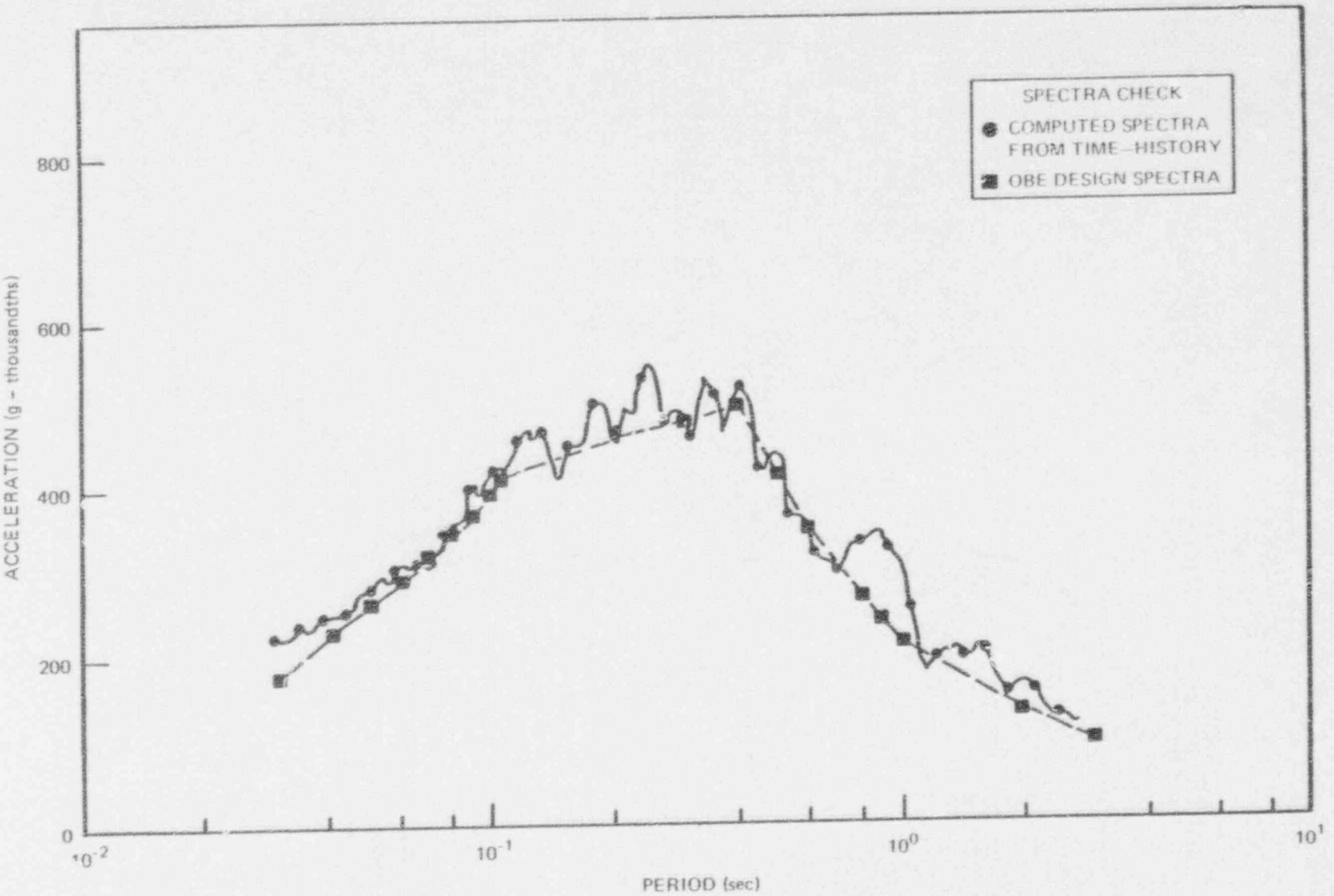

Figure 3.7–7 SYNTHETIC TIME HISTORY, FIRST HORIZONTAL DIRECTION, DAMPING RATIO 0.07

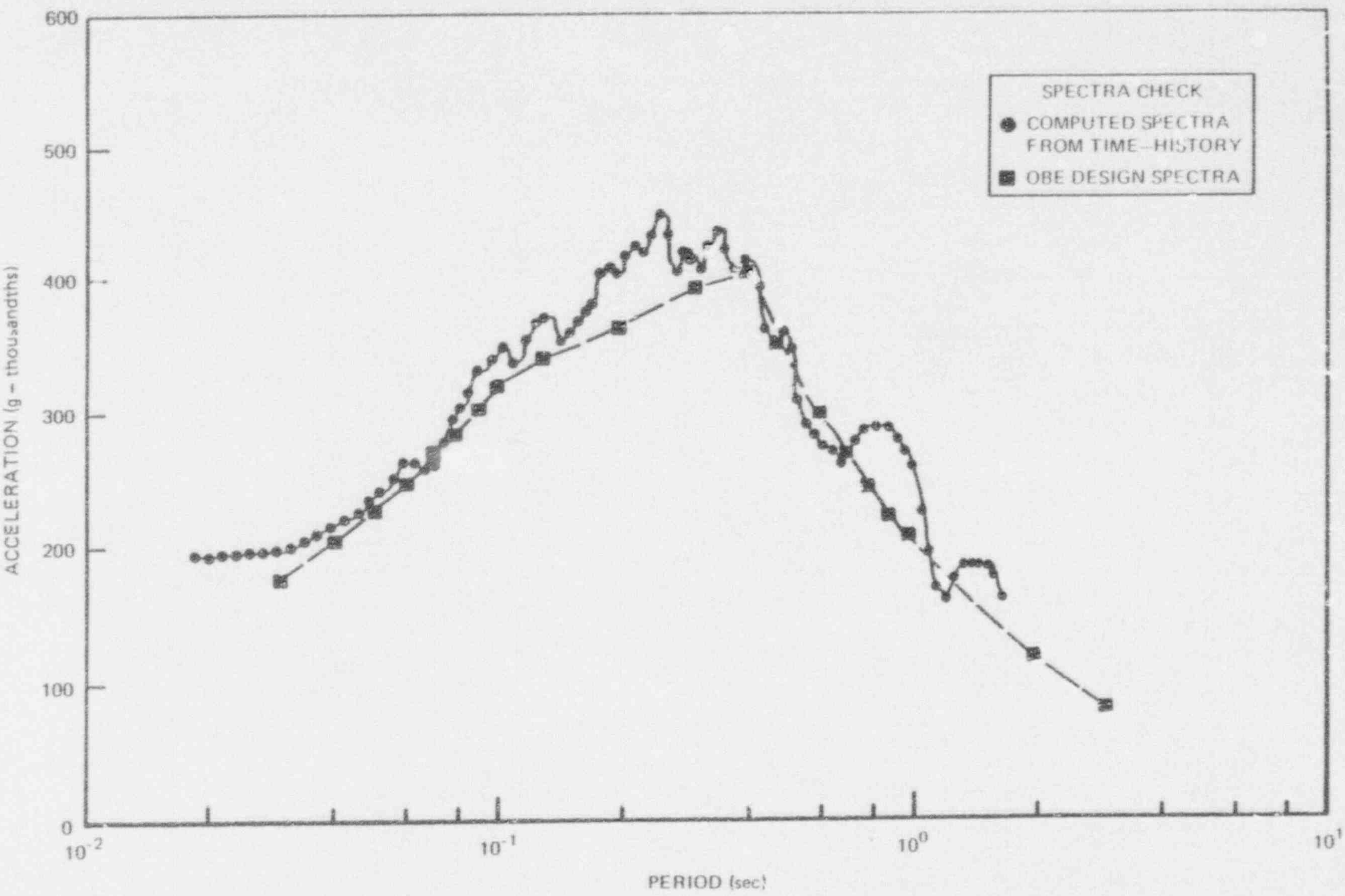

Figure 3.7–8 SYNTHETIC TIME HISTORY, FIRST HORIZONTAL DIRECTION, DAMPING RATIO 0.10

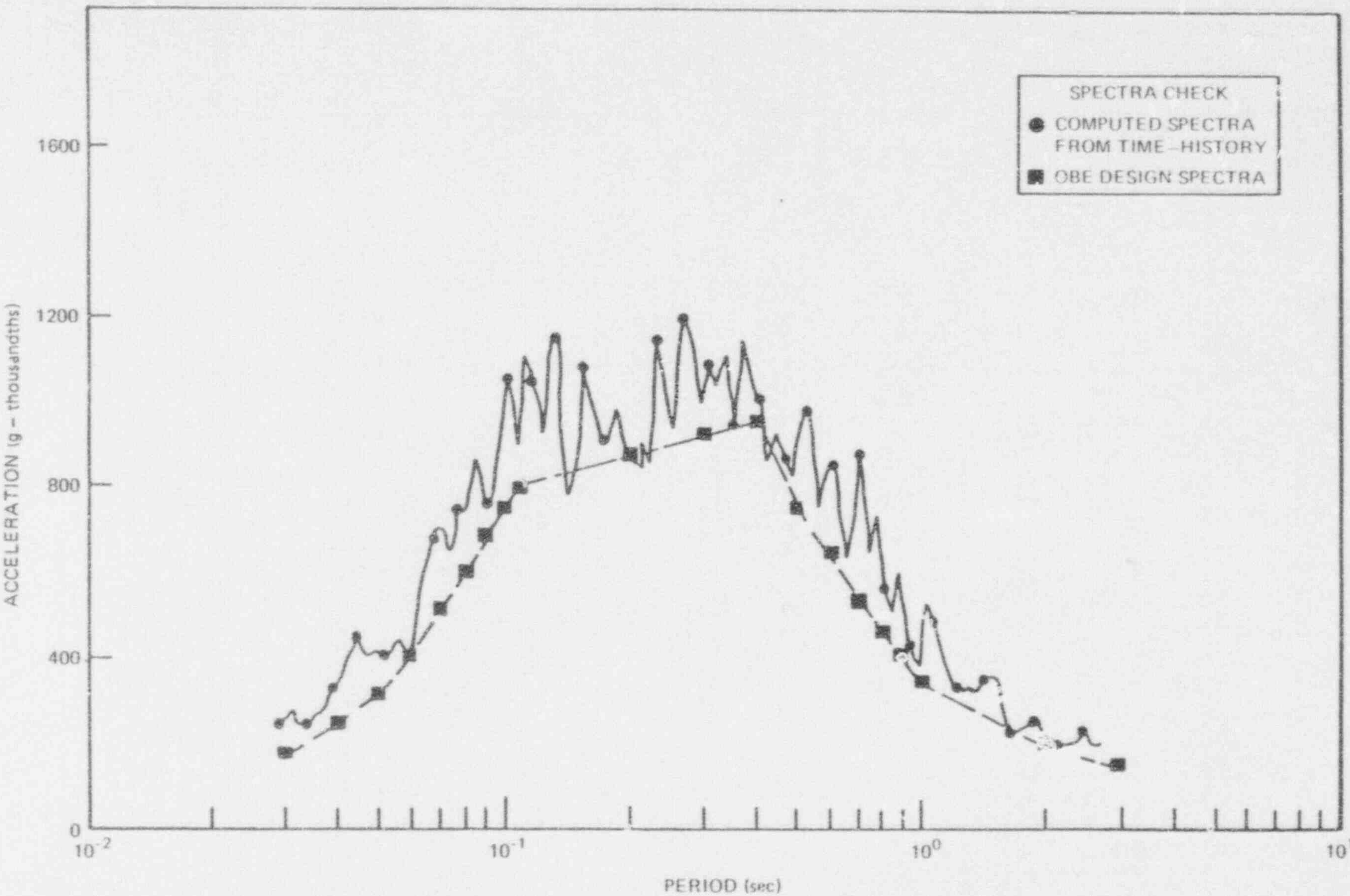

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Figure 3.7–9 SYNTHETIC TIME HISTORY, SECOND HORIZONTAL DIRECTION, DAMPING RATIO 0.01

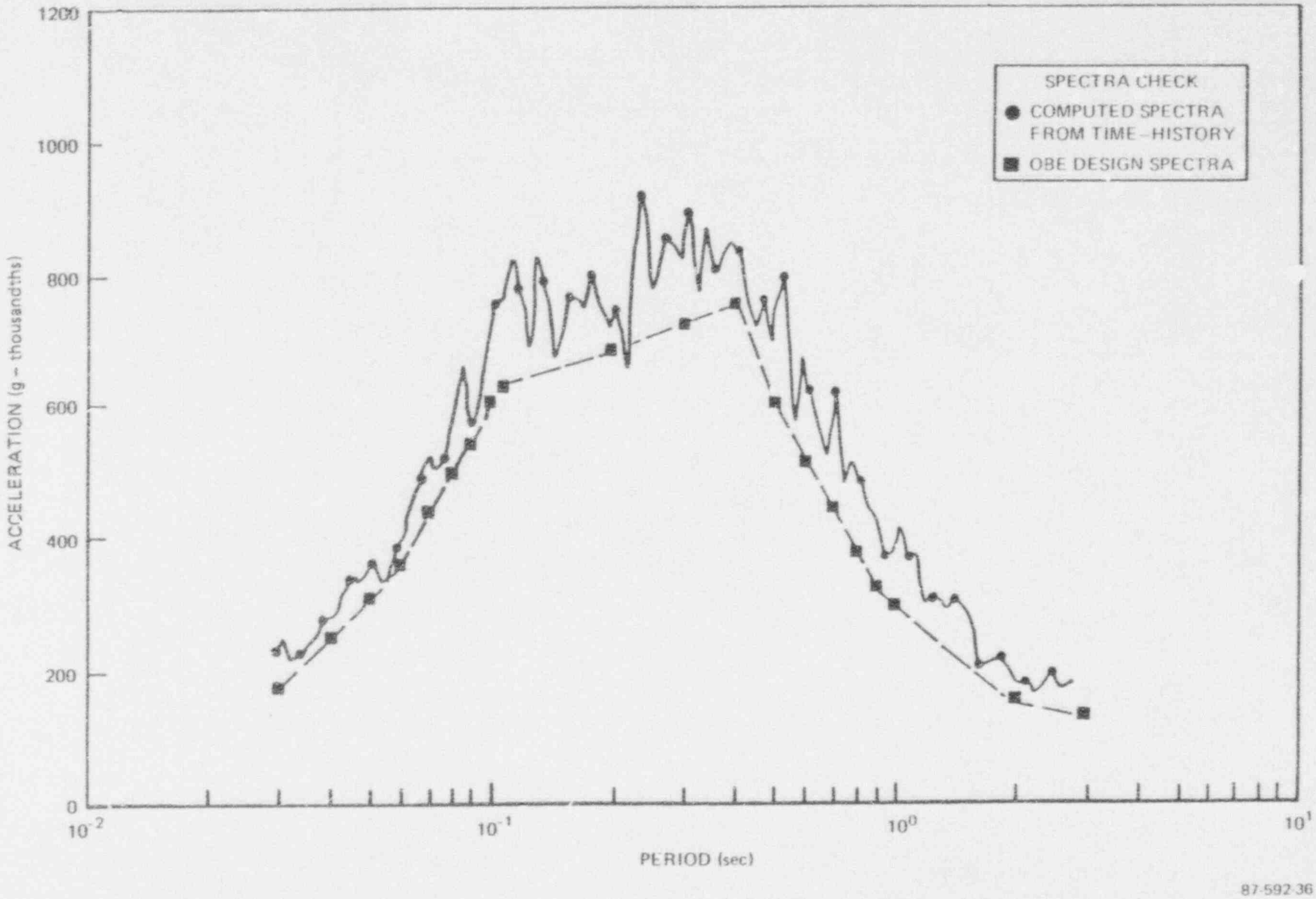

Figure 3.7-10 SYNTHETIC TIME HISTORY, SECOND HORIZONTAL DIRECTION, DAMPING RATIO 0.02

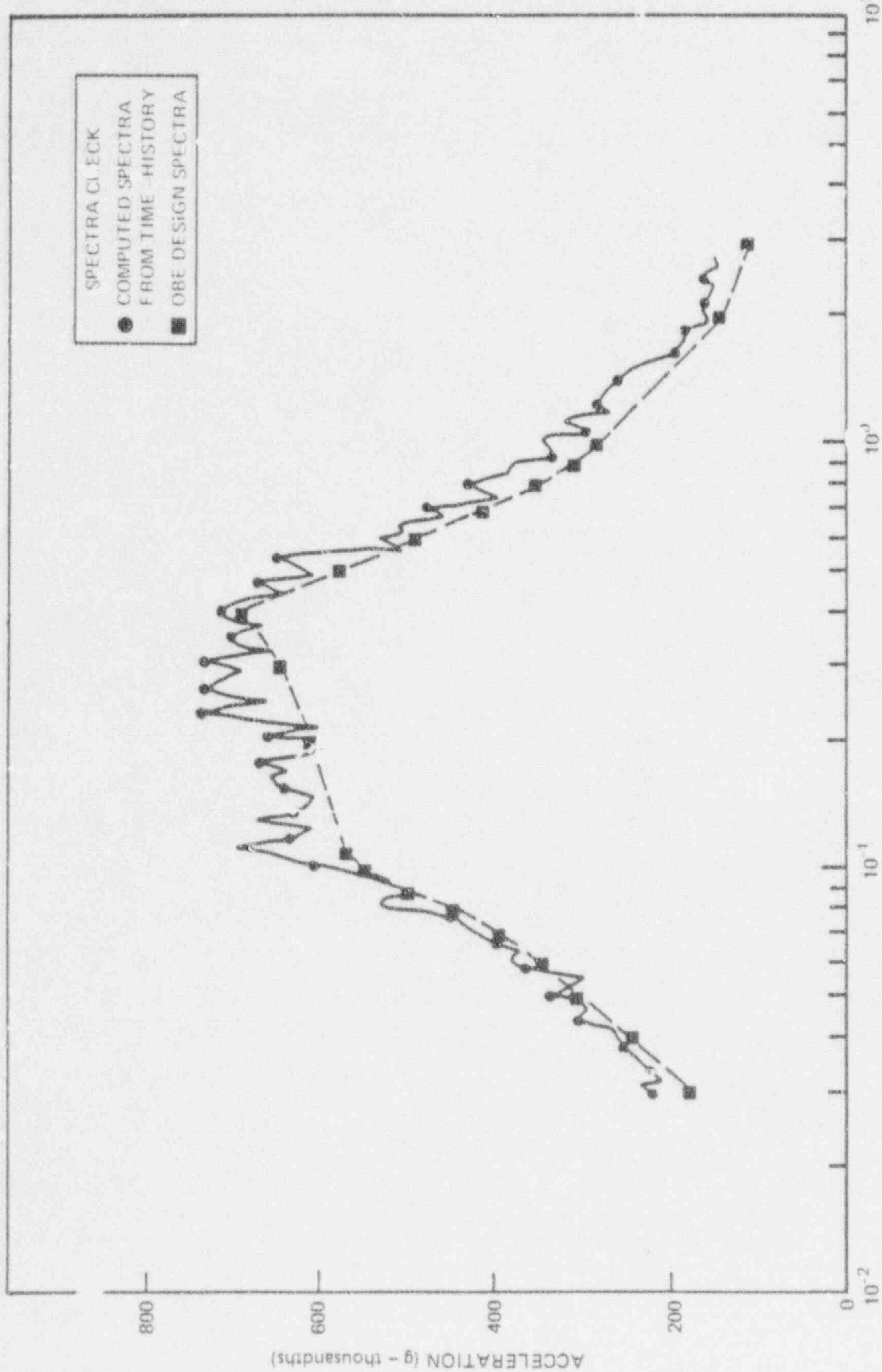

Figure 3.7–11 SYNTHETIC TIME HISTORY, SECOND HORIZONTAL DIRECTION, DAMPING RATIO 0.03

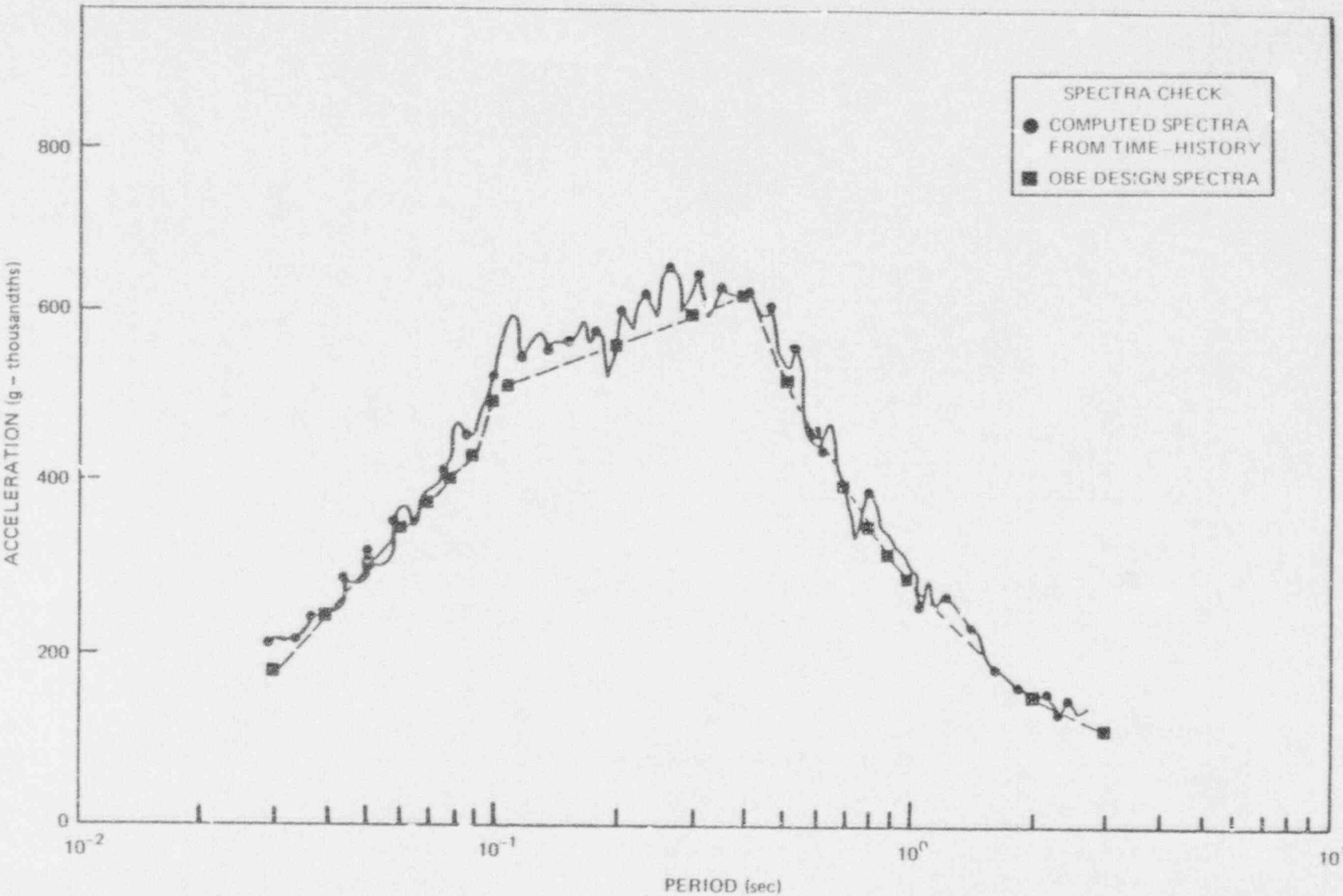

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Figure 3.7–12 SYNTHETIC TIME HISTORY, SECOND HORIZONTAL DIRECTION, DAMPING RATIO 0.04

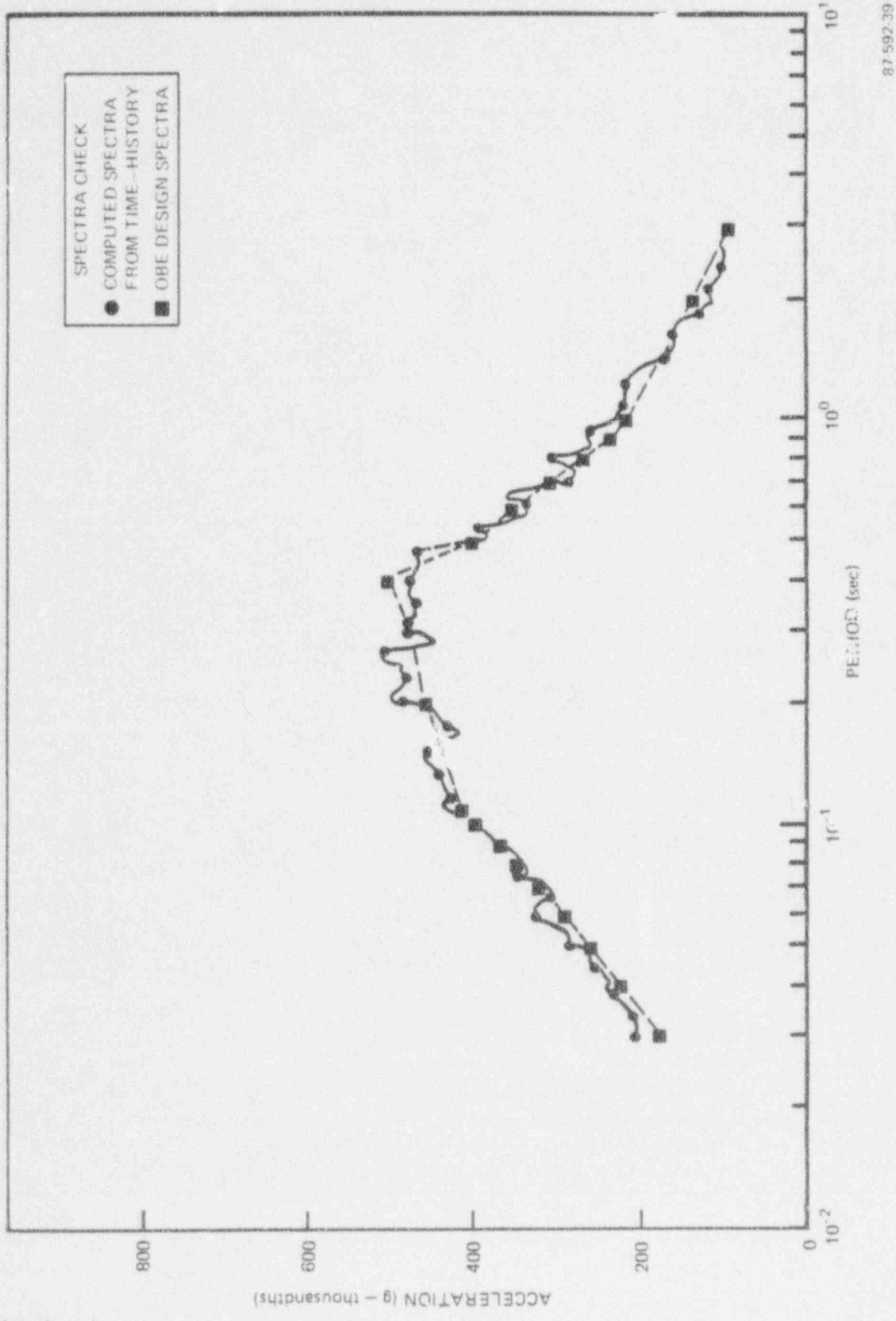

Figure 3.7-13 SYNTHETIC TIME HISTORY, SECOND HORIZONTAL DIRECTION, DAMPING RATIO 0.07

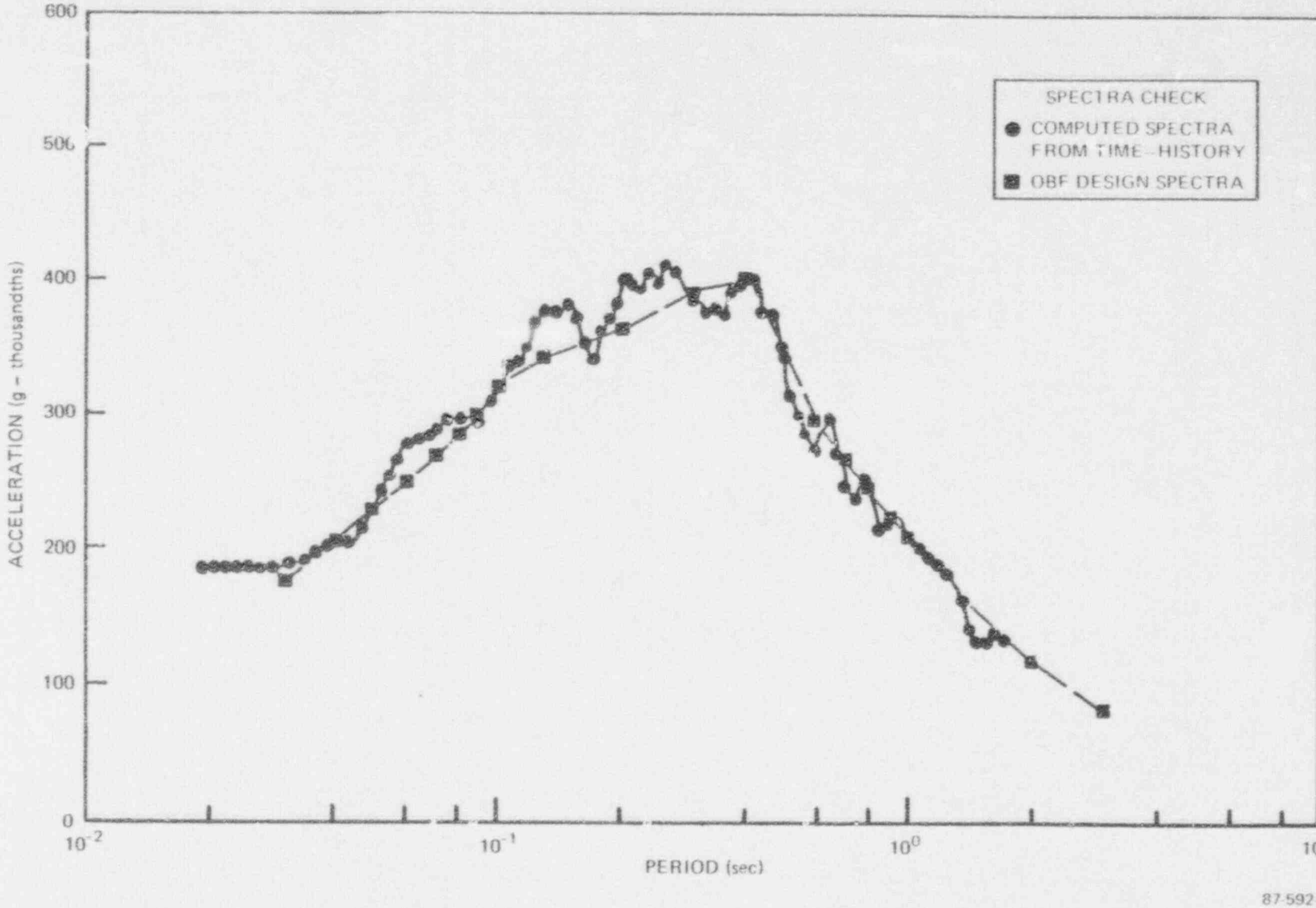

Figure 3.7–14 SYNTHETIC TIME HISTORY, SECOND [illegible]AL DIRECTION, DAMPING RATIO 0.10

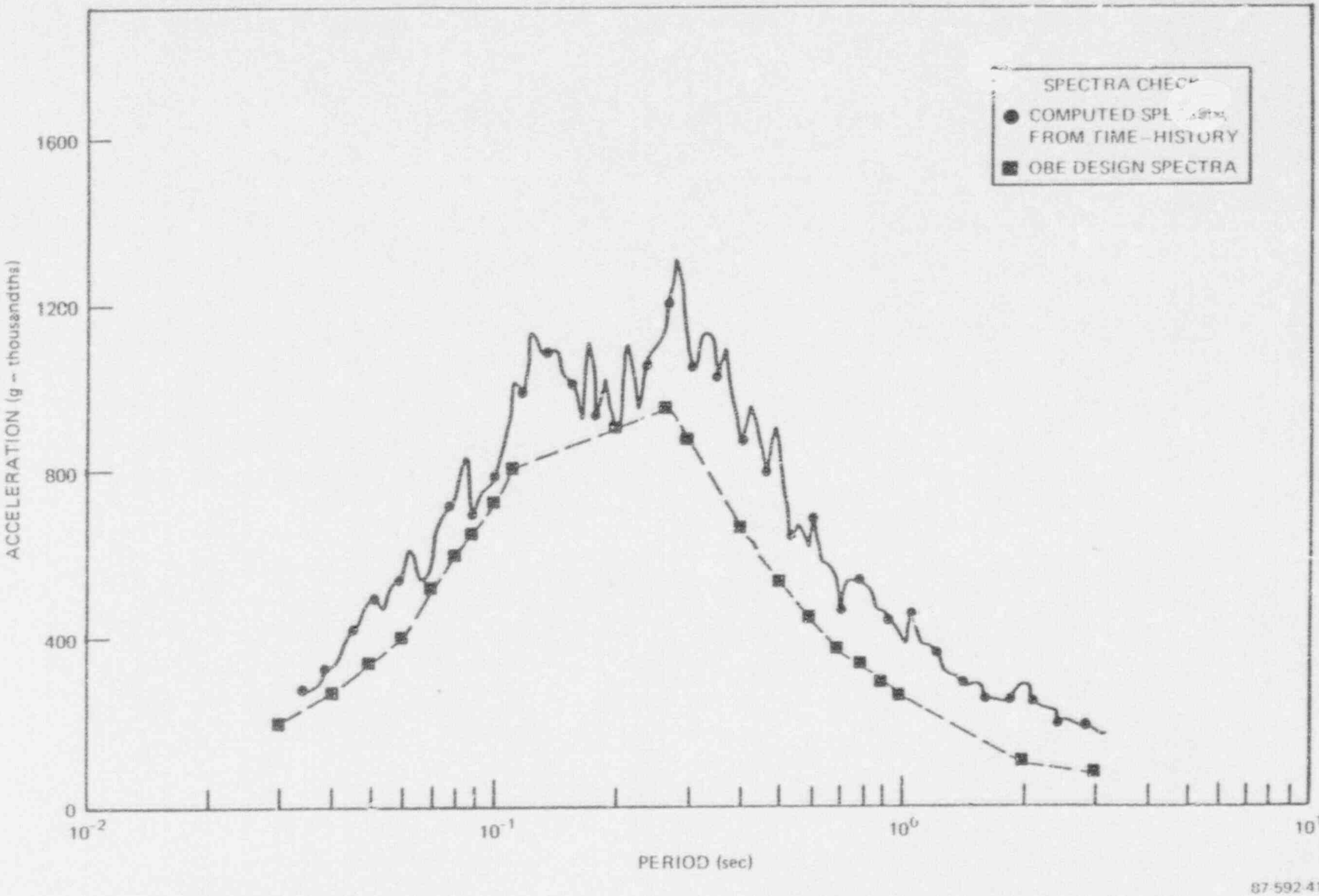

Figure 3.7-15 SYNTHETIC TIME HISTORY, VERTICAL DIRECTION, DAMPING RATIO 0.01

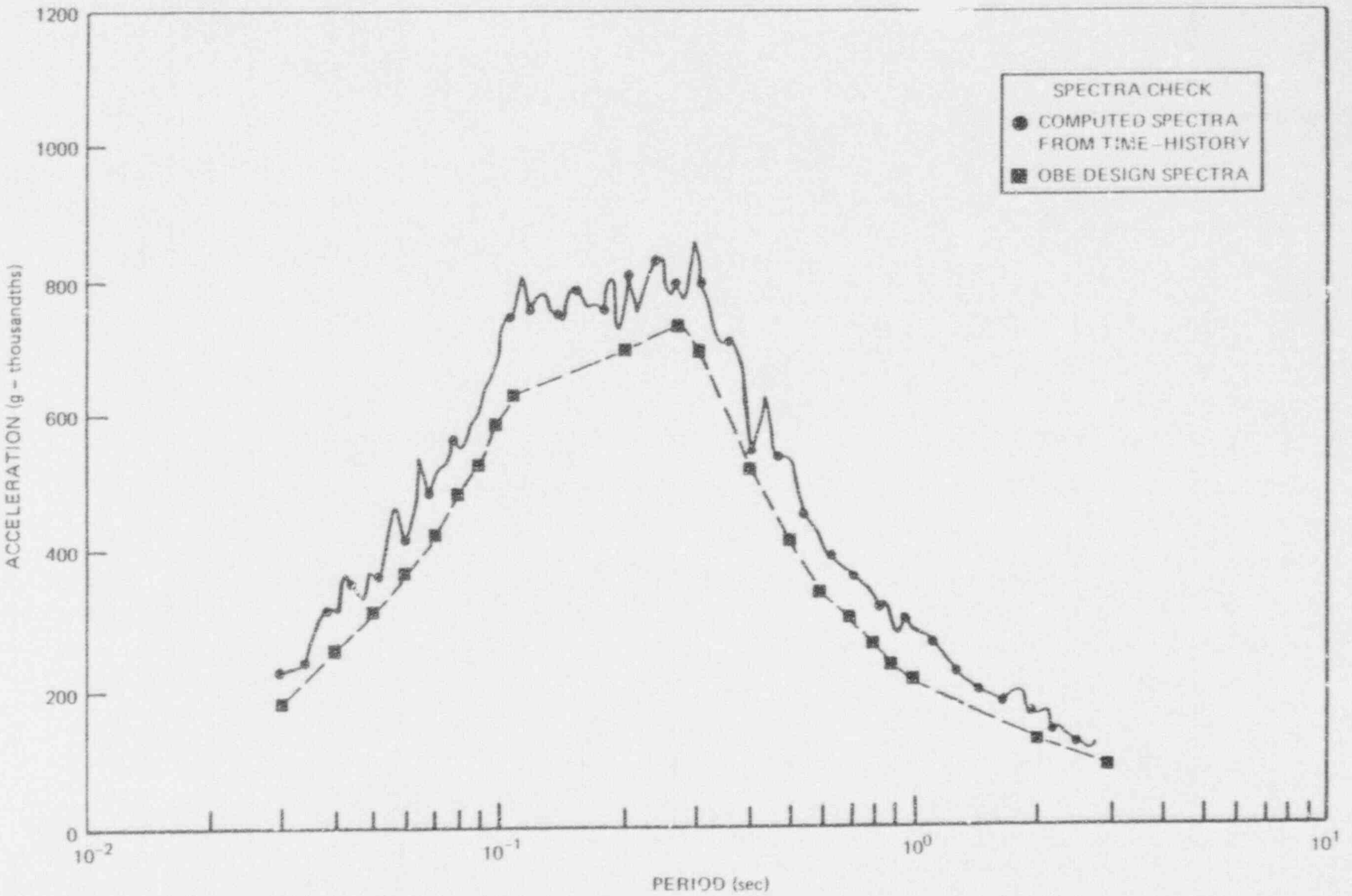

Figure 3.7–16 SYNTHETIC TIME HISTORY, VERTICAL DIRECTION, DAMPING RATIO 0.02

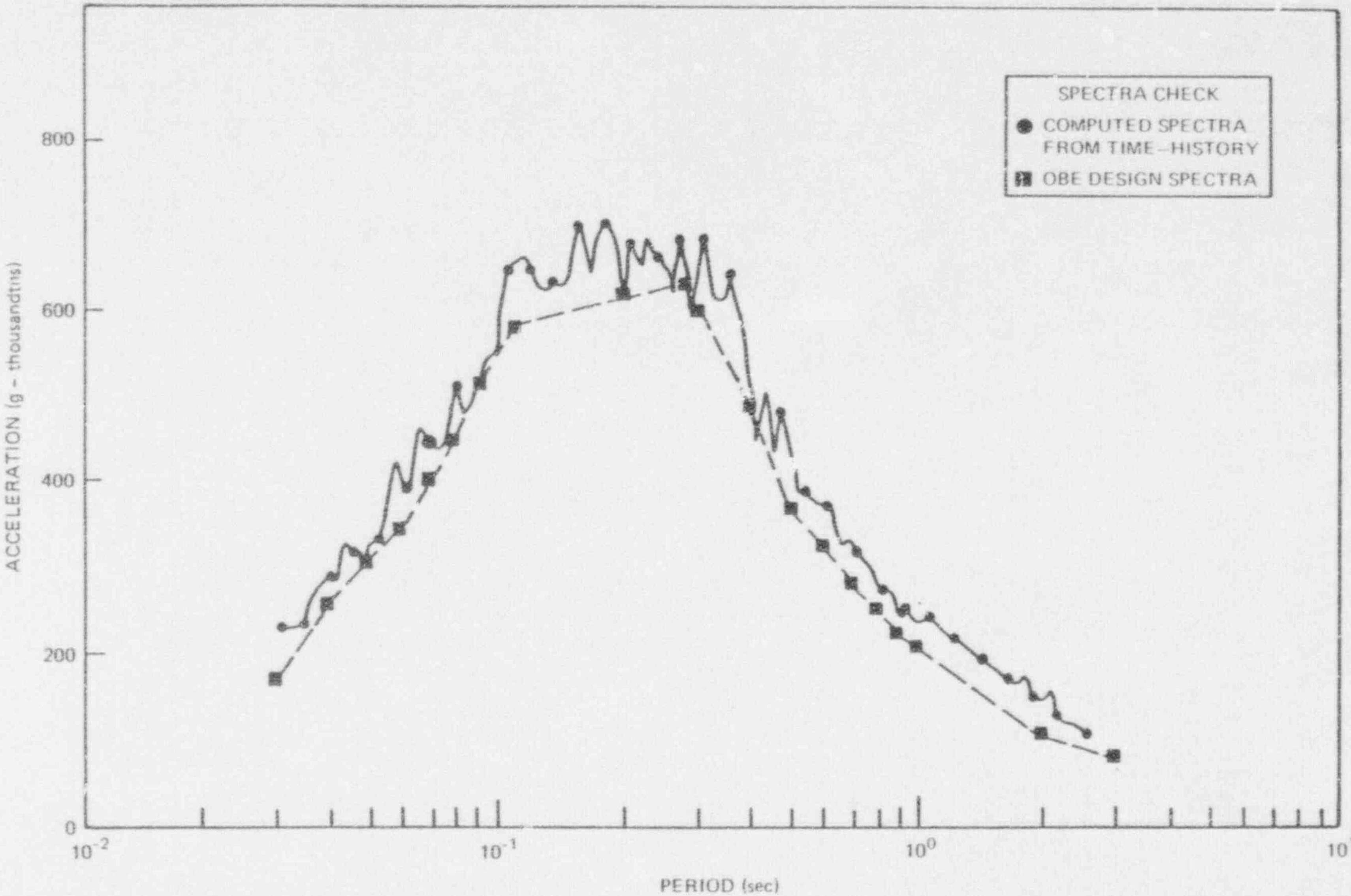

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Figure 3.7–17 SYNTHETIC TIME HISTORY, VERTICAL DIRECTION, DAMPING RATIO 0.03

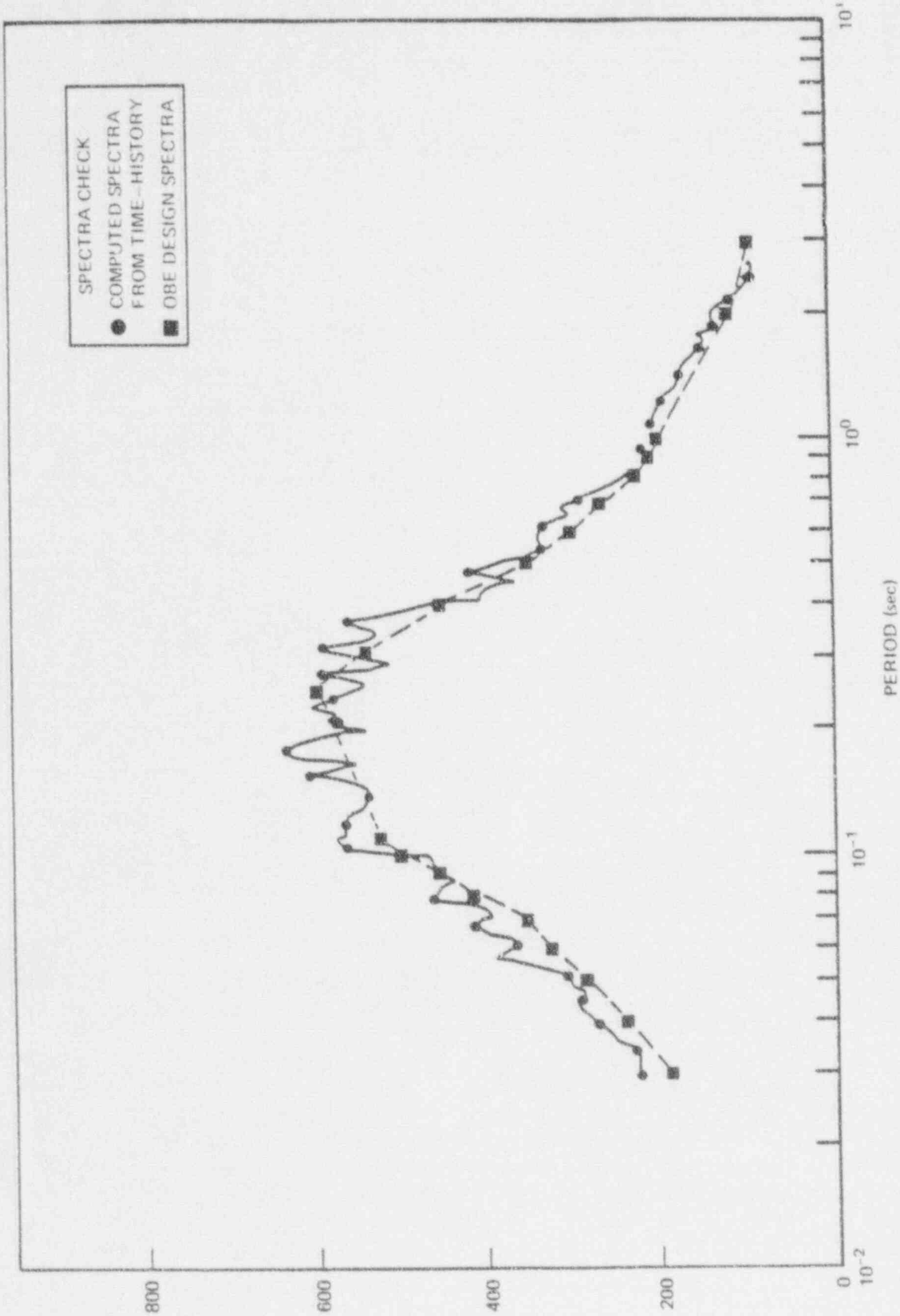

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Figure 3.7-18 SYNTHETIC TIME HISTORY, VERTICAL DIRECTION, DAMPING RATIO 0.04

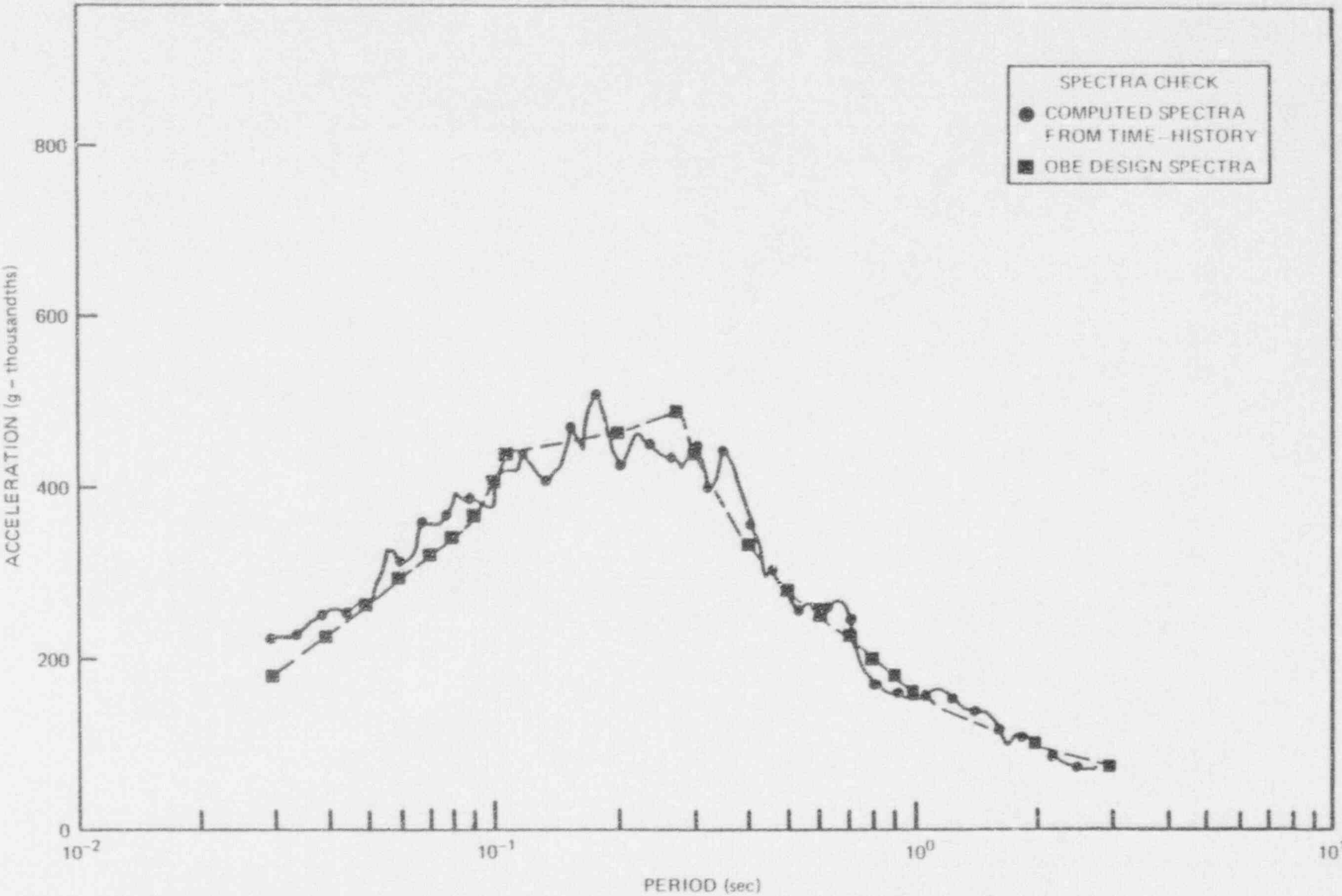

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Figure 3.7–19 SYNTHETIC TIME HISTORY, VERTICAL DIRECTION, DAMPING RATIO 0.07

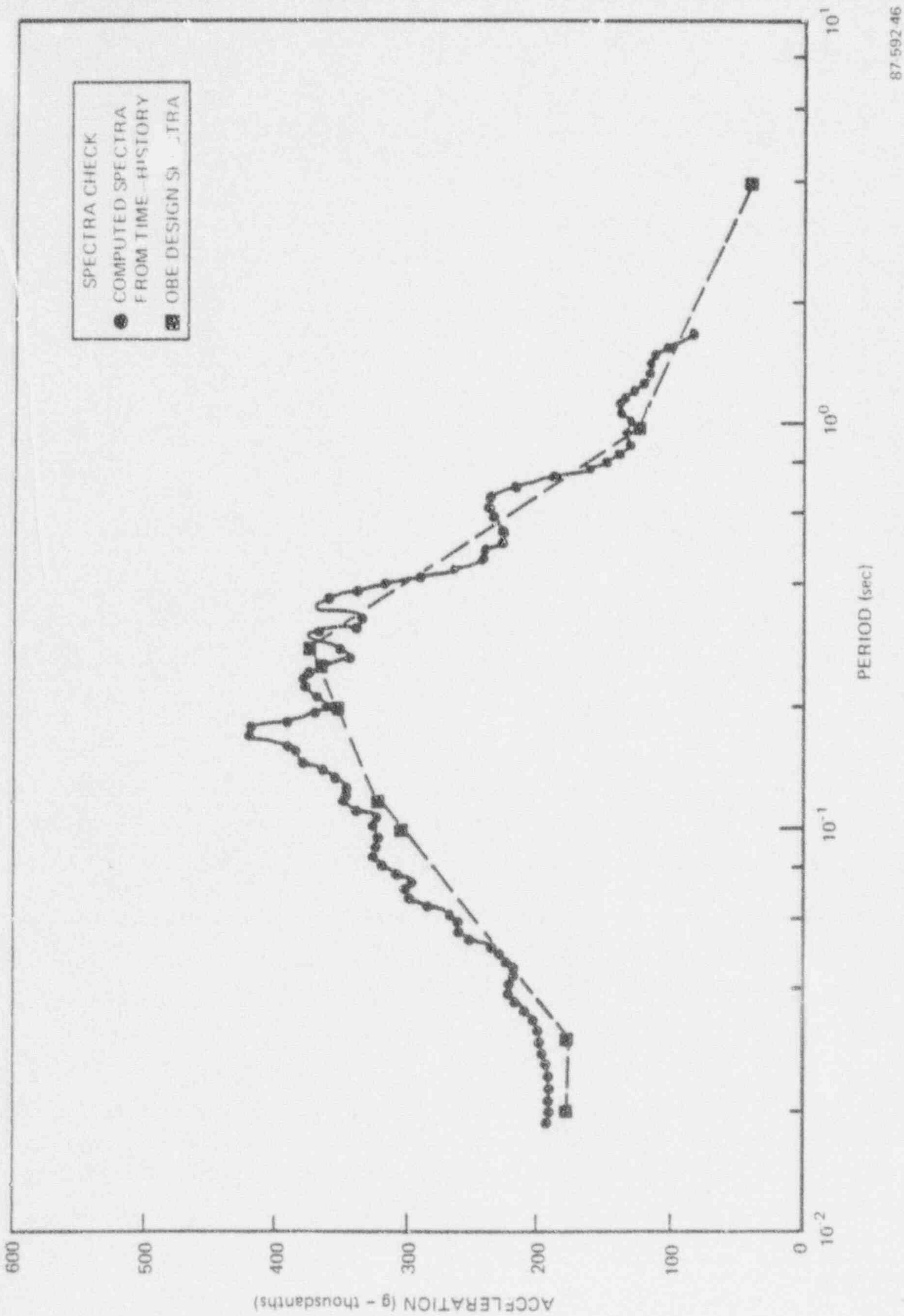

Figure 3.7-20 SYNTHETIC TIME HISTORY, VERTICAL DIRECTION, DAMPING RATIO 0.10

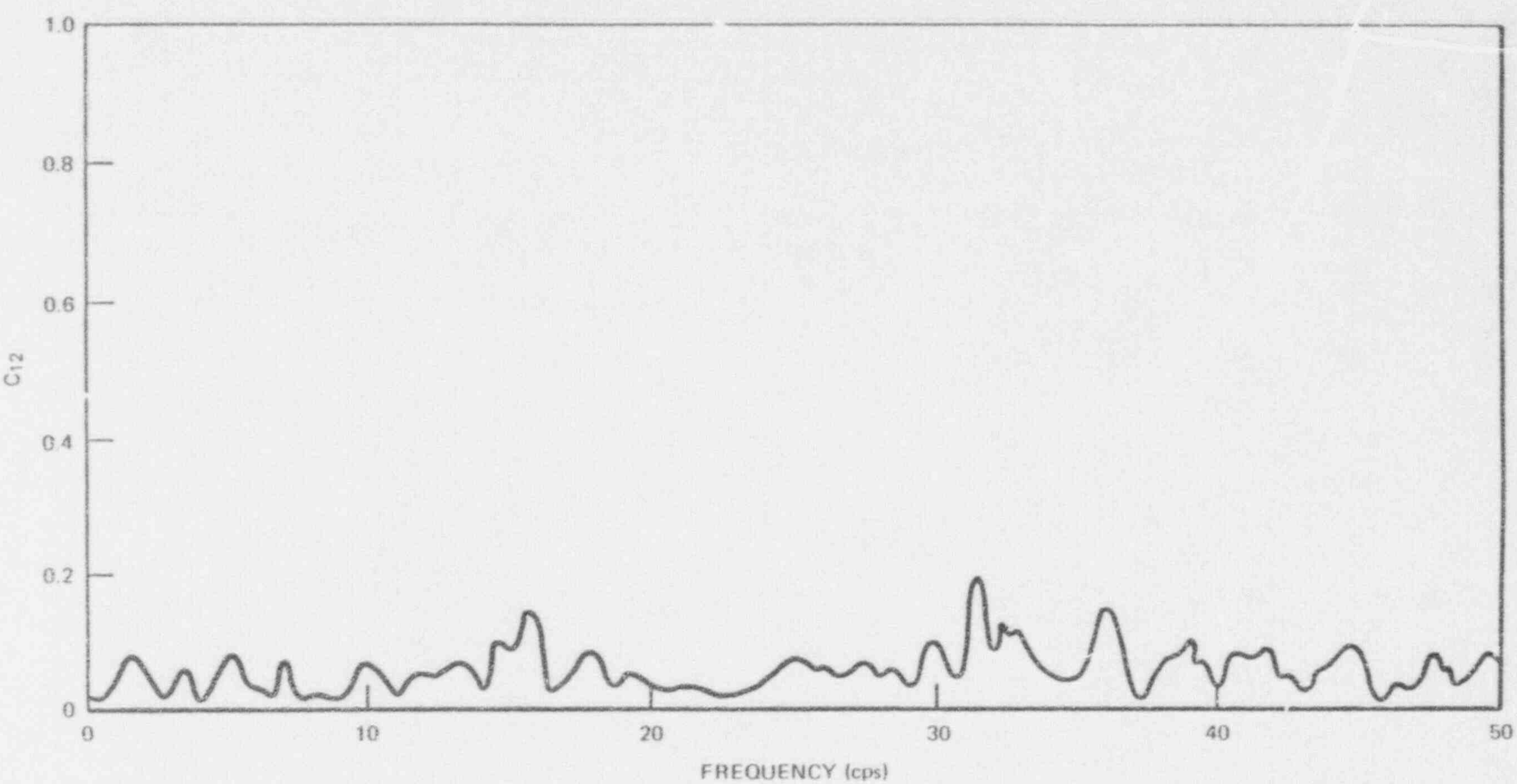

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Figure 3.7-21 COHERENCE FUNCTION C_{12} FOR EARTHQUAKE COMPONENTS H1 AND H2

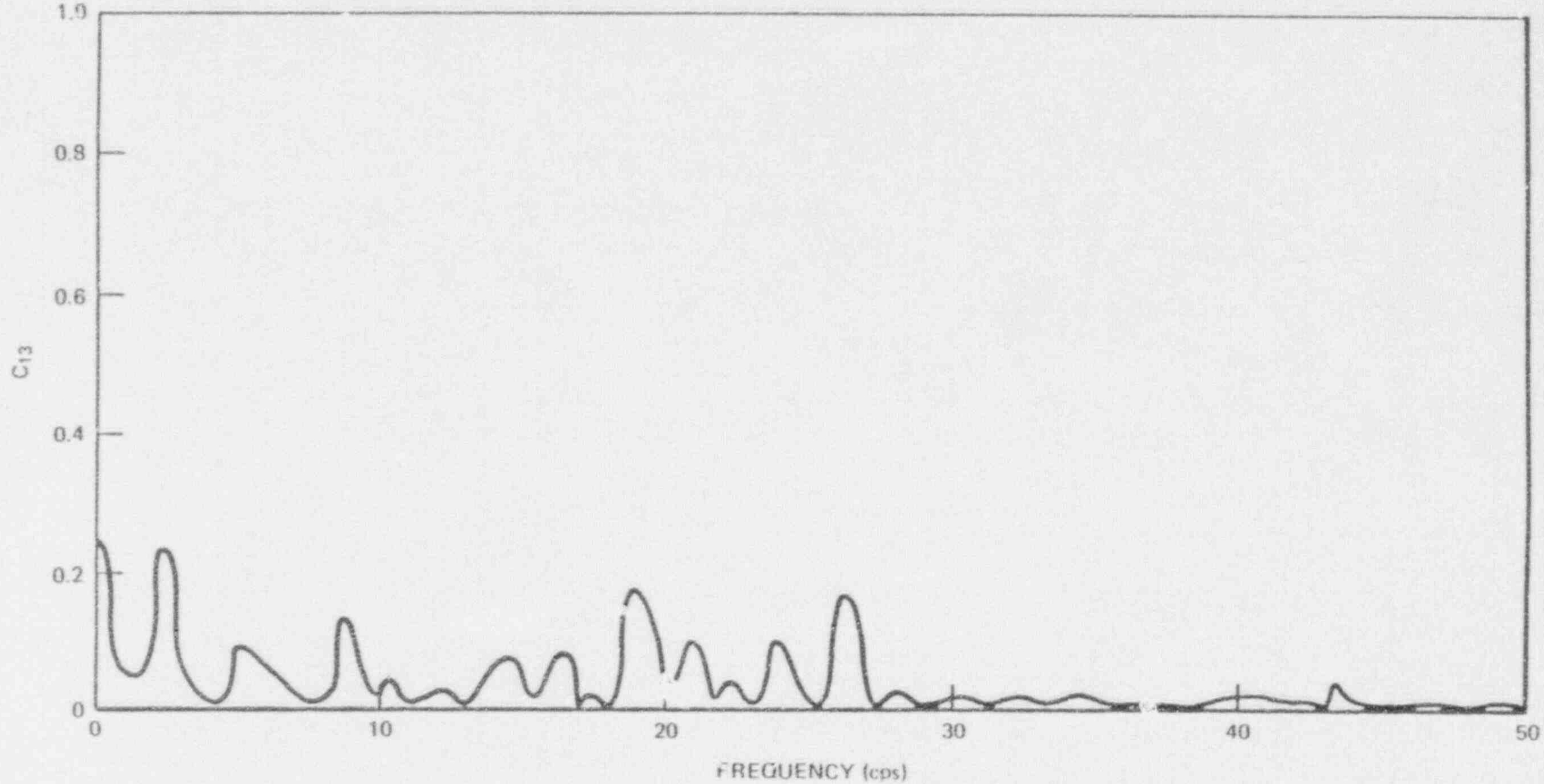

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Figure 3.7-22 COHERENCE FUNCTION C_{13} FOR EARTHQUAKE COMPONENTS H1 AND V

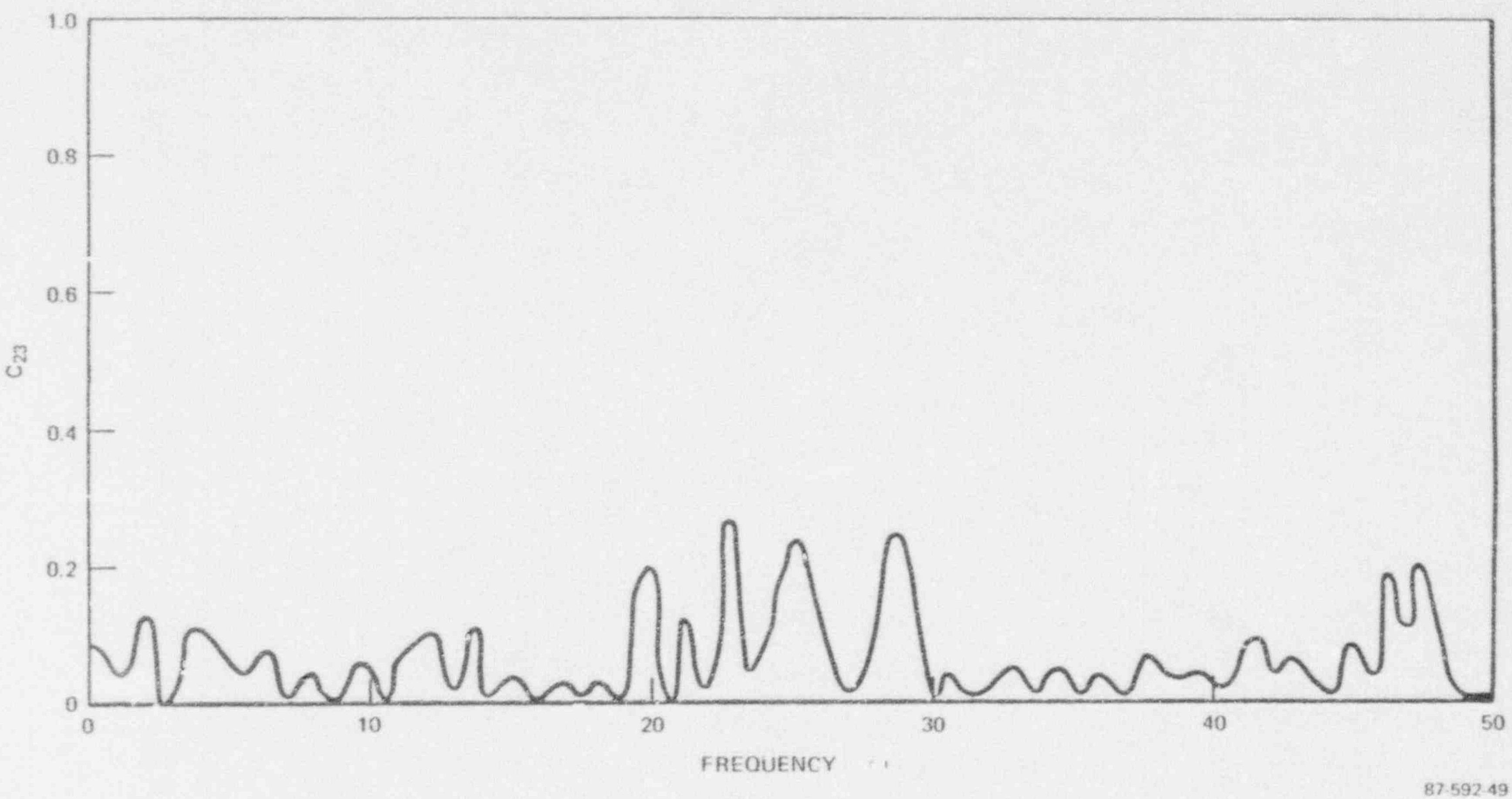


Figure 3.7–23 COHERENCE FUNCTION C_{23} FOR EARTHQUAKE COMPONENTS H2 AND V

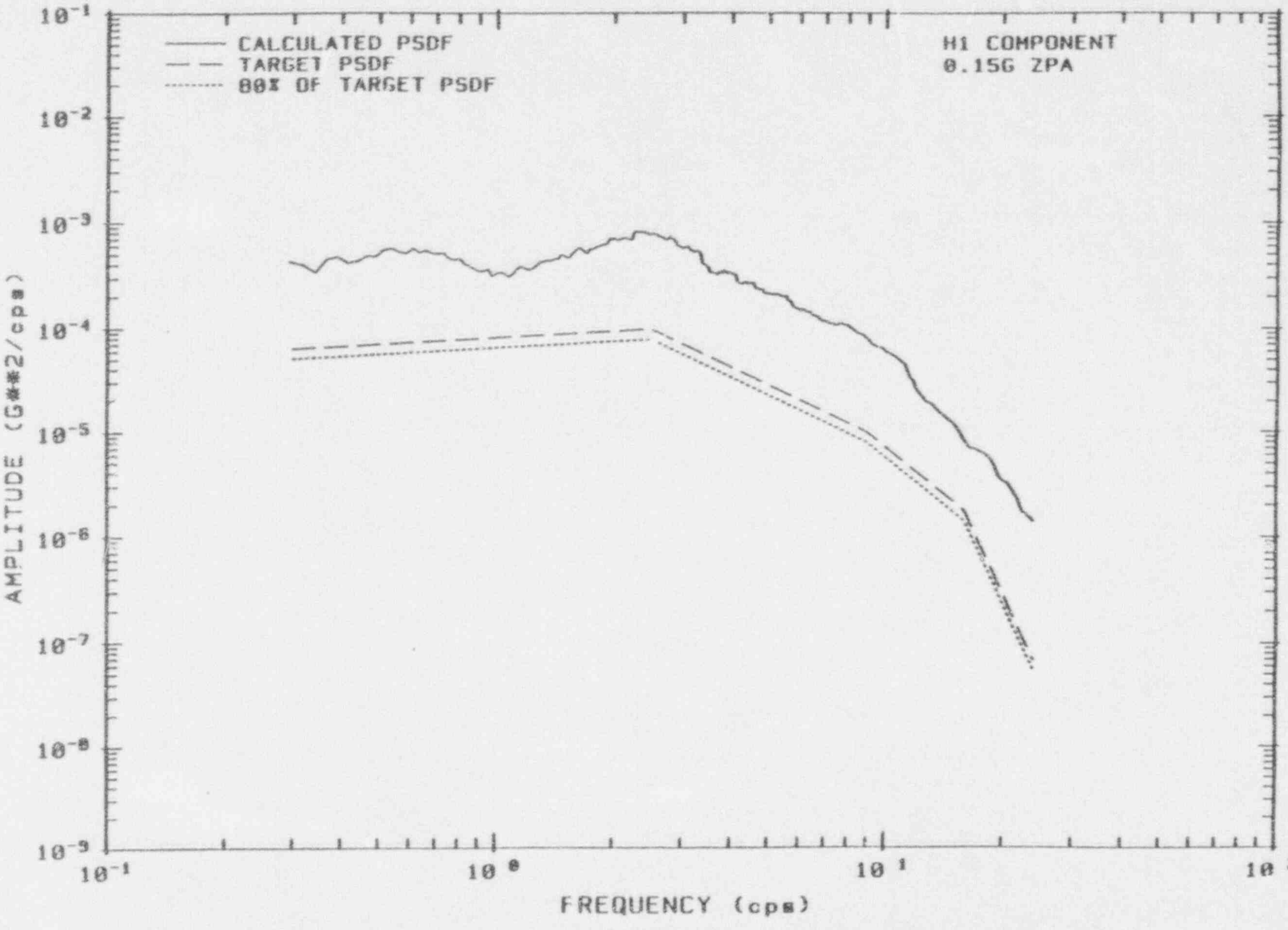


Figure 3.7-24 POWER SPECTRAL DENSITY FUNCTION OF SYNTHETIC H1 TIME HISTORY

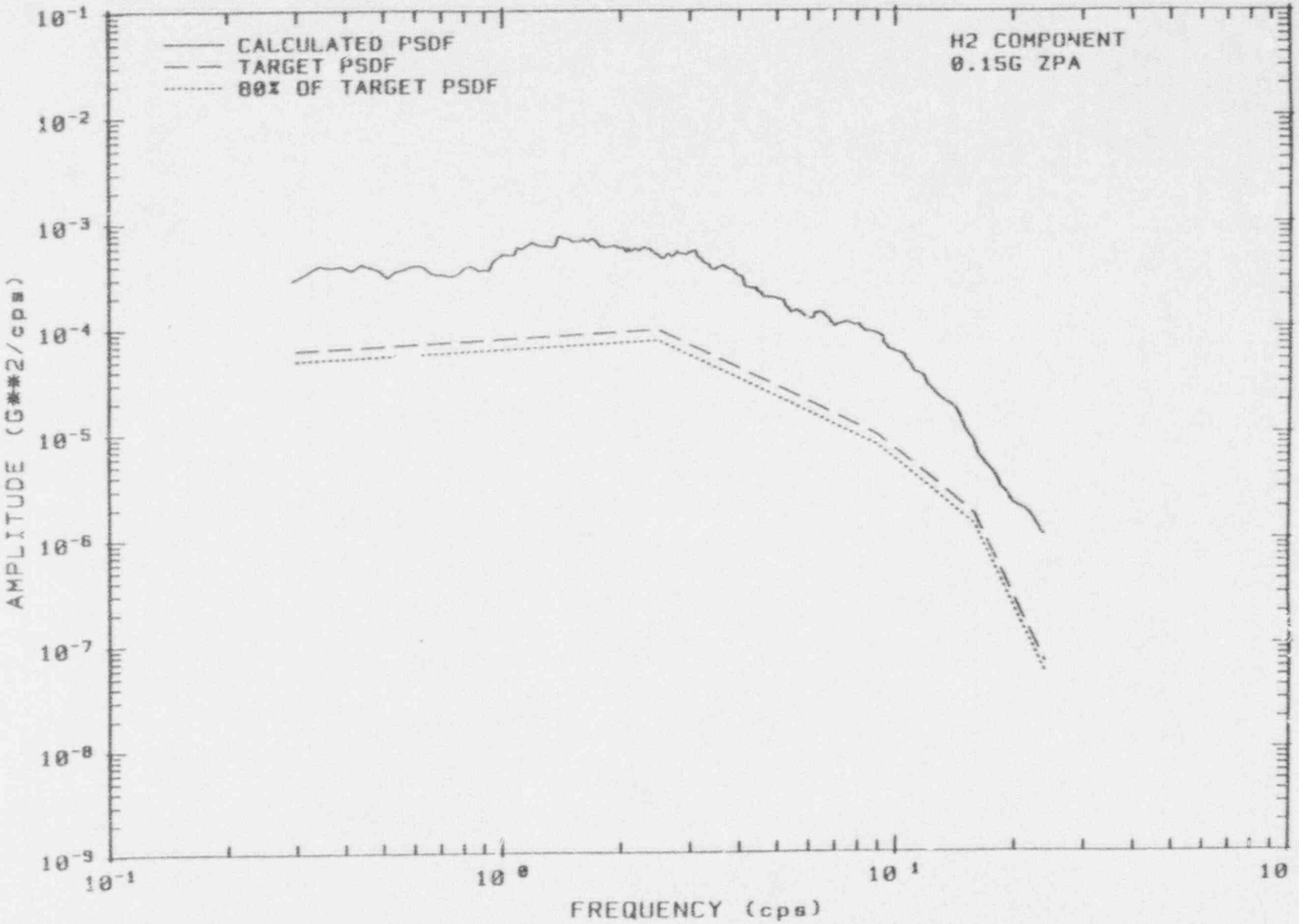


Figure 3.7-25 POWER SPECTRAL DENSITY FUNCTION OF SYNTHETIC H2 TIME HISTORY

Figure 3.7-26 (Deleted)

Figure 3.7-27 (Deleted)

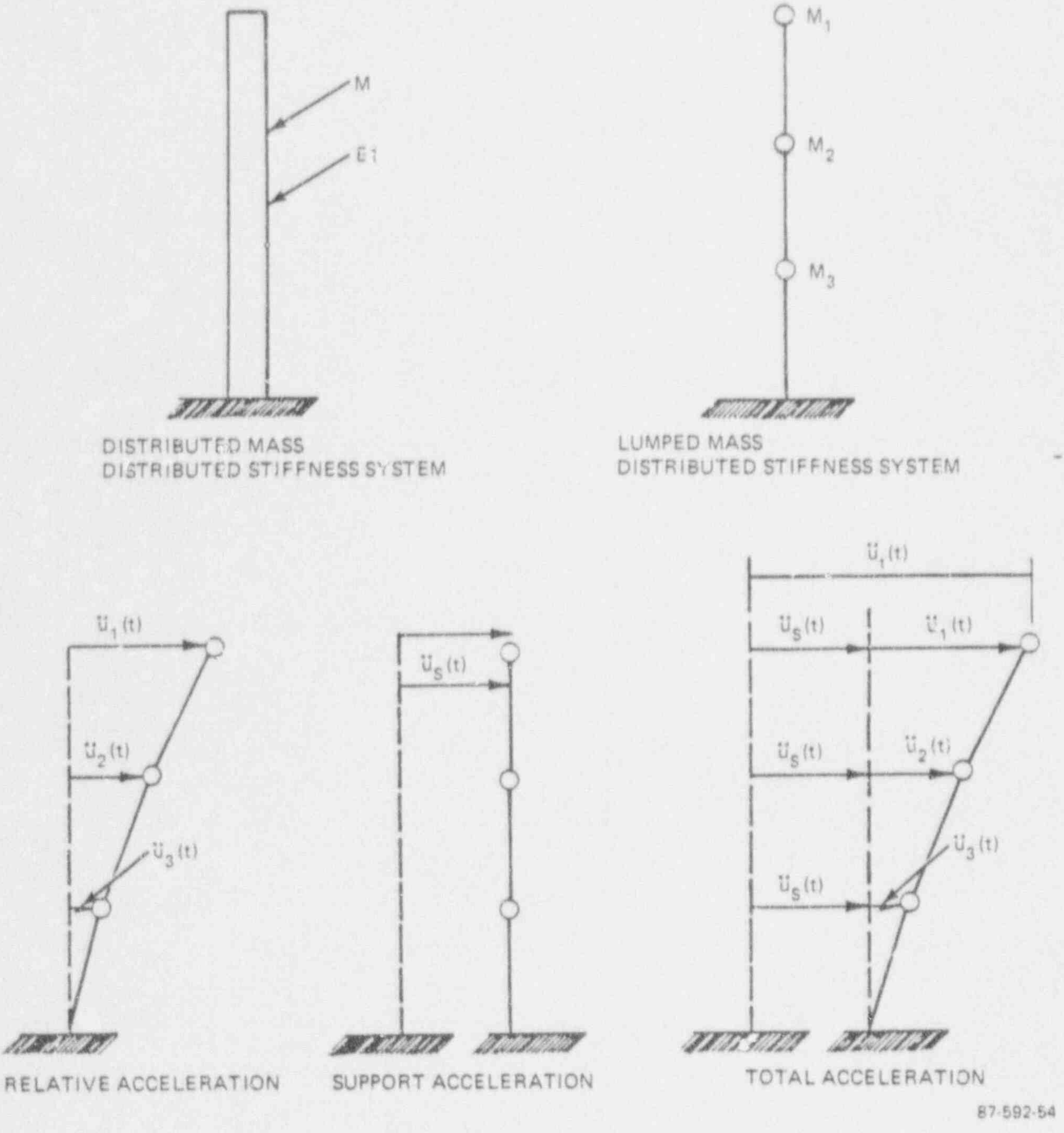
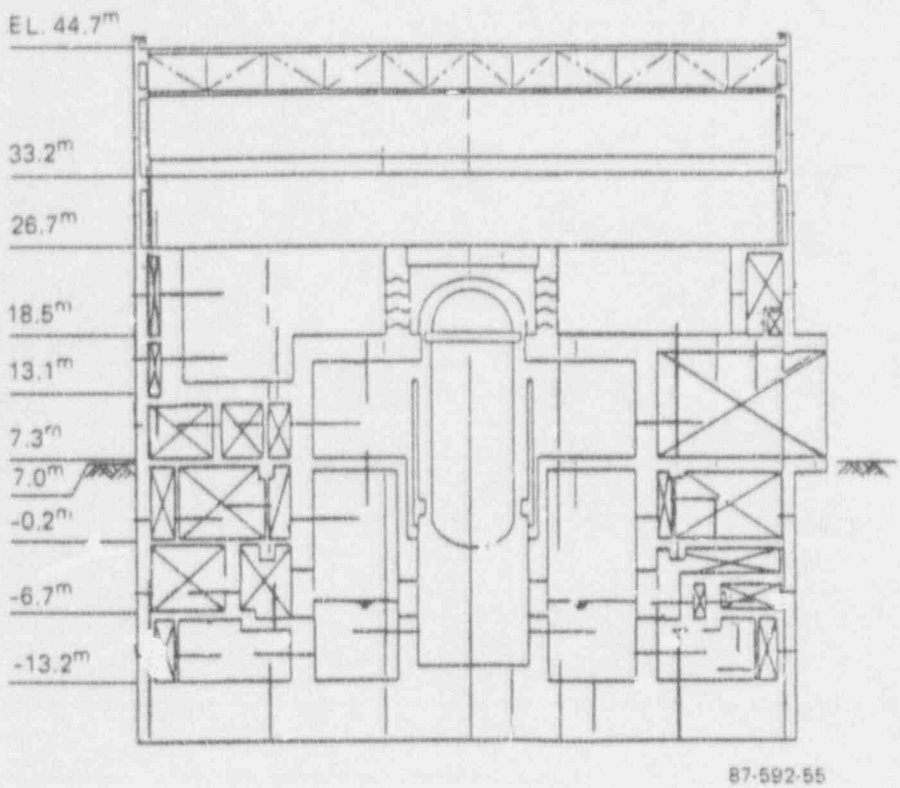
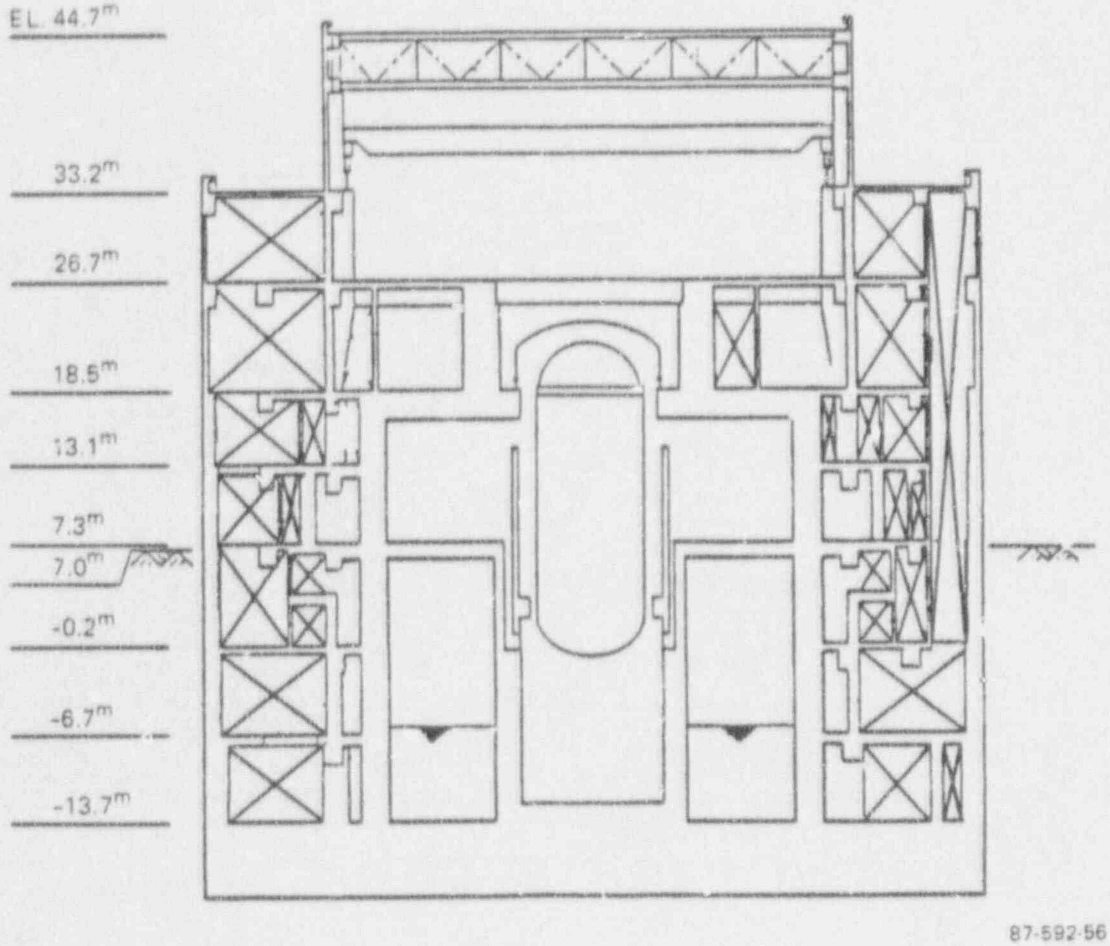


Figure 3.7-28 SEISMIC SYSTEM ANALYTICAL MODEL



NOTE: ELEVATIONS ARE RELATIVE TO THE RPV BOTTOM HEAD

Figure 3.7-29 REACTOR BUILDING ELEVATION (0°-180° SECTION)



NOTE: ELEVATIONS ARE RELATIVE TO RPV BOTTOM HEAD

Figure 3.7-30 REACTOR BUILDING ELEVATION (90°-270° SECTIONS)

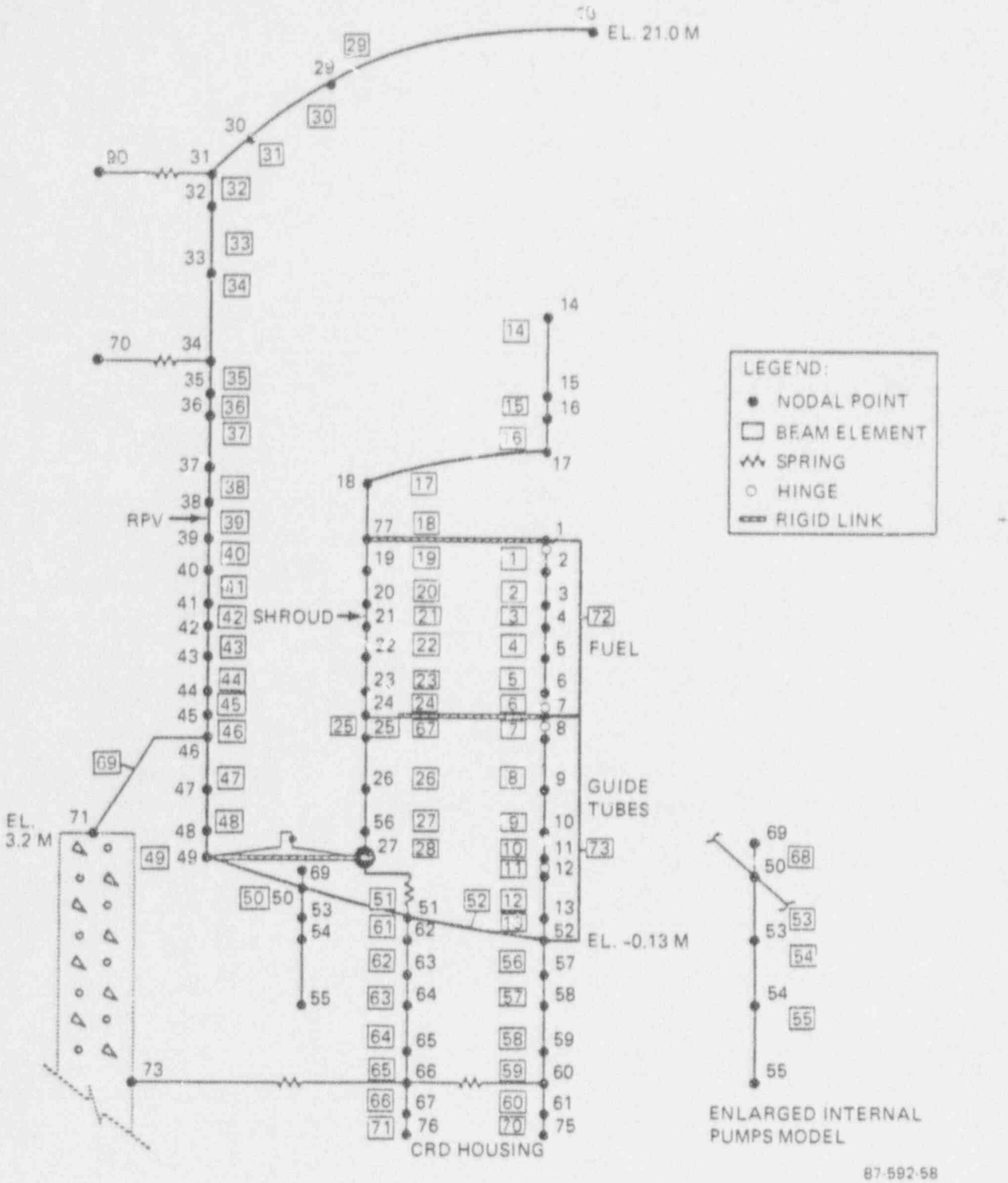
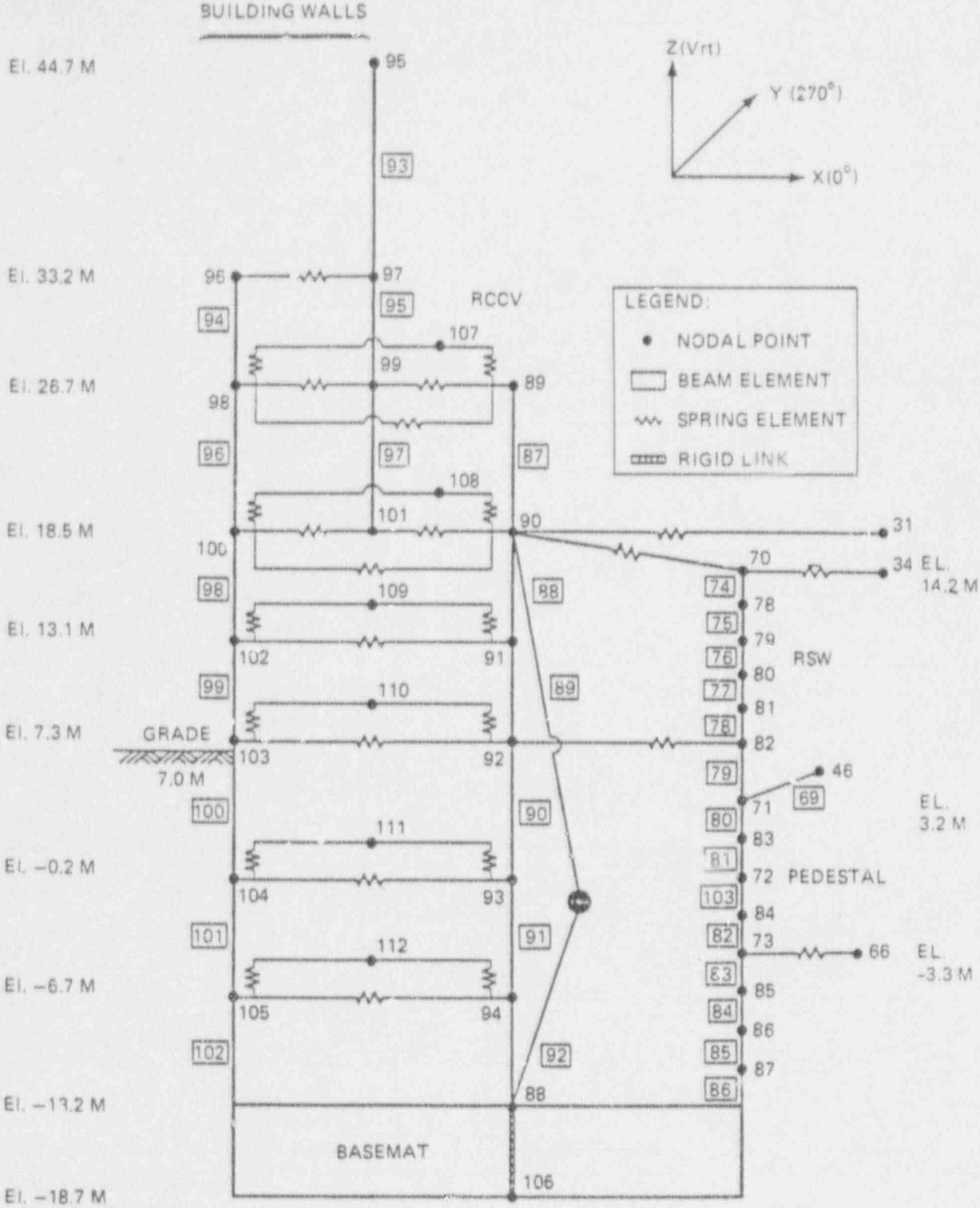


Figure 3.7-32 REACTOR PRESSURE VESSEL (RPV) AND INTERNALS MODEL



NOTES: 1) IN Y–Z PLANE, THE TWO REACTOR BUILDING STICKS ABOVE El. 18.5m COMBINE INTO ONE SINGLE STICK.

2) THE ROTATIONAL SPRING BETWEEN NODES 90 AND 88 IS PRESENTED ONLY IN THE X–Z PLANE

Figure 3.7-31 REACTOR BUILDING MODEL

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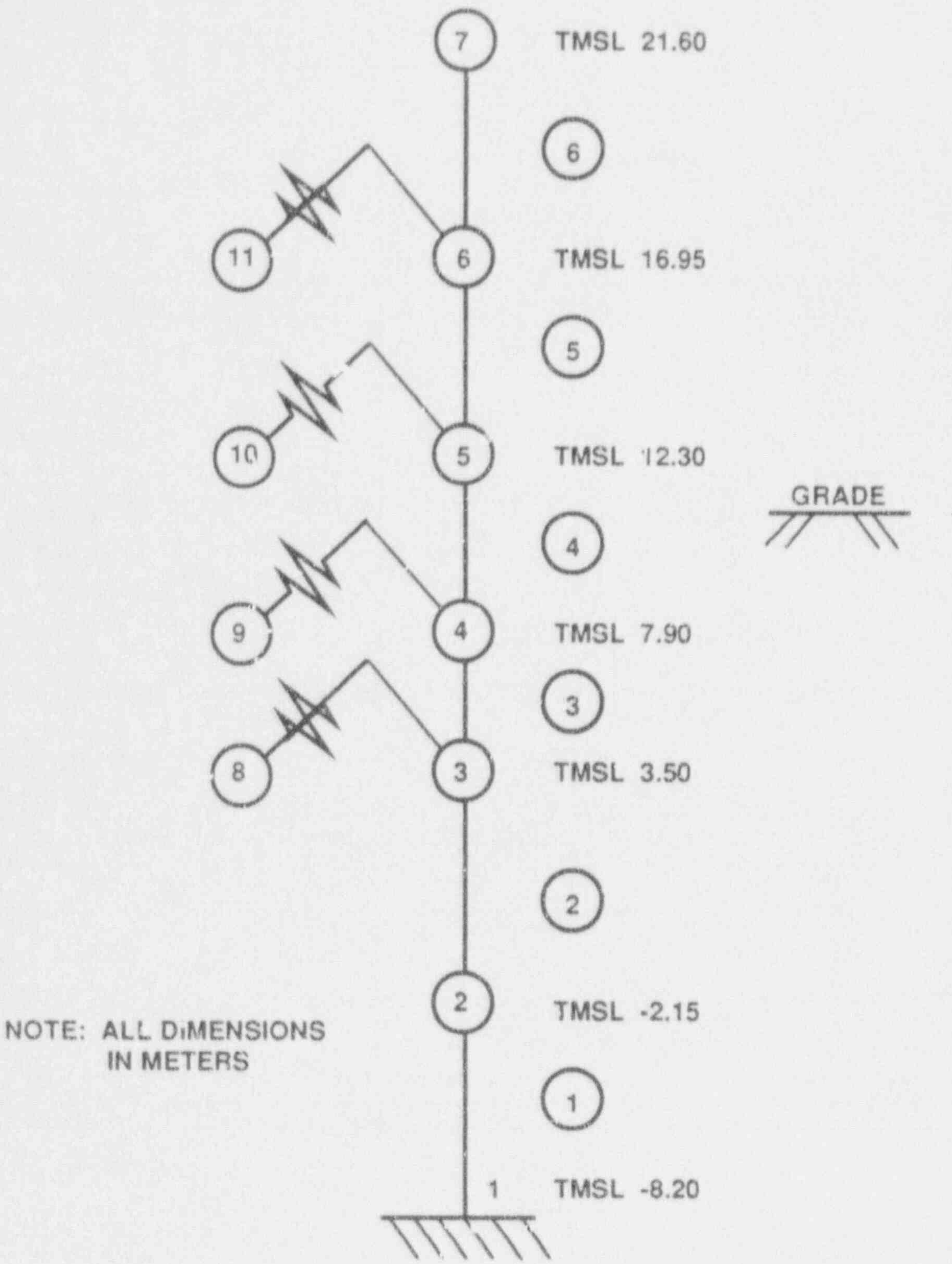

Figure 3.7-33 CONTROL BUILDING DYNAMIC MODEL

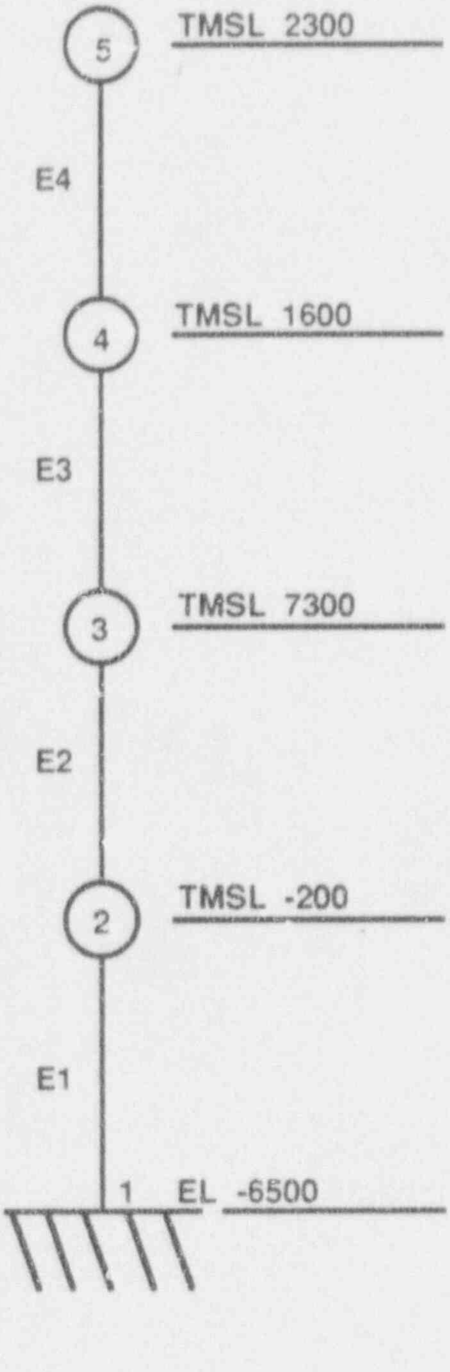

Figure 3.7-34 RADWASTE BUILDING SEISMIC MODEL

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TABLES

SECTION 3.9

TABLES (Continued)

ILLUSTRATIONS

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Design Transients

The plant events affecting the mechanical systems components and equipment are summarized in Table 3.9-1 in two groups: (1) plant operating events during which thermal-hydraulic transients occur, and (2) dynamic loading events due to accidents, earthquakes and certain operating conditions. The number of cycles associated with each event for the design of the reactor pressure vessel (RPV) as an example are listed in Table 3.9-1. The plant operating conditions are identified as normal, upset, emergency, faulted, or testing as defined in Subsection 3.9.3.1.1. Appropriate Service Levels (A, B, C, D or testing) as defined in ASME Code, Section III, are designated for design limits. The design and analysis of safety-related piping and equipment using specific applicable thermal-hydraulic transients which are derived from the system behavior during the events listed in Table 3.9-1 are documented in the design specification and/or stress report of the respective equipment. Table 3.9-2 shows the loading combinations and the standard acceptance criteria.

3.9.1.2 Computer Programs Used in Analyses

The computer programs used in the analysis of the major safety-related components are described in Appendix 3D.

The computer programs used in the analyses of Seismic Category I components are maintained either by General Electric or by outside computer program developers. In either case, the quality of the programs and the computed results are controlled. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature including analytical results or numerical results to the benchmark problems.

The updates to Appendix 3D will be provided to indicate any additional programs used or the later version of the described programs, and the method of their verification.

3.9.1.3 Experimental Stress Analysis

The following subsections list those NSSS components for which experimental stress analysis is performed in conjunction with analytical evaluation. The experimental stress analysis methods are used in compliance with the provisions of Appendix II of the ASME Code, Section III.

3.9.1.3.1 Piping Snubbers and Restraints

The following components have been tested to verify their design adequacy:

(1) piping seismic snubbers, and

(2) pipe whip restraints.

Descriptions of the snubber and whip restraint tests are contained in Subsection 3.9.3.4 and Section 3.6, respectively.

3.9.1.3.2 Fine Motion Control Rod Drive (FMCRD)

Experimental data were used in developing the hydraulic analysis computer called the FMCRD01. The output of FMCRD01 is used in the dynamic analysis of both ASME Code and non-Code parts. Pressures used in the analysis of these parts are also determined during actual testing of prototype control rod drives.

3.9.1.4 Considerations for the Evaluation of Faulted Condition

All Seismic Category I equipment are evaluated for the faulted (Service Level D) loading conditions identified in Tables 3.9-1 and 3.9-2. In all cases, the calculated actual stresses are within the allowable Service Level D limits. The following subsections address the evaluation methods and stress limits used for the equipment and identify the major components evaluated for faulted conditions. Additional discussion of faulted analysis can be found in

Subsections 3.9.2.5, 3.9.3, and 3.9.5.

Deformations under faulted conditions are evaluated in critical areas and the necessary design deformation limits, such as clearance limits, are satisfied.

3.9.1.4.1 Control Rod Drive System Components

3.9.1.4.1.1 Fine Motion Control Rod Drive

The fine motion control rod drive (FMCRD) major components that are part of the reactor coolant pressure boundary are analyzed and evaluated for the faulted conditions in accordance with the ASME Code, Section III, Appendix F.

3.9.1.4.1.2 Hydraulic Control Unit

The hydraulic control unit (HCU) is analyzed and tested for withstanding the faulted condition loads. Dynamic tests establish the "g" loads in horizontal and vertical directions as the HCU capability for the frequency range that is likely to be experienced in the plant. These tests also insure that the scram function of the HCU can be performed under these loads. Dynamic analysis of the HCU with the mounting beams is performed to assure that the maximum faulted condition loads remain below the HCU capability.

3.9.1.4.2 Reactor Pressure Vessel Assembly

The reactor pressure vessel assembly includes: (1) the reactor pressure vessel boundary ut to and including the nozzles and housings for FMCRD, internal pump and in-core instrumentation; (2) support skirt; and (3) the shroud support, including legs, cylinder, and plate. The design and analysis of these three parts comply with subsections NB, NF, and NG, respectively, of the ASME Code, Section III. For faulted conditions, the reactor vessel is evaluated using elastic analysis. For the support skirt and shroud support, an elastic analysis is performed, and buckling is evaluated for compressive load cases for certain locations in the assembly.

3.9.1.4.3 Core Support Structures and Other Safety Reactor Internal Components

The core support structures and other safety class reactor internal components are evaluated for faulted conditions. The basis for determining the faulted loads for seismic events and other dynamic events is given in Section 3.7 and Subsection 3.9.5, respectively. The allowable Service Level D limits for evaluation of these structures are provided in Subsection 3.9.5.

3.9.1.4.4 RPV Stabilizer and FMCRD - and In-Core Housing Restraints (Supports)

The calculated maximum stresses meet the allowable stress limits stated in Table 3.9-1 and 3.9-2 under faulted conditions for the RPV stabilizer and supports for the fine motion control rod drive housing and in-core housing for faulted conditions. These supports restrain the components during earthquake, pipe rupture or other reactor building vibration events. 210.30

3.9.1.4.5 Main Steam Isolation Valve, Safety/Relief Valve and Other ASME Class 1 Valves.

Elastic analysis methods and standard design rules, as defined in ASME Code Section III, are utilized in the analysis of the pressure boundary, Seismic Category I, ASME Class 1 valves. The Code-allowable stresses are applied to assure integrity under applicable loading conditions including faulted condition. Subsection 3.9.3.2.4 discusses the operability qualification of the major active valves including main steam isolation valve and the main steam safety/relief valve for seismic and other dynamic conditions. The allowable stresses for various operating conditions, including faulted, for active ASME Class 1 valves are provided in a footnote to Table 3.9-2.

3.9.1.4.6 ECCS and SLC Pumps, RRS and RHR Heat Exchangers, RCIC Turbine, and RRS Motor

The ECCS (RHR, RCIC and HPCF) pumps, SLC pumps, RHR heat exchangers, and RCIC turbine are analyzed for the faulted loading conditions. The ECCS and SLC pumps are active ASME Class 2 components. The allowable stresses for active pumps are provided in a footnote to Table 3.9-2.

The reactor coolant pressure boundary components of the reactor recirculation system (RRS) pump motor assembly, and recirculation motor cooling (RMC) subsystem heat exchanger are ASME Class 1 and Class 3, respectively, and are analyzed for the faulted loading conditions. All equipment stresses are within the elastic limits.

3.9.1.4.7 Fuel Storage and Refueling Equipment

Storage, refueling, and servicing equipment which is important to safety is classified as essential components per the requirements of 10CFR50 Appendix A. This equipment and other equipment which in case of a failure would degrade an essential component is defined in Section 9.1 and is classified as Seismic Category I. These components are subjected to an elastic dynamic finite-element analysis to generate loadings. This analysis utilizes appropriate floor response spectra and combines loads at frequencies up to 33 Hz for seismic loads and up to 60 Hz for other dynamic loads in three directions. Imposed stresses are generated and combined for normal, upset, and faulted conditions. Stresses are compared, depending on the specific safety class of the equipment, to Industrial Codes, ASME, ANSI or Industrial Standards, AISC, allowables.

3.9.1.4.8 Fuel Assembly (Including Channel)

GE BWR fuel assembly (including channel) design bases, and analytical and evaluation methods including those applicable to the faulted conditions are the same as those contained in References 1 and 2.

3.9.1.4.9 ASME Class 2 and 3 Vessels

Elastic analysis methods are used for evaluating faulted loading conditions for Class 2 and 3 vessels. The equivalent allowable stresses using elastic techniques are obtained from NC/ND-3300 and NC-3200 of the ASME Code Section III. These allowables are above elastic limits.

3.9.1.4.10 ASME Class 2 and 3 Pumps

Elastic analysis methods are used for evaluating faulted loading conditions for Class 2 and 3 pumps. The equivalent allowable stresses for nonactive pumps using elastic techniques are obtained from NC/ND-3400 of the ASME Code Section III. These allowables are above elastic limits. The allowables for active pumps are provided in a footnote to Table 3.9-2.

3.9.1.4.11 ASME Class 2 and 3 Valves

Elastic analysis methods and standard design rules are used for evaluating faulted loading conditions for Class 2, and 3 valves. The equivalent allowable stresses for nonactive valves using elastic techniques are obtained from NC/ND-3500 of ASME Code, Section III. These allowables are above elastic limits. The allowables for active valves are provided in a footnote to Table 3.9-2.

3.9.1.4.12 ASME Class 1, 2 and 3 Piping

Elastic analysis methods are used for evaluating faulted loading conditions for Class 1, 2, and 3 piping. The equivalent allowable stresses using elastic techniques are obtained from Appendix F (for Class 1) and NC/ND-3600 (for Class 2 and 3 piping) of the ASME Code Section III. These allowables are above elastic limits. The allowables for functional capability of the essential piping are provided in a footnote to Table 3.9-2.

3.9.1.5 Inelastic Analysis Methods

Inelastic analysis is only applied to ABWR components to demonstrate the acceptability of three types of postulated events. Each event is an extermly low-probability occurence and the equipment affected by these events would not be reused. These three events are:

(1) Postulated gross piping failure.

(2) Postulated blowout of a reactor internal recirculation (RIP) motor casing due to a weld failure.

(3) Postulated blowout of a control rod drive (CRD) housing due to a weld failure.

The loading combinations and design criteria for pipe whip restraints utilized to mitigate the effects of postulated piping failures are provided in Subsection 3.6.2.3.3.

In the case of the RIP motor casing failure event, there are specific restraints applied to mitigate the effects of the failure. The mitigation arrangement consists of lugs on the RPV bottom head to which are attached two long rods for each RIP. The lower end of each rod engages two lugs on the RIP motor/cover. The use of inelastic analysis methods is limited to the middle slender body of the rod itself. The attachment lugs, bolts and clevises are shown to be adequate by elastic analysis. The selection of stainless steel for the rod is based on its high ductility assumed for energy absorption during inelastic deformation.

The mitigation for the CRD housing attachment weld failure is by somewhat different means than are those of the RIP in that the components with regular functions also function to mitigate the weld failure effect. The components are specifically:

(1) Core support plate

(2) Control rod guide tube

(3) Control rod drive housing

(4) Control rod drive outer tube

(5) Bayonet fingers

Only the cylindrical bodies of the control rod guide tube, control rod drive housing and control rod drive outer tube are analyzed for energy absorption by inelastic deformation.

Inelastic analysis for there latter two events together with the criteria used for evaluation are consistent with the procedures described in Subsection 3.6.2.3.3 for the different components of a pipe whip restraint. Figure 3.9-6 shows the stress-strain curve used for the blowout restraints.

3.9.2 Dynamic Testing and Analysis

3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

The overall test program is divided into two phases; the preoperational test phase and the initial startup test phase. Piping vibration, thermal expansion and dynamic effects testing will be performed during both of these phases as described in Chapter 14. Subsections 14.2.12.1.51, 14.2.12.2.10 and 14.2.12.2.11 relate the specific role of this testing to the overall test program. Discussed below are the general requirements for this testing. It 210.31

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

should be noted that because one goal of the dynamic effects testing is to verify the adequacy of the piping support system, such components are addressed in the subsections that follow. However, the more specific requirements for the design and testing of the piping support system are described in Subsection 3.9.3.4.1.

3.9.2.1.1 Vibration and Dynamic Effects Testing

The purpose of these tests is to confirm that the piping, components, restraints and supports of specified high- and moderate-energy systems have been designed to withstand the dynamic effects of steady state flow-induced vibration and anticipated operational transient conditions. The general requirements for vibration and dynamic effects testing of piping systems are specified in Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors". More specific vibration testing requirements are defined in ANSI/ASME OM3, "Requirements for Preoperational and Initial Startup Vibration Testing of Nuclear Power Plant Piping Systems". Preparation of detailed test specifications will be in full accordance with this standard and will address such issues as prerequisites, test conditions,
210.31 precautions, measurement techniques, monitoring requirements, test hold points and acceptance criteria. The development and specification of the types of measurements required, the systems and locations to be monitored, the test acceptance criteria, and the corrective actions that may be necessary are discussed in more detail below.

3.9.2.1.1.1 Measurement Techniques

There are essentially three methods available for determining the acceptability of steady state and transient vibration for the affected systems. These three measurement techniques are visual observation, local measurements, or remotely monitored/recorded measurements. The technique used in each case will depend on such factors as the safety significance of the particular system, the expected mode and/or magnitude of the vibration, the assessability of the system during designated testing conditions, or the need for a time history recording of the vibratory behavior. Typically, the systems where vibration has the greatest safety implication will be subject to more rigorous testing and precise instrumentation requirements and, therefore, will require remote monitoring techniques. Local measurement techniques, such as the use of a hand-held vibrometer, are more appropriate in cases where it is expected that the vibration will be less complex and of lessor magnitude. Many systems that are assessable during the preoperational test phase and that do not show significant intersystem interactions will fall into this category. Visual observations are utilized where 210.31
vibration is expected to be minimal and the need for a time history record of transient behavior is not anticipated. However, unexpected visual observations or local indications may require that a more sophisticated technique be used. Also, the issue of assessability should be considered. Application of these measurement techniques is detailed in the appropriate testing specification consistent with the guidelines contained in ANSI/ASME OM3.

3.9.2.1.1.2 Monitoring Requirements

As described in Subsection 14.2.12.1.51, 14.2.12.2.10 and 14.2.12.2.11 all safety-related piping systems will be subjected to steady state and transient vibration measurements. The scope of such testing shall include safety-related instrumentation piping and attached small-bore piping (branch piping). Special attention should 210.32
be given to piping attached to pumps, compressors, and other rotating or reciprocating equipment. Monitoring location selection considerations should include the proximity of isolation valves, pressure or flow control valves, flow orifices, distribution headers, pumps and other elements where shock or high turbulence may be 210.31
of concern. Location and orientation of instrumentation and/or measurements will be detailed in the appropriate test specification. Monitored data should include actual deflections and frequencies as well as related system operating conditions. Time duration of data recording should be sufficient to indicate whether the vibration is continuous or transient. Steady state monitoring should be performed at critical conditions such as minimum or maximum flow, or abnormal combinations or configurations of system pumps or valves. Transient monitoring should include anticipated system and total plant operational transients where critical piping or components are expected to show

significant response. Steady state conditions and transient events to be monitored will be detailed in the appropriate testing specification consistent with OM3 guidelines.

3.9.2.1.1.3 Test Evaluation and Acceptance Criteria

The piping response to test conditions shall be considered acceptable if the review of the test results indicates that the piping responds in a manner consistent with predictions of the stress report and/or that piping stresses are within ASME Code Section III (NB-36000) limits. Acceptable limits are determined after the completion of piping systems stress analysis and are provided in the piping test specifications.

To ensure test data integrity and test safety, criteria have been established to facilitate assessment of the test while it is in progress. For steady state and transient vibration the pertinent acceptance criteria are usually expressed in terms of maximum allowable displacement/deflection. Visual observation should only be used to confirm the absence of significant levels of vibration and not to determine acceptability of any potentially excessive vibra-
210.31 tion. Therefore, in some cases other measurement techniques will be required with appropriate quantitative acceptance criteria.

There are typically two levels of acceptance criteria for allowable vibration displacements/deflections. Level 1 criteria are bounding type criteria associated with safety limits while Level 2 criteria are stricter criteria associated with system or component expectations. For steady state vibration the Level 1 criteria are based on the endurance limit (10,000 psi) to assure no failure from fatigue over the life of the plant. The corresponding Level 2 criteria are based on one half the endurance limit (5,000 psi). For transient vibration the Level 1 criteria are based on either the ASME-III code upset primary stress limit or the applicable snubber load capacity. Level 2 criteria are based on a given tolerance about the expected deflection value.

3.9.2.1.1.4 Reconciliation and Corrective Actions

During the course of the tests, the remote measurements will be regularly checked to verify compliance with acceptance criteria. If trends indicate that criteria may be violated, the measurements should be monitored at more frequent intervals. The test will be held or terminated as soon as criteria are violated. As soon as possible after the test hold or termination appropriate investigative and corrective actions will be taken. If practicable, a walkdown of the piping and suspension system should be made in an attempt to identify potential obstructions or improperly operating suspension components. Hangers and snubbers should be positioned such that they can accommodate the expected deflections without bottoming out or extending fully. All signs of damage to piping supports or anchors shall be investigated.

Instrumentation indicating criteria failure shall be checked for proper operation and calibration including comparison with other instrumentation located in the proximity of the excessive vibration. The assumptions used in the calculations that generated the applicable lim-
its should be verified against actual conditions 210.31
and discrepancies noted should be accounted for in the criteria limits. This may require a reanalysis at actual system conditions.

Should the investigation of instrumentation and calculations fail to reconcile the criteria violations, then physical corrective actions may be required. This might include identification and reduction or elimination of offending forcing functions, detuning of resonant piping spans by appropriate modifications, addition of bracing, or changes in operating procedures to avoid troublesome conditions. Any such modifications will require retest to verify vibrations have been sufficiently reduced.

3.9.2.1.2 Thermal Expansion Testing

A thermal expansion preoperational and startup testing program performed through the use of visual observation and remote sensors has been established to verify that normal unrestrained thermal movement occurs in specified safety-related high- and moderate-energy piping systems. The purpose of this program is to ensure the following:

(1) the piping system during system heatup and cooldown is free to expand and move without unplanned obstruction or restraint in the x, y, and z directions;

(2) the piping system does shakedown after a few thermal expansion cycles;

(3) the piping system is working in a manner consistent with the assumption of the stress analysis;

(4) there is adequate agreement between calculated values and measured values of displacements; and

(5) there is consistency and repeatability in thermal displacements during heatup and cooldown of the systems.

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The general requirements for thermal expansion testing of piping systems are specified in Regulatory Guide 1.68, "Preoperational and Initial Startup Testing Programs for Water-Cooled Power Reactors." More specific requirements are defined in ANSI/ASME OM7 "Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems." Detailed test specifications will be prepared in full accordance with this standard and will address such issues as prerequisites, test conditions, precautions, measurement techniques, monitoring requirements, test hold points and acceptance criteria. The development and specification of the types of measurements required, the systems and locations to be monitored, the test acceptance criteria, and the corrective actions that may be necessary are discussed in more detail below.

3.9.2.1.2.1 Measurement Techniques

Verification of acceptable thermal expansion of specified piping systems can be accomplished by several methods. One method is to physically walkdown the piping system and verify by visual observation that free thermal movement is unrestrained. This might include verification that piping supports such as snubbers and spring hangers are not fully extended or bottomed out and that the piping (including branch lines and instrument lines) and its insulation is not in hard contact with other piping or support structures. Another method would involve local measurements, using a hand held scale or ruler, against a fixed reference or by recording the position of a snubber or spring can. A more precise method would be using permanent or temporary instrumentation that directly measures displacement, such as a lanyard potentiometer, that can be monitored via a remote indicator or recording device. The technique to be used will depend on such factors as the amount of movement predicted and the assessability of the piping.

Measurement of piping temperature is also of importance when evaluating thermal expansion. This may be accomplished either indirectly via the temperature of the process fluid or by direct measurement of the piping wall temperature and such measurements may be obtained either locally or remotely. The choice of technique used shall depend on such considerations as the accuracy required and the assessability of the piping.

3.9.2.1.2.2. Monitoring Requirements

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As described in Subsections 14.2.12.1.51 and 14.2.12.2.10 all safety-related piping shall be included in the thermal expansion testing program. Thermal expansion of specified piping systems should be measured at both the cold and hot extremes of their expected operating conditions. Physical walkdowns and recording of hanger and snubber positions should also be conducted where possible considering assessability and local environmental and radiological conditions in the hot and cold states. Displacements and appropriate piping/process temperatures shall be recorded for those systems and conditions specified. Sufficient time shall have passed before taking such measurements to ensure the piping system is at a steady state condition. In selecting locations for monitoring piping response, consideration shall be given to the maximum responses predicted by the piping analysis. Specific consideration should also be given to the first run of pipe attached to component nozzles and pipe adjacent to structures requiring a controlled gap.

3.9.2.1.2.3 Test Evaluation and Acceptance Criteria

To ensure test data integrity and test safety, criteria have been established to facilitate assessment of the test while it is in progress. Limits of thermal expansion displacements are established prior to start of piping testing to which the actual measured displacements are compared to determine acceptability of the actual motion. If the measured displacement does not vary from the acceptance limits values by more than the specified tolerance, the piping system is responding in a manner consistent with the predictions and is therefore acceptable. The piping response to test conditions shall be considered acceptable if the review of the test results indicates that the piping responds in a manner consistent with the predictions of the stress report and/or that piping stresses are within ASME Code Section III (NB-3600) limits. Acceptable thermal expansion limits are determined after the completion of piping systems stress analysis and are provided in the piping test specifications. Level 1 criteria are bounding criteria based on ASME-III Code stress limits. Level 2 criteria are stricter criteria based the predicted movements using the calculated deflections plus a selected tolerance.

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3.9.2.1.2.4 Reconciliation and Corrective Actions

During the course of the tests, the remote measurements will be regularly checked to verify compliance with acceptance criteria. If trends indicate that criteria may be violated, the measurements should be monitored at more frequent intervals. The test will be held or terminated as soon as criteria are violated. As soon as possible after the test hold or termination appropriate investigative and corrective actions will be taken. If practicable, a walkdown of the affected piping and suspension system should be made in an attempt to identify potential obstruction to free piping movement. Hangers and snubbers should be positioned within their expected cold and hot settings. All signs of damage to piping or supports shall be investigated.

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Instrumentation indicating criteria failure shall be checked for proper operation and calibration including comparison with other instrumentation located in the proximity of the out-of-bounds movement. Assumptions, such as piping temperature, used in the calculations that generated the applicable limits should be compared with actual test conditions. Discrepancies noted should be accounted for in the criteria limits including possible reanalysis.

Should the investigation of instrumentation and calculations fail to reconcile the criteria violations or should the visual inspection reveal an unintended restraint, then physical corrective actions may be required. This might include complete or partial removal of an interfering structure; replacing, readjusting or repositioning piping system supports; modifying the pipe routing; or modifying system operating procedures to avoid the temperature conditions that resulted in the unacceptable thermal expansion.

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3.9.2.2 Seismic Qualification of Safety-Related Mechanical Equipment (Including Other RBV Induced Loads)

This subsection describes the criteria for dynamic qualification of safety-related mechanical equipment and associated supports, and also describes the qualification testing and/or analysis applicable to the major components on a component by component basis. Seismic and other events that may induce reactor building vibration (RBV)-(see Appendix 3B) are considered. In some cases, a module or assembly consisting of mechanical and electrical equipment is qualified as a unit (e.g., ECCS pumps). These modules are generally discussed in this subsection and Subsection 3.9.3.2 rather than providing discussion of the separate electrical parts in Section 3.10. Electrical supporting equipment such as control consoles, cabinets, and panels
210.31 are discussed in Section 3.10.

3.9.2.2.1 Tests and Analysis Criteria and Methods

The ability of equipment to perform its safety function during and after the application of a dynamic load is demonstrated by tests and/or analysis. The analysis is performed in accordance with Section 3.7. Selection of Testing, analysis or a combination of the two is determined by the type, size, shape, and complexity of the equipment being considered. When practical, the equipment operability is demonstrated by testing. Otherwise, operability is demonstrated by mathematical analysis.

Equipment which is large, simple, and/or consumes large amounts of power is usually qualified by analysis or static bend test to show that the loads, stresses and deflections are less than the allowable maximum. Analysis and/or static bend testing is also used to show there are no natural frequencies below 33 Hz for seismic loads and 60 Hz for other RBV loads*. If a natural frequency lower than 33 Hz in the case of seismic loads and 60 Hz in

** The 60 Hz frequency cutoff for dynamic analysis of suppression pool dynamic loads is the minimum requirement based on a generic Reference 8, using the missing strain energy method, performed for representative BWR equipment under high-frequency input loadings.*

the case of other RBV induced loads is discovered, dynamic tests and/or mathematical analyses may be used to verify operability and structural integrity at the required dynamic input conditions.

When the equipment is qualified by dynamic test, the response spectrum or time history of the attachment point is used in determining input motion.

Natural frequency may be determined by running a continuous sweep frequency search using a sinusoidal steady-state input of low magnitude. Dynamic load conditions are simulated by testing using random vibration input or single frequency input (within equipment capability) over the frequency range of interest. Whichever method is used, the input amplitude during testing envelopes the actual input amplitude expected during the dynamic loading condition.

The equipment being dynamically tested is mounted on a fixture which simulates the intended service mounting and causes no dynamic coupling to the equipment.

Equipment having an extended structure, such as a valve operator, is analyzed by applying static equivalent dynamic loads at the center of gravity of the extended structure. In cases where the equipment structural complexity makes mathematical analysis impractical, a static bend test is used to determine spring constant and operational capability at maximum equivalent dynamic load conditions.

3.9.2.2.1.1 Random Vibration Input

When random vibration input is used, the actual input motion envelopes the appropriate floor input motion at the individual modes. However, single frequency input such as sine beats can be use provided one of the following conditions are met:

(1) the characteristics of the required input motion is dominated by one frequency;

(2) the anticipated response of the equipment is adequately represented by one mode; or

(3) the input has sufficient intensity and duration to excite all modes to the required magnitude so that the testing response spectra will envelop the corresponding response spectra of the individual modes.

3.9.2.2.1.2 Application of Input Modes

When dynamic tests are performed, the input motion is applied to one vertical and one horizontal axis simultaneously. However, if the equipment response along the vertical direction is not sensitive to the vibratory motion along the horizontal direction and vice versa, then the input motion is applied to one direction at a time. In the case of single frequency input, the time phasing of the inputs in the vertical and horizontal directions are such that a purely rectilinear resultant input is avoided.

3.9.2.2.1.3 Fixture Design

The fixture design simulates the actual service mounting and causes no dynamic coupling to the equipment.

3.9.2.2.1.4 Prototype Testing

Equipment testing is conducted on prototypes of the equipment to be installed in the plant.

3.9.2.2.2 Qualification of Safety-Related Mechanical Equipment

The following subsections discuss the testing or analytical qualification of the safety-related major mechanical equipment, and other ASME III equipment, including equipment supports.

3.9.2.2.2.1 CRD and CRD Housing

The qualification of the CRD housing (with enclosed CRD) is done analytically, and the stress results of their analysis establish the structural integrity of these components. Preliminary dynamic tests are conducted to verify the operability of the control rod drive during a dynamic event. A simulated test, imposing dynamic deflection in the fuel channels up to values greater than the expected seismic response, is performed with the CRD demonstrated functioning satisfactorily.

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The test was conducted in two phases due to facility limitations. The seismic test facility cannot be pressurized while shaking therefore the charging pressure of the hydraulic control unit is reduced to simulate the back pressure that is applied in the reactor. The appropriate adjustment was determined by first running scram tests with the full reactor pressure and with peak transient pressure. Then with the test vessel at atmospheric pressure, the scram tests were repeated with reduced charging pressures until the scram performance matched that of the pressurized tests. This was repeated for the peak pressure also. The seismic tests were then performed with the appropriate pressure adjustments for the conditions being tested. The tests were run for various vibration levels with fuel channel deflections being the independent variable. The test facility was driven to vibration levels that produced various channel deflections up to 1.6 inches and the scram curves recorded. The 1.6 inch channel deflection is several times the channel deflection calculated for the actual seismic condition. The correlation of the test with analysis is via the channel deflection not the housing structural analysis since scramability is controlled by channel deflection not housing deflection.

3.9.2.2.2.2 Core Support (Fuel Support and CR Guide Tube)

A detailed analysis imposing dynamic effects due to seismic and other RBV events is performed to show that the maximum stresses developed during these events are much lower than the maximum allowed for the component material.

3.9.2.2.2.3 Hydraulic Control Unit (HCU)

The HCU is analyzed for the seismic and other RBV loads faulted condition and the maximum stress on the HCU frame is calculated to be below the maximum allowable for the faulted condition. As discussed in Subsection 3.9.1.4.1.2, the faulted condition loads are calculated to be below the HCU maximum capability.

3.9.2.2.2.4 Fuel Assembly (Including Channel)

GE BWR fuel channel design bases, analytical methods, and seismic considerations are similar to those contained in References 1 and 2. The resulting combined acceleration profiles, including fuel lift for all normal/upset and faulted events are to be shown less than the respective design basis acceleration profiles.

3.9.2.2.2.5 Reactor Internal Pump and Motor Assembly

The reactor internal pump (RIP) and motor assembly, including its appurtenances and support, is classified as Seismic Category I, but not active, and is designed to withstand the seismic forces, including other RBV loads. The qualification of the assembly is done analytically, and with a dynamic test.

3.9.2.2.2.6 ECCS Pump and Motor Assembly

A prototype ECCS (RHR and HPCF) pump motor assembly is qualified for seismic and other RBV loads via a combination of dynamic analysis and dynamic testing. The complete motor assembly is qualified via dynamic testing in accord- ance with IEEE 344. The qualification test program includes demonstration of startup capability as well as operability during dynamic loading conditions. This is discussed in more detail in Subsection 3.9.3.2.1.4.

The pump and motor assemblies, as units operating under seismic and other RBV load conditions, are qualified by dynamic analysis and results of the analysis indicate that the pump and motor are capable of sustaining the above loadings without exceeding the allowable stresses. This is discussed in more detail in Subsections 3.9.3.2.1.1 and 3.9.3.2.1.2.

3.9.2.2.2.7 RCIC Pump and Turbine Assembly

The RCIC pump construction is a horizontal, multistage type and is supported on a pedestal. The RCIC pump assembly is qualified analytically by static analysis for seismic and other RBV loadings as well as the design operating loads of pressure, temperature, and external piping loads. The results of this analysis confirm that the stresses are less than the allowable. This is also discussed in Subsection 3.9.3.2.2.

The RCIC turbine is qualified for seismic and other RBV loads via a combination of static analysis and dynamic testing. This is also discussed in Subsection 3.9.3.2.1.5. The turbine assembly consists of rigid masses (wherein static analysis is utilized) interconnected with control levers and elec- tronic control systems, necessitating final qualification via dynamic testing. Static loading analyses are employed to verify the structural integrity of the turbine assembly and the adequacy of bolting under operating, seismic, and other RBV loading conditions. The complete turbine assembly is qualified via dynamic testing in accordance with IEEE 344. The qualification test program includes demonstration of startup capability as well as operability during dynamic loading conditions. Operability under normal load conditions is assured by comparison to the operability of similar turbines in operating plants.

3.9.2.2.2.8 Standby Liquid Control Pump and Motor Assembly

The SLC positive displacement pump and motor assembly which is mounted on a common base plate is qualified analytically by static analysis of seismic and other RBV loadings as well as the design operating loads of pressure, temperature, and external piping loads. The results of this analysis confirm that the stresses are less than the allowables. This is also discussed in Subsection 3.9.3.2.2.

3.9.2.2.2.9 RMC and RHR Heat Exchangers

A three dimensional finite-element model is developed for each of the recirculation motor cooling (RMC) and residual heat removal (RHR) system heat exchangers and supports. The model is used to dynamically analyze the heat exchanger and its supports using the response spectrum analysis method, and to verify that the heat exchanger and supports can withstand seismic and other RBV loads. The same model is used to statically analyze and evaluate the nozzles due to the effect of the external piping loads and dead weight in order to ensure that nozzle load criteria and limits are met. Critical location stresses are evaluated and compared with the allowable stress criteria. The results of the analysis demonstrate that the stresses at all investigated locations are less than their corresponding allowable values.

3.9.2.2.2.10 Standby Liquid Control Tank

The standby liquid control storage tank is a cylindrical tank, with approximate dimensions of ten feet diameter and sixteen feet height, bolted to the concrete floor. The standby liquid control tank is qualified for seismic and other RBV loads by analysis for:

(1) stresses in the tank bearing tank plate;

(2) bolt stresses;

(3) sloshing loads imposed at the sloshing natural frequency;

(4) minimum wall thickness; and

(5) buckling.

The results of this analysis confirm that the calculated stresses at all investigated locations are less than their corresponding allowable values.

3.9.2.2.2.11 Main Steam Isolation Valves

The main steam isolation valves (MSIV) are qualified for seismic and other RBV loads. The fundamental requirement of the MSIV following an SSE or other faulted RBV loadings is to close and remain closed after the event. This capability is demonstrated by the test and analysis as outlined in Subsection 3.9.3.2.4.1.

3.9.2.2.2.12 Standby Liquid Control Valve (Injection Valve)

The motor-operated standby liquid control valve is qualified by type test to IEEE 344 for seismic and other RBV loads. The qualification test as discussed in Subsection 3.9.3.2.4.3 demonstrates the ability to remain operable after the application of horizontal and vertical dynamic loading in excess of the required response spectra. The valve and motor assemblies are qualified by dynamic analysis and the results of the analysis indicate the valve is capable of sustaining the dynamic loads without overstressing the pressure retaining components.

3.9.2.2.2.13 Main Steam Safety/Relief Valves

Due to the complexity of the structure and the performance requirements of the valve, the total assembly of the SRV (including electrical and pressure devices) is tested at dynamic accelerations equal to or greater than the combined SSE and other RBV loadings determined for the plant. Tests and analysis as discussed in Subsection 3.9.3.2.4.2 demonstrate the satisfactory operation of the valves during and after the test.

3.9.2.2.2.14 Fuel Pool Cooling and Cleanup System Pump and Motor Assembly

A static analysis is performed on the pump and motor assembly of the fuel pool cooling and cleanup system. This analysis shows that the pump and motor will continue to operate if subjected to a combination of SSE, other RBV, and normal operating loads. Analysis also ensures that pump running clearances, which include deflection of the pump shaft and pump pedestal, are met during seismic and other RBV loadings.

3.9.2.2.2.15 Other ASME III Equipment

Other equipment including associated supports is qualified for seismic and other RBV loads to ensure its functional integrity during and after the dynamic event. The equipment is tested, if necessary, to ensure its ability to perform its specified function before, during, and following a test.

Dynamic load qualification is done by a combination of test and/or analysis as described in Subsection 3.9.2.2.1. Natural frequency when determined by an exploratory test is in the form of a single-axis continuous-sweep frequency search using a sinusoidal steady-state input at the lowest possible amplitude which is capable of determining resonance. The search is conducted on each principal axis with a minimum of two continuous sweeps over the frequency range of interest at a rate no greater than one octave per minute. If no resonances are located, then the equipment is considered as rigid and single frequency tests at every 1/3 octave frequency interval are acceptable. Also, if all natural frequencies of the equipment are greater than 33 Hz for seismic loads and 60 Hz for other RBV loads, the equipment may be considered rigid and analyzed statically as such. In this static analysis, the dynamic forces on each component are obtained by concentrating the mass at the center of gravity and multiplying the mass by the appropriate floor acceleration. The dynamic stresses are then added to the operating stresses and a determination made of the adequacy of the strength of the equipment. The search for the natural frequency is done analytically if the equipment shape can be defined mathematically and/or by prototype testing.

If the equipment is a rigid body while its support is flexible, the overall system can be modeled as a single-degree-of-freedom system consisting of a mass and a spring. The natural frequency of the system is computed; then the acceleration is determined from the floor response spectrum curve using the appropriate damping value. A static analysis is then performed using this acceleration value. In lieu of calculating the natural frequency, the peak acceleration from the spectrum curve is used. The critical damping values for welded steel structures from Table 3.7-1 are employed.

In case the equipment cannot be considered as a rigid body, it can be modeled as a multi-degree-of-freedom system. It is divided into a sufficient number of mass points to ensure adequate representation. The mathematical model can be analyzed using modal analysis technique or direct integration of the equations of motion. Specified structural damping is used in the analysis unless justification for other values can be provided. A stress analysis is performed using the appropriate inertial forces or equivalent static loads obtained from the dynamic analysis of each mode.

For a multiple degree of freedom modal analysis, the modal response accelerations can be taken directly from the applicable floor response spectrum. The maximum spectral values within $\pm 10\%$ band of the calculated frequencies of the equipment are used for computation of modal dynamic response inertial loading. The total dynamic stress is obtained by combining the modal stresses. The dynamic stresses are added to the operating stresses using the loading combinations stipulated in the specific equipment specification and then compared with the allowable stress levels.

If the equipment being analyzed has no definite orientation, the worst possible orientation is considered. Furthermore, equipment is considered to be in its operational configuration (i.e., filled with the appropriate fluid and/or solid). The investigation ensures that the point of maximum stress is considered. Lastly a check is made to ensure that partially filled or empty equipment do not result in higher response than the operating condition. The analysis includes evaluation of the effects of the calculated stresses on mechanical strength, alignment, electrical performance (microphonics, contact bounce, etc.) and noninterruption of function. Maximum displacements are computed and interference effects determined and justified.

Individual devices are tested separately, when necessary, in their operating condition. Then the component to which the device is assembled is tested with a similar but inoperative device installed upon it.

The equipment, component, or device to be tested is mounted on the vibration generator in a manner that simulates the final service mounting. If the equipment is too large, other means of simulating the service mounting are used. Support structures such as air conditioning units, consoles, racks, etc., could be vibration tested without the equipment and/or devices being in operation provided they are performance tested after the vibration test. However, the components are in their operational configuration during the vibration test. The goal is to determine that, at the specified vibratory accelerations, the support structure does not amplify the forces beyond that level to which the devices have been qualified.

Equipment could alternatively be qualified by presenting historical performance data which demonstrates that the equipment satisfactorily sustains dynamic loads which are equal to greater than those specified for the equipment and that the equipment performs a function equal to or better than that specified for it.

Equipment for which continued function is not required after a seismic and other RBV loads event, but its postulated failure could produce an unacceptable influence on the performance of systems having a primary safety function, are evaluated. Such equipment is qualified to the extent required to ensure that an SSE including other RBV loads, in combination with normal operating conditions, would not cause unacceptable failure. Qualification requirements are satisfied by ensuring that the equipment in its functional configuration, complete with attached appurtenances, remains structurally intact and affixed to the interface. The structural integrity of internal components is not required; however, the enclosure of such components is required to be adequate to ensure their confinement. Where applicable, fluid or pressure boundary integrity is demonstrated. With a few exceptions, simplified analytical techniques are adequate.

Historically, it has been shown that the main cause for equipment damage during a dynamic excitation has been the failure of its anchorage. Stationary equipment is designed with anchor bolts or other suitable fastening strong enough to prevent overturning or sliding. The effects of friction on the ability to resist sliding is neglected. The effect of upward dynamic loads on overturning forces and moments is considered. Unless specifically specified otherwise, anchorage devices are designed in accordance with the requirements of ASME Code Section III, Division 1, Subsection NF, or the AISC Manual of Steel Construction and ACI 318.

Dynamic design data are provided in the form of acceleration response spectra for each floor area of the equipment. Dynamic data for the ground or building floor to which the equipment is attached is used. For the case of equipment having supports with different dynamic motions, the most severe floor response spectrum is applied to all of the supports.

Refer to Subsections 3.9.3.2.3.1.4 and 3.9.3.2.5.1.2 for additional information on the dynamic qualification of active pumps and valves, respectively.

3.9.2.2.2.16 Supports

Subsections 3.9.3.4 and 3.9.3.5 address analyses or tests that are performed for component supports to assure their structural capability to withstand the seismic and other dynamic excitations.

3.9.2.3 Dynamic Response of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

The major reactor internal components within the vessel are subjected to extensive testing coupled with dynamic system analyses to properly evaluate the resulting flow-induced vibration phenomena during normal reactor operation and from anticipated operational transients.

In general, the vibration forcing functions for operational flow transients and steady-state conditions are not predetermined by detailed analysis. Special analysis of the response signals measured for reactor internals of many similar designs are performed to obtain the parameters which determine the amplitude and model contributions in the vibration responses. These studies provide useful predictive information for extrapolating the results from tests of components with similar designs to components of different designs. This vibration prediction method is appropriate where standard hydrodynamic theory cannot be applied due to complexity of the structure and flow conditions. Elements of the vibration prediction method are outlined as follows:

(1) Dynamic analysis of major components and subassemblies is performed to identify vibration modes and frequencies. The analysis models used for Seismic Category I structures are similar to those outlined in Subsection 3.7.2.

(2) Data from previous plant vibration measurements are assembled and examined to identify predominant vibration response modes of major components. In general, response modes are similar but response amplitudes vary among BWRs of differing size and design.

(3) Parameters are identified which are expected to influence vibration response amplitudes among the several reference plants. These include hydraulic parameters such as velocity and steam flow rates and structural parameters such as natural frequency and significant dimensions.

(4) Correlation functions of the variable parameters are developed which, multiplied by response amplitudes, tend to minimize the statistical variability between plants. A correlation function is obtained for each major component and response mode.

(5) Predicted vibration amplitudes for components of the prototype plant are obtained from these correlation functions based on applicable values of the parameters for the prototype plant. The predicted amplitude for each dominant response mode is stated in terms of a range taking into account the degree of statistical variability in each of the correlations. The predicted mode and frequency are obtained from the dynamic analyses of item (1).

The dynamic modal analysis forms the basis for interpretation of the preoperational and initial startup test results (Subsection 3.9.2.4). Modal stresses are calculated and relationships are obtained between sensor response amplitudes and peak component stresses for each of the lower normal modes. The allowable amplitude in each mode is that which produces a peak stress amplitude of ± 10,000 psi.

Vibratory loads are continuously applied during normal operation and the stresses are limited to ± 10,000 psi to prevent fatigue failure. Prediction of vibration amplitudes, mode shapes, and frequencies of normal reactor operations are based on statistical extrapolation of actual measured results on the same or similar components in reactors now in operation

The dynamic loads due to flow-induced vibration from the feedwater jet impingement have no significant effect on the steam separator assembly. Analysis is performed to show that the impingement feedwater jet velocity is below the critical velocity. Also, it can be shown that the excitation frequency of the steam separator skirt is very different from the natural frequency of the skirt.

The calculated stresses due to hydrodynamic forces during core flooding operation are small and considered negligible when compared to the design-allowable stresses. Locations for which calculations were made include the weld joints, elbows, and rings.

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Reactor internals vibration measurement and inspection programs is conducted during preoperational and initial startup testing in accordance with guidelines of Regulatory Guide 1.20 for prototype reactor internals. These programs are conducted in the three phases described as follows:

(1) Preoperational tests prior to fuel loading. Steady-state test conditions include balanced recirculation system operation and unbalanced operation over the full range of flow rates up to rated flow. Transient flow conditions include single- and multiple pump trips from rated flow. This subjects major components to a minimum of 10^6 cycles of vibration at the anticipated dominant response frequency and at the maximum response amplitudes. Vibration measurements are obtained during this test and a close visual inspection of internals is conducted before and after the test.

(2) Precritical testing with fuel. This vibration measurement series is conducted with the reactor assembly complete but prior to reactor criticality. Flow conditions include balanced, unbalanced, and transient conditions as for the first test series. The purpose of this series is to verify the anticipated effects of the fuel on the vibration response of internals. Previous vibration measurements in BWRs (Reference 3) have shown that the fuel adds damping and reduces vibrations amplitudes of major internal structures; thus, the first test series (without fuel) is a conservative evaluation of the vibration levels of these structures

(3) Initial Startup testing. Vibration measurements are made during reactor startup at conditions up to 100% rated flow and power. Balance, unbalanced, and transient conditions of recirculation system operation will be evaluated. The primary purpose of this test series is to verify the anticipated effect of two-phase flow on the vibration response of internals. Previous vibration measurements in BWRs (Reference 3) have shown that the effect of the two-phase flow is to broaden the frequency response spectrum and diminish the maximum response amplitude of the shroud and core support structures.

Vibration sensor types may include strain gages, displacement sensors (linear variable transformers), and accelerometers.

Accelerometers are provided with double integration signal conditioning to give a displacement output. Sensor locations include the following:

(1) top of shroud head, lateral acceleration (displacement);

(2) top of shroud, lateral displacement;

(3) control rod drive housings, bending strain;

(4) incore housings. bending strain; and

(5) core flooder internal piping, bending strain.

In addition to these components, vibration of the core spray sparger is measured during preoperational testing of that system at the designated prototype.

In all prototype plant vibration measurements, only the dynamic component of strain or displacement is recorded. Data are recorded on magnetic tape and provision is made for selective online analysis to verify the overall quality and level of the data. Interpretation of the data requires identification of the dominant vibration modes of each component by the test engineer using frequency, phase, and amplitude information for the component dynamic analyses. Comparison of measured vibration amplitudes to predicted and allowable amplitudes is then to be made on the basis of the analytically obtained normal mode which best approximates the observed mode.

The visual inspections conducted prior to and following preoperational testing are for vibration, wear, or loose parts. At the completion of preoperational testing, the reactor vessel head and the shroud head are removed, the vessel is drained, and major components are inspected on a selected basis. The inspections cover the shroud, shroud head, core support structures, recirculation internal pumps, the peripheral control rod drive, and incore guide tubes. Access is provided to the reactor lower plenum for these inspections.

The analysis, design and/or equipment that are to be utilized in a facility will comply with Regulatory Guide 1.20 as explained below.

Regulatory Guide 1.20 describes a comprehensive vibration assessment program for reactor internals during preoperational and initial startup testing. The vibration assessment program meets the requirements of Criterion 1, Quality Standards and Record, Appendix A to 10CFR50 and Section 50.34, Contents of Applications; Technical Information, of 10CFR50. This Regulatory Guide is applicable to the core support structures and other reactor internals.

Vibration testing of reactor internals is performed on all GE-BWR plants. At the time of original issue of Regulatory Guide 1.20, test programs for compliance were instituted for the then designed reactors. The first ABWR plant is considered a prototype and is instrumented and subjected to preoperation and startup flow testing to demonstrate that flow-induced vibrations similar to those expected during operation will not cause damage. Subsequent plants which have internals similar to those of the prototypes are also tested in compliance with the requirements of Regulatory Guide 1.20. GE is committed to confirm satisfactory vibration performance of internals in these plants through preoperational flow testing followed by inspection for evidence of excessive vibration. Extensive vibration measurements in prototype plants together with satisfactory operating experience in all BWR plants have established the adequacy of reactor internal designs. GE continues these test programs for the generic plants to verify structural integrity and to establish the margin of safety.

See Subsection 3.9.7.1 for COL license information pertaining to the reactor internals vibration testing program.

3.9.2.5 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

The faulted events that are evaluated are defined in Subsection 3.9.5.2.1. The loads that occur as a result of these events and the analysis performed to determine the response of the reactor internals are as follows:

(1) **Reactor Internal Pressures** - The reactor internal pressure differentials (Figure 3.9-1a) due to assumed break of main steam or feedwater line are determined by analysis as described in Subsection 3.9.5.2.2. In order to assure that no significant dynamic amplification of load occurs as a result of the oscillatory nature of the blowdown forces during an accident, a comparison is made of the periods of the applied forces and the natural periods of the core support structures being acted upon by the applied forces. These periods are determined from a comprehensive vertical dynamic model of the RPV and internals with 12 degrees of freedom. Besides the real masses of the RPV and core support structures, account is made for the water inside the RPV.

(2) **External Pressure and Forces on the Reactor Vessel**-An assumed break of the main steam line, the feedwater line or the RHR line at the reactor vessel nozzle results in jet reaction and impingement forces on the vessel and asymmetrical pressurization of the annulus between the reactor vessel and the shield wall. These time-varying pressures are applied to the dynamic model of the reactor vessel system. Except for the nature and locations of the forcing functions, the dynamic model and the dynamic analysis method are identical to those for seismic analysis as described below. The resulting loads on the reactor internals, defined as LOCA loads, are considered as shown in Table 3.9.2.

(3) **Safety/Relief Valve Loads (SRV Loads)**-The discharge of the SRVs result in reactor building vibration (RBV) due to suppression pool dynamics as described in Appendix 3B. The response of the reactor

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internals to the RBV is also determined with dynamic model and dynamic analysis method described below for seismic analysis.

(4) **LOCA Loads**-The Assumed LOCA also results in RBV due to suppression pool dynamics as described in Appendix 3B and the response of the reactor internals are again determined with the dynamic model and dynamic analysis method used for seismic analysis. Various types of LOCA loads are identified on Table 3.9-2.

(5) **Seismic Loads**-The theory, methods, and computer codes used for dynamic analysis of the reactor vessel, internals, attached piping and adjoining structures are described in Section 3.7 and Subsection 3.9.1.2. Dynamic analysis is performed by coupling the lumped-mass model of the reactor vessel and internals with the building model to determine the system natural frequencies and mode shapes. The
210.33 relative displacement, acceleration, and load response is then determined by either the time-history method or the resonse-spectrum method. The load on the reactor internals due to faulted event SSE are obtained from this analysis.

The above loads are considered in combination as defined in Table 3.9-2. The SRV. LOCA (SBL, IBL or LBL) and SSE loads as defined in Table 3.9-2 are all assumed to act in the same direction. The peak colinear responses of the reactor internals to each of these loads are added by the square root of the sum of the squares (SRSS) method. The resultant stresses in the reactor internal structures are directly added with stress resulting from the static and steady state loads in the faulted load combination, including the stress due to peak reactor internal pressure differential during the LOCA. The reactor internals satisfy the stress deformation and fatigue limits as defined in Subsection 3.9.5.3.

3.9.2.6 Correlations of Reactor Internals Vibration Tests With the Analytical Results

Prior to initiation of the instrumented vibration measurement program for the prototype plant, extensive dynamic analyses of the reactor and internals are performed. The results of these analyses are used to generate the allowable vibration levels during the vibration test. The vibration data obtained during the test will be analyzed in detail.

The results of the data analyses, vibration amplitudes, natural frequencies, and mode shapes are then compared to those obtained from the theoretical analysis.

Such comparisons provide the analysts with added insight into the dynamic behavior of the reactor internals. The additional knowledge gained from previous vibration tests has been utilized in the generation of the dynamic models for seismic and loss of coolant accident (LOCA) analyses for this plant. The models used for this plant are similar to those used for the vibration analysis of earlier prototype BWR plants.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

This section delineates the criteria for selection and definition of design limits and loading combination associated with normal operation, postulated accidents, and specified seismic and other reactor building vibration (RBV) events for the design of safety-related ASME Code components (except containment components which are discussed in Section 3.8).

This section discusses the ASME Class 1, 2, and 3 equipment and associated pressure retaining parts and identifies the applicable loadings, calculation methods, calculated stresses, and allowable stresses. A discussion of major equipment is included on a component-by-component basis to provide examples. Design transients and dynamic loading for ASME Class 1, 2, and 3 equipment are covered in Subsection 3.9.1.1. Seismic-related loads and dynamic analyses are discussed in Section 3.7. The suppression pool-related RBV loads are described in Appendix 3B. Table 3.9-2 presents the combinations of dynamic events to be considered for the design and analysis of all ABWR ASME Code Class 1, 2, and 3 components, component supports, core support structures and equipment. Specific loading combinations considered for evaluation of each specific equipment are derived from Table 3.9-2 and are contained in the design specifications and/or design reports of the respective equipment. (See Subsection 3.9.7.4 for COL license information)

Table 3.9-2 also presents the evaluation models and criteria. The predicted loads or stresses and the design or allowable values for the most critical areas of each component are compared in accordance with the applicable code criteria or other limiting criteria. The calculated results meet the limits.

The design life for the ABWR Standard Plant is 60 years. A 60 year design life is a requirement for all major plant components with reasonable expection of meeting this design life. However, all plant operational components and equipment except the reactor vessel are designed to be replaceable, design life not withstanding. The design life requirement allows for refurbishment and repair, as appropriate, to assure the design life of the overall plant is achieved. In effect, essentially all piping systems, components and equipment are designed for a 60 year design life. Many of these components are classified as ASME Class 2 or 3 or Quality Group D. Applicants referencing the ABWR design will identify these ASME Class 2, 3 amd Quality Group D components and provide the analyses required by the ASME Code, Subsection NB. These analysis will include the appropriate operating vibration loads and for the effects of mixing hot and cold fluids.

3.9.3.1.1 Plant Conditions

All events that the plant will or might credibly experience during a reactor year are evaluated to establish design basis for plant equipment. These events are divided into four plant conditions. The plant conditions described in the following paragraphs are based on event probability (i.e., frequency of occurrence as discussed in Subsection 3.9.3.1.1.5) and correlated to service levels for design limits defined in the ASME Boiler and Pressure Vessel Code Section III as shown in Tables 3.9-1 and 3.9-2.

3.9.3.1.1.1 Normal Condition

Normal conditions are any conditions in the course of system startup, operation in the design power range, normal hot standby (with condenser available), and system shutdown other than upset, emergency, faulted, or testing.

3.9.3.1.1.2 Upset Condition

An upset condition is any deviation from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include system operational transients (SOT) which result from any single operator error or control malfunction, from a fault in a system component requiring its isolation from the system, from a loss of load or power, or from an operating basis earthquake. Hot standby with the main condenser isolated is an upset condition.

3.9.3.1.1.3 Emergency Condition

An emergency condition includes deviations from normal conditions which require shutdown for correction of the condition(s) or repair of damage in the reactor coolant pressure boundary (RCPB). Such conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. Emergency condition events include but are not limited to infrequent operational transients (IOT) caused by one of the following: (a) a multiple valve blowdown of the reactor vessel; (b) LOCA from a small break or crack (SBL) which does not depressurize the reactor systems, does not actuate automatically the ECCS operation, nor results in leakage beyond normal makeup system capacity, but which requires the safety functions of isolation of containment and shutdown and may involve inadvertent actuation of automatic depressurization system (ADS); (c) improper assembly of the core during refueling; or (d) improper or sudden start of one recirculation pump. Anticipated transient without scram (ATWS) or reactor overpressure with delayed scam (see Tables 3.9-1 and 3.9-2) is an IOT classified as an emergency condition.

3.9.3.1.1.4 Faulted Condition

A faulted condition is any of those combinations of conditions associated with extremely low-probability postulated events whose consequences are such that the integrity and operability of the system may be impaired to the extent that considerations of public health and safety are involved. Faulted conditions encompass events, such as LOCA, that are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These events are the most drastic that must be considered in the design and thus represent limiting design bases. Faulted condition events include but are not limited to one of the following: (a) a control rod drop accident; (b) a fuel-handling accident; (c) a main steam line or feedwater line break; (d) the combination of any small/intermediate break LOCA (SBL or IBL) with the safe shutdown earthquake, and a loss of offsite power; or (e) the safe shutdown earthquake plus large break LOCA (LBL) plus a loss of offsite power.

The IBL classification covers those breaks for which the ECCS system operation will occur during the blowdown, and which results in reactor depressurization. The LBL classification covers the sudden, double ended severance of a main steam line inside or outside the containment that results in transient reactor depressurization, or any pipe rupture of equivalent flow cross sectional area with similar effects.

3.9.3.1.1.5 Correlation of Plant Condition with Event Probability

The probability of an event occurring per reactor year associated with the plant conditions is listed below. This correlation identifies the appropriate plant conditions and assigns the appropriate ASME Section III service levels for any hypothesized event or sequence of events.

Plant Condition	ASME Code Service Level	Event Encounter Probability per Reactor Year
Normal (planned)	A	1.0
Upset (moderate probability)	B	$1.0 > P \geq 10^{-2}$
Emergency (low probability)	C	$10^{-2} > P \geq 10^{-4}$
Faulted (extremely low probability)	D	$10^{-4} > P > 10^{-6}$

3.9.3.1.1.6 Safety Class Functional Criteria

For any normal or upset design condition event Safety Class 1, 2, and 3 equipment and piping (see Subsection 3.2.3) shall be capable of accomplishing its safety functions as required by the event and shall incur no permanent changes that could deteriorate its ability to accomplish its safety functions as required by any subsequent design condition event.

For any emergency or faulted design condition event, Safety Class 1, 2, and 3 equipment and piping shall be capable of accomplishing its safety functions as required by the event but repairs could be required to ensure its ability to accomplish its safety functions as required by any subsequent design condition event.

Specific stress criteria to meet the functional requirements are identified in a footnote to Table 3.9-2.

3.9.3.1.2 Reactor Pressure Vessel Assembly

The reactor vessel assembly consists of the reactor pressure vessel, vessel support skirt, and shroud support.

The reactor pressure vessel, vessel support skirt, and shroud support are constructed in accordance with the ASME Boiler and Pressure Vessel Code Section III. The shroud support consists of the shroud support plate and the shroud support cylinder and its legs. The reactor pressure vessel assembly components are classified as an ASME Class 1. Complete stress reports on these components are prepared in accordance with ASME Code requirements. NUREG-0619 (Reference 5) is also considered for feedwater nozzle and other such RPV inlet nozzle design.

The stress analysis is performed on the reactor pressure vessel, vessel support skirt, and shroud support for various plant operating conditions (including faulted conditions) by using the elastic methods except as noted in Subsection 3.9.1.4.2. Loading conditions, design stress limits, and methods of stress analysis for the core support structures and other reactor internals are discussed in Subsection 3.9.5.

3.9.3.1.3 Main Steam (MS) System Piping

210.6 210.5 210.36 The piping systems extending from the reactor pressure vessel to and including the outboard main steam isolation valve are constructed in accordance with the ASME Boiler and Pressure Vessel Code Section III, Class 1 criteria. The rules contained in Appendix F of ASME Code Section III are used in evaluating faulted loading conditions independently of other design and operating conditions. Stresses calculated on an elastic basis are evaluated in accordance with F-1360.

The MS system piping extending from the outboard main steam isolation valve to the turbine stop valve is constructed in accordance with the ASME Boiler and Pressure Vessel Code Section III, Class 2 Criteria. 210.6

3.9.3.1.4 Recirculation Motor Cooling (RMC) Subsystem

The RMC system piping loop between the recirculation motor casing and the heat exchanger is constructed in accordance with the ASME Boiler and Pressure Vessel Code Section III, Subsection NB-3600. The rules contained in Appendix F of ASME Code Section III are used in evaluating faulted loading conditions independently of all other design and operating conditions. Stresses calculated on an elastic basis are evaluated in accordance with F-1360. 210.36

3.9.3.1.5 Recirculation Pump Motor Pressure Boundary

The motor casing of the recirculation internal pump is a part of and welded into an RPV nozzle and is constructed in accordance with the requirements of an ASME Boiler and Pressure Vessel Code Section III, Class 1 component. The motor cover is a part of the pump/motor assembly and is constructed as an ASME Class 1 component. These pumps are not required to operate during the safe shutdown earthquake or after an accident. 210.36

3.9.3.1.6 Standby Liquid Control (SLC) Tank

The standby liquid control tank is constructed in accordance with the requirements of an ASME Boiler and Pressure Vessel Code Section III, Class 2 component. 210.36

3.9.3.1.7 RRS and RHR Heat Exchangers

The primary and secondary sides of the RRS (reactor recirculation system) are constructed in accordance with the requirements of an ASME Boiler and Pressure Vessel Code Section III, Class 1 and Class 2 component, respectively. The primary and secondary side of the RHR system heat exchanger is constructed as an ASME class 2 and class 3 component respectively. 210.36

3.9.3.1.8 RCIC Turbine

Although not under the jurisdiction of the ASME Code, the RCIC turbine is designed and evaluated and fabricated following the basic guidelines of ASME Code Section III for Class 2 components.

3.9.3.1.9 ECCS Pumps

210.36 The RHR, RCIC, and HPCF pumps are constructed in accordance with the requirements of an ASME Code Section III, Class 2 component.

3.9.3.1.10 Standby Liquid Control (SLC) Pump

210.36 The SLC system pump is constructed in accordance with the requirements for ASME Code Section III, Class 2 component.

3.9.3.1.11 Standby Liquid Control (SLC) Valve (Injection Valve)

210.36 The SLC system injection valve is constructed in accordance with the requirements for ASME Code Section III, Class 1 component.

3.9.3.1.12 Main Steam Isolation and Safety/Relief Valves

210.36 The main steam isolation valves and SRVs are constructed in accordance with ASME Boiler and Pressure Vessel Code Section III, Subsection NP-3500, requirements for Class 1 component.

3.9.3.1.13 Safety/Relief Valve Piping

The relief valve discharge piping extending from the relief valve discharge flange to the diaphram floor penetration is constructed in accordance with ASME Boiler and Pressure Vessel Code Section III, requirements for Class 3 components. The relief valve discharge piping extending from the diaphram floor penetration to the quenchers is constructed in accordance with ASME Boiler and Pressure Vessel Code, Section III, requirements for Class 2 components.

3.9.3.1.14 Reactor Water Cleanup (RWCU) System Pump and Heat Exchangers

The RWCU pump and heat exchangers (regenerative and nonregenerative) are not part of a safety system and are non-Seismic Category I equipment. ASME Boiler and Pressure Vessel Code Section III for Class 3 components is used as a guide in constructing the RWCU System pump and heat exchanger components. 210.36

3.9.3.1.15 Fuel Pool Cooling and Cleanup System Pumps and Heat Exchangers

The pumps and heat exchangers are constructed in accordance with the requirements for ASME Boiler and Pressure Vessel Code Section III, Class 3 component. 210.36

3.9.3.1.16 ASME Class 2 and 3 Vessels

The Class 2 and 3 vessels (all vessels not previously discussed) are constructed in accordance with the ASME Boiler and Pressure Vessel Code Section III. The stress analysis of these vessels is performed using elastic methods. 210.36

3.9.3.1.17 ASME Class 2 and 3 Pumps

The Class 2 and 3 pumps (all pumps not previously discussed) are designed and evaluated in accordance with the ASME Boiler and Pressure Vessel Code Section III. The stress analysis of these pumps is performed using elastic methods. See Subsection 3.9.3.2 for additional information on pump operability. 210.36

3.9.3.1.18 ASME Class 1, 2 and 3 Valves

The Class 1, 2, and 3 valves (all valves not previously discussed) are constructed in accordance with the ASME Boiler and Pressure Vessel Code Section III. 210.36

All valves and their extended structures are designed to withstand the accelerations due to seismic and other RBV loads. The attached piping is supported so that these accelerations are not exceeded. The stress analysis of these valves is performed using elastic methods. See Subsection 3.9.3.2 for additional information on valve operability.

3.9.3.1.19 ASME Class 1, 2 and 3 Piping

The Class 1, 2 and 3 piping (all piping not previously discussed) is constructed in accord- 210.36

ance with the ASME Boiler and Pressure Vessel Code Section III. For Class 1 piping, for the faulted plant condition, stresses are calculated on an elastic basis and evaluated in accordance with Appendix F of the Code. For Class 2 and 3 piping, stresses are calculated on an elastic basis and evaluated in accordance with NC/ND-3600 of the Code.

3.9.3.2 Pump and Valve Operability Assurance

Active mechanical (with or without electrical operation) equipment are Seismic Category I and each is designed to perform a mechanical motion for its safety-related function during the life of the plant under postulated plant conditions. Equipment with faulted condition functional requirements include active pumps and valves in fluid systems such as the residual heat removal system, emergency core cooling system, and main steam system.

This Subsection discusses operability assurance of active ASME Code Section III pumps and valves, including motor, turbine or operator that is a part of the pump or valve (See Subsection 3.9.2.2).

Safety-related valves and pumps are qualified by testing and analysis and by satisfying the stress and deformation criteria at the critical locations within the pumps and valves. Operability is assured by meeting the requirements of the programs defined in Subsection 3.9.2.2, Section 3.10, Section 3.11 and the following subsections.

Section 4.4 of GE's Environmental Qualification Program (Reference 6) applies to this subsection, and the seismic qualification methodology presented therein is applicable to mechanical as well as electrical equipment.

3.9.3.2.1 ECCS Pumps, Motors and Turbine

Dynamic qualification of the ECCS (RHR, RCIC and HPCF) pumps with motor or turbine assembly is also described in Subsections 3.9.2.2.2.6 and 3.9.2.2.2.7.

3.9.3.2.1.1 Consideration of Loading, Stress, and Acceleration Conditions in the Analysis

In order to avoid damage to the ECCS pumps during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, other RBV loads, and dynamic system loads are limited to the material elastic limit. A three dimensional finite-element model of the pump and associated motor (see Subsections 3.9.3.2.2 and 3.9.3.2.1.5 for RCIC pump and turbine, respectively) and its support is developed and analyzed using the response spectrum and the dynamic analysis method. The same is analyzed due to static nozzle loads, pump thrust loads, and dead weight. Critical location stresses are compared with the allowable stresses and the critical location deflections with the allowables; and accelerations are checked to evaluate operability. The average membrane stress σm for the faulted condition loads is limited to 1.2S or approximately 0.75 σ_y (σ_y = yield stress), and the maximum stress in local fibers (σm + bending stress σb) is limited to 1.8S or approximately 1.1 σ_y. The max- imum faulted event nozzle loads are also con- sidered in an analysis of the pump supports to assure that a system misalignment cannot occur.

Performing these analyses with the conservative loads stated and with the restrictive stress limits as allowables assures that critical parts of the pump and associated motor or turbine will not be damaged during the faulted condition and that the operability of the pump for post-faulted condition operation will not be impaired.

3.9.3.2.1.2 Pump/Motor Operation During and Following Dynamic Loading

Active ECCS pump/motor rotor combinations are designed to rotate at a constant speed under all conditions. Motors are designed to withstand short periods of severe overload. The high rotary inertia in the operating pump rotor and the nature of the random short duration loading characteristics of the dynamic event prevents the rotor from becoming seized. The seismic and other RBV loadings can be predicted to require only a slight increase, if any, in the torque (i.e., motor current) necessary to drive the pump at the constant design speed; therefore, the pump is expected to operate at the design speed during the faulted event loads.

The functional ability of the active pumps after a faulted condition is assured since only normal operating loads and steady state nozzle loads exist. For the active pumps, the faulted condition loads are greater than the normal condition loads only due to the SSE and other RBV transitory loads. These faulted events are infrequent and of relatively short duration compared to the design life of the equipment. Since it is demonstrated that the pumps would not be damaged during the faulted condition, the post-faulted condition operating loads will be no worse than the normal plant operating limits. This is assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and post-faulted conditions be limited to the magnitudes of the normal condition nozzle loads. The post-faulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

3.9.3.2.1.3 ECCS Pumps

All active ECCS (RHR, RCIC and HPCF) pumps are qualified for operability by first being subjected to rigid tests both prior to installation in the plant and after installation in the plant. The in-shop tests include: (1) hydrostatic tests of pressure-retaining parts of 125% of the design pressure; (2) seal leakage tests; and (3) performance tests while the pump is operated with flow to determine total developed head, minimum and maximum head and net positive suction head (NPSH) requirements. Also monitored during these operating tests are bearing temperatures (except water cooled bearings) and vibration levels. Both are shown to be below specified limits. After the pump is installed in the plant, it undergoes the cold hydro tests, functional tests, and the required periodic inservice inspection and operation. These tests demonstrate reliability of the pump for the design life of the plant.

In addition to these tests, these pumps are analyzed for operability during a faulted condition by assuring that (1) the pump will not be damaged during the dynamic (SSE and LOCA) event, and (2) the pump will continue operating despite the dynamic loads.

3.9.3.2.1.4 ECCS Motors

Qualification of the Class 1E motors used for the ECCS motors complies with IEEE 323. The qualification of all motor sizes is based on completion of a type test, followed up with review and comparison of design and material details, and seismic and other RBV loads analyses of production units, ranging from 600 to 3500 Bhp, with the motor used in the type test. All manufacturing, inspection, and routine tests by motor manufacturer on production units are performed on the test motor.

The type test is performed on a 1250-hp vertical motor in accordance with IEEE 323, first simulating a normal operation during the design life, then subjecting the motor to a number of vibratory tests, and then to the abnormal environmental condition possible during and after a LOCA. The test plans for the type test is as follows:

(1) Thermal aging of the motor electrical insulation system (which is a part of the stator only) is based on extrapolation in accordance with the temperature life characteristic curve from IEEE 275 for the insulation type used on the ECCS motors. The amount of aging equals the total estimated operation days at maximum insulation surface temperature.

(2) Radiation aging of the motor electrical insulation equals the maximum estimated integrated dose of gamma during normal and abnormal conditions.

(3) The normal operational induced current vibration effect on the insulation system is simulated by 1.5g horizontal vibration

acceleration at current frequency for one hour duration.

(4) The dynamic load deflection analysis on the rotor shaft is performed to ensure adequate rotation clearance, and is verified by static loading and deflection of the rotor for the type test motor.

(5) Dynamic load aging and testing is performed on a biaxial test table in accordance with IEEE 344. During this test, the shake table is activated to simulate the maximum design limit for the safe shutdown earthquake and other RBV loads with as many motor starts and operation combinations consistent with the plant events of Table 3.9-1 and the ECCS inadvertent injections and tests planned over the life of the plant.

(6) An environmental test simulating a LOCA condition with a duration of 100 days is performed with the test motor fully loaded, simulating pump operation. The test consists of startup and six hours operation at 212°F ambient temperature and 100% steam environment. Another startup and operation of the test motor after one hour standstill in the same environment is followed by sufficient operation at high humidity and temperature based on extrapolation in accordance with the temperature life characteristic curve from IEEE 275 for the insulation type used on the ECCS motors.

3.9.3.2.1.5 RCIC Turbine

The RCIC turbine is qualified by a combination of static analysis and dynamic testing as described in Subsection 3.9.2.2.2.7. The turbine assembly consists of rigid masses (wherein static analysis is utilized) interconnected with control levers and electronic control systems, necessitating final qualification by dynamic testing. Static loading analysis has been employed to verify the structural integrity of the turbine assembly, and the adequacy of bolting under operating and dynamic conditions. The complete turbine assembly is qualified via dynamic testing, in accordance with IEEE 344. The qualification test program includes demonstration of startup capability, as well as operability during dynamic loading conditions. Operability under normal load conditions is assured by comparison to operability of similar turbines in operating plants.

3.9.3.2.2 SLC Pump and Motor Assembly and RCIC Pump Assembly

These equipment assemblies are small, compact, rigid assemblies with natural frequencies well above 33 Hz. With this fact verified, each equipment assembly is qualified by the static analysis for seismic and other RBV loads. This qualification assures structural loading stresses within Code limitations, and verifies operability under seismic and other RBV loads. This is also discussed in Subsections 3.9.2.2.2.8 and 3.9.2.2.2.7.

3.9.3.2.3 Other Active Pumps

The active pumps not previously discussed are ASME Class 2 or 3 and Seismic Category I. They are designed to perform their function including all required mechanical motions during and after a dynamic (seismic and other RBV) loads event and to remain operative during the life of the plant.

The program for the qualification of Seismic Category I components conservatively demonstrates that no loss of function results either before, during, or after the occurrence of the combination of events for which operability must be assured. No loss of function implies that the pressure boundary integrity will be maintained, that the component will not be caused to operate improperly, and that components required to respond actively will respond properly as appropriate to the specific equipment. In general, operability assurance is established during and after the dynamic loads event for active components.

3.9.3.2.3.1 Procedures

Procedures have been established for qualifying the mechanical portions of Seismic Category I pumps such as the body which forms a fluid pressure boundary including the suction and discharge nozzles, the shaft and seal retainers, the impeller assembly including the blading, shaft, and bearings for active pumps, and integral supports.

All active pumps are qualified for operability by first being subjected to rigid tests both prior to installation and after installation in the plant. Electric motors for active pumps and instrumentation, including electrical devices which must function to cause the pump to accomplish its intended function, are discussed separately in Subsection 3.9.3.2.5.1.3.

3.9.3.2.3.1.1 Hydrostatic Test

All seismic-active pumps shall meet the hydrostatic test requirements of ASME Code Section III according to the class rating of the given pump.

3.9.3.2.3.1.2 Leakage Test

The fluid pressure boundary is examined for leaks at all joints, connections, and regions of high stress such as around openings or thickness transition sections while the pump is undergoing a hydrostatic test or during performance testing. Leakage rates that exceed the rates permitted in the design specification are eliminated and the component retested to establish an observed leakage rate. The actual observed leakage rate, if less than permitted, is documented and made a part of the acceptable documentation package for the component.

3.9.3.2.3.1.3 Performance Test

The pump is demonstrated capable of meeting all hydraulic requirements while operating with flow at the total developed head, minimum and maximum head, NPSH, and other parameters as specified in the equipment specification.

Bearing temperature (except water cooled bearings) and vibration levels are also monitored during these operating tests. Both are shown to be below specified levels.

3.9.3.2.3.1.4 Dynamic Qualification

The safety-related active pumps are analyzed for operability during dynamic loading event by assuring that the pump is not damaged during the seismic event and the pump continues operating despite the dynamic loads.

A test or dynamic analysis is performed for a pump to determine the dynamic seismic and other RBV load from the applicable floor response spectra.

Response spectra for the horizontal vibration are used in two orthogonal horizontal directions simultaneously with the response spectra for the vertical vibration. The effects from the three simultaneous accelerations are combined by the square root of the sum of the squares method. The pump is demonstrated by test or analysis that the faulted condition nozzle loads do not impair the operability of the pumps during or following the faulted condition. Components of the pump are considered essentially rigid when having a natural frequency above 33 Hz. A static shaft deflection analysis of the motor rotor is performed with the conservative SSE accelerations acting in horizontal and vertical direction simultaneously.

The deflections determined from the static shaft analysis are compared to the allowable rotor clearances. The allowable rotor clearances are limited by the deflection which would cause the rotor to just make contact with the stator. In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE and dynamic system loads are limited to the material elastic limit.

The average membrane stress (σ_m) for the faulted conditions loads is limited to 1.2S or approximately 0.75 σ_y (σ_y = yield stress), and the maximum stress in local fibers (σm + bending stress σb) is limited to 1.8S or approximately 1.1 σ_y. The maximum dynamic nozzle loads are also considered in an analysis of the pump supports to assure that a system misalignment cannot occur.

If the natural frequency is found to be below 33 Hz, an analysis is performed to determine the amplified input accelerations necessary to perform the static analysis.

In completing the seismic qualification procedures, the pump motor and all components vital to the operation of the pump are independently qualified for operation during the maximum seismic event by IEEE 344.

If the testing option is chosen, sine-beat testing for electrical equipment is performed by satisfying one or more of the following requirements to demonstrate multi-frequency response is negligible or the sine-beat input is of sufficient magnitude to conservatively account for this effect.

(1) The equipment response is basically due to one mode.

(2) The sine-beat response spectra envelope the floor response spectra in the region of significant response.

(3) The floor response spectra consist of one dominant mode and has a peak at this frequency.

The degrees of cross coupling in the equipment shall determine if a single or multi-axis test is required. Multi-axis testing is required if there is considerable cross coupling. If coupling is very light, then single axis testing is justified. Or, if the degree of coupling can be determined, then single-axis testing can be used with input sufficiently increased to include the effect of coupling on the response of the equipment.

The combined stresses of the support structures are designed to be within the limits of ASME Code Section III, Subsection NF, component Support Structures and/or other comparable limits of industry standards such as the AISC Specification for Buildings, plus Addenda for building support structures.

An analysis or test is accomplished which conservatively demonstrates structural integrity and/or functionality of the equipment supports.

The impeller, shaft, and bearings for active pumps are analyzed to determine adequacy while operating with the seismic and other RBV loading effects applied in addition to the applicable operating loads including nozzle loads. Functional requirements are partially demonstrated by a suitable analysis which conservatively shows the following:

(1) The stresses in the shaft do not exceed the minimum yield strength of the material used for its construction.

(2) The deflections of the shaft and/or impeller blades do not cause the impeller assembly to seize.

(3) The bearing temperature does not attain limits which may allow stresses in the bearing or bearing support to exceed minimum yield strength levels or jeopardize lubrication.

3.9.3.2.3.2 Documentation

All of the preceding requirements (Subsection 3.9.3.2.3.1) are satisfied to demonstrate that functionality is assured for active pumps. The documentation is prepared in a format that clearly shows that each consideration has been properly evaluated and tests have been validated by a designated quality assurance representative. The analysis is included as a part of the certified stress report for the assembly.

3.9.3.2.4 Major Active Valves

Some of the major safety-related active valves (see Table 6.2-2, 6.2-3 and 3.2-1) discussed in this subsection for illustration are the main steam line isolation valves and safety/relief valves, and standby liquid control valves and high pressure core flooder valves (motor-operated). These valves are designed to meet the ASME Code Section III requirements and perform their mechanical motion in conjunction with a dynamic (SSE and other RBV) load event. These valves are supported entirely by the piping, i. e., the valve operators are not used as attachment points for piping supports (See Subsection 3.9.3.4.1). The dynamic qualification for operability is unique for each valve type; therefore, each method of qualification is detailed individually below.

3.9.3.2.4.1 Main Steam Isolation Valve

The typical Y-pattern MSIVs described in Subsection 5.4.5.2 are evaluated by analysis and test for capability to operate under the design loads that envelop the predicted loads during a design basis accident and safe shutdown earthquake.

The valve body is designed, analyzed and tested in accordance with the ASME Code Section III, Class 1 requirements. The MSIVs are modeled mathematically in the main steam line system analysis. The loads, amplified accelerations and resonance frequencies of the valves are determined from the overall steamline analysis. The piping supports (snubbers, rigid restraints, etc.) are located and designed to limit amplified accelerations of and piping loads in the valves to the design limits.

As described in Subsection 5.4.5.3, the MSIV and associated electrical equipment (wiring, solenoid valves, and position switches) are dynamically qualified to operate during an accident condition.

3.9.3.2.4.2 Main Steam Safety/Relief Valve

The typical SRV design described in Subsection 5.2.2.4.1 is qualified by type test to IEEE 344 for operability during a dynamic event. Structural integrity of the configuration during a dynamic event is demonstrated by both Code (ASME Class 1) analysis and test.

(1) Valve is designed for maximum moments on inlet and outlet which may be imposed when installed in service. These moments are resultants due to dead weight plus dynamic loading of both valve and connecting pipe, thermal expansion of the connecting pipe, and reaction forces from valve discharge.

(2) A production SRV is demonstrated for operability during a dynamic qualification (shake table) type test with moment and "g" loads applied greater than the required equipment's design limit loads and conditions.

A mathematical model of this valve is included in the main steam line system analysis, as with the MSIVs. This analysis assures the equipment design limits are not exceeded.

3.9.3.2.4.3 Standby Liquid Control Valve (Injection Valve)

The typical SLC Injection Valve design is qualified by type test to IEEE 344. The valve body is designed, analyzed and tested per the ASME Code, Section III, Class 1. The qualification test demonstrates the ability to remain operable after the application of the horizontal and vertical dynamic loading exceeding the predicted dynamic loading.

3.9.3.2.4.4 High Pressure Core Flooder Valve (Motor-Operated)

The typical HPCF valve body design, analysis and testing is in accordance with the requirements of the ASME Code Section III, Class 1 or 2 components. The Class 1E electrical motor actuator is qualified by type test in accordance with IEEE 382, as discussed in Subsection 3.11.2. A mathematical model of this valve is included in the HPCF piping system analysis. The analysis results are assured not to exceed the horizontal and vertical dynamic acceleration limits acting simultaneously for a dynamic (SSE and other RBV) event, which is treated as an emergency condition.

3.9.3.2.5 Other Active Valves

Other safety-related active valves are ASME Class 1, 2 or 3 and are designed to perform their mechanical motion during dynamic loading conditions. The operability assurance program ensures that these valves will operate during a dynamic seismic and other RBV event.

3.9.3.2.5.1 Procedures

Qualification tests accompanied by analyses are conducted for all active valves. Procedures for qualifying electrical and instrumentation components which are depended upon to cause the valve to accomplish its intended function are described in Subsection 3.9.3.2.5.1.3.

3.9.3.2.5.1.1 Tests

Prior to installation of the safety-related valves, the following tests are performed: (1) shell hydrostatic test to ASME Code Section III requirements; (2) back seat and main seat leakage tests; (3) disc hydrostatic test; (4) functional tests to verify that the valve will open and close within the specified time limits when subject to the design differential pressure; and (5) operability qualification of valve actuators for the environmental conditions over the installed life. Environmental qualification procedures for operation follow those specified in Section 3.11. The results of all required tests are properly documented and included as a part of the operability acceptance documentation package.

3.9.3.2.5.1.2 Dynamic Load Qualification

The functionality of an active valve during and after a seismic and other RBV event may be demonstrated by an analysis or by a combination of analysis and test. The qualification of electrical and instrumentation components controlling valve actuation is discussed in Subsection 3.9.3.2.5.1.3. The valves are designed using either stress analyses or the pressure temperature rating requirements based upon design conditions. An analysis of the extended structure is performed for static equivalent dynamic loads applied at the center of gravity of the extended structure. See Subsection 3.9.2.2 for further details.

The maximum stress limits allowed in these analyses confirm structural integrity and are the limits developed and accepted by the ASME for the particular ASME Class of valve analyzed. Additional detail on stress limits for operability is provided in a footnote to Table 3.9-2.

Dynamic load qualification is accomplished in the following way:

(1) All the active valves are designed to have a fundamental frequency which is greater than the high frequency asymptote (ZPA) of the dynamic event. This is shown by suitable test or analysis.

(2) The actuator and yoke of the valve system is statically loaded to an amount greater than that due to a dynamic event. The load is applied at the center to gravity of the actuator alone in the direction of the weakest axis of the yoke. The simulated operational differential pressure is simultaneously applied to the valve during the static deflection tests.

(3) The valve is then operated while in the deflected position (i.e., from the normal operating position to the safe position). The valve is verified to perform its safety-related function within the specified operating time limits.

(4) Motor operators and other electrical appurtenances necessary for operation are qualified as operable during a dynamic event by appropriate qualification tests prior to installation on the valve. These motor operators then have individual Seismic Category I supports attached to decouple the dynamic loads between the operators and valves themselves.

The piping, stress analysis, and pipe support design maintain the motor operator accelerations below the qualification levels with adequate margin of safety.

If the fundamental frequency of the valve, by test or analysis, is less than that for the ZPA, a dynamic analysis of the valve performed to determine the equivalent acceleration to be applied during the static test. The analysis provides the amplification of the input acceleration considering the natural frequency of the valve and the frequency content of the applicable plant floor response spectra. The adjusted accelerations have been determined using the same conservatism contained in the horizontal and vertical accelerations used for rigid valves. The adjusted acceleration is then used in the static analysis and the valve operability is assured by the methods outlined in Steps (2) through (4), using the modified acceleration input. Alternatively, the valve including the actuator and all other accessories is qualified by shake table test.

Valves which are safety-related but can be classified as not having an overhanging structure, such as check valves and pressure-relief valves, are considered as follows:

3.9.3.2.5.1.2.1 Active Check Valves

Due to the particular simple characteristics of the check valves, the active check valves are qualified by a combination of the following tests and analysis:

(1) Stress analysis including the dynamic loads where applicable;

(2) in-shop hydrostatic tests;

(3) in-shop seat leakage test; and

(4) periodic in-situ valve exercising and inspection to assure the functional capability of the valve.

3.9.3.2.5.1.2.2 Active Pressure-Relief Valves

The active pressure-relief valves (RVs) are qualified by the following procedures. These valves are subjected to test and analysis similar to check valves, stress analyses including the dynamic loads, in-shop hydrostatic seat leakage, and performance tests. In addition to these tests, periodic in-situ valve inspection, as applicable, and periodic valve removal, refurbishment, performance testing, and reinstallation are performed to assure the functional capability of the valve. Tests of the RV under dynamic loading conditions demonstrate that valve actuation can occur during application of the loads. The tests include pressurizing the valve inlet with nitrogen and subjecting the valve to accelerations equal to or greater than the dynamic event (SSE plus other RBV) loads.

3.9.3.2.5.1.3 Qualification of Electrical and Instrumentation Components Controlling Valve Actuation

A practical problem arises in attempting to describe tests for devices (relays, motors, sensors, etc.) as well as for complex assemblies such as control panels. It is reasonable to assume that a device, as an integral part of an assembly, can be subjected to dynamic loads tests while in an operating condition and its performance monitored during the test. However, in the case of complex panels, such a test is not always practical. In such a situation, the following alternate approach is recommended.

The individual devices are tested separately in an operating condition and the test levels recorded as the qualification levels of the devices. The panel, with similar devices installed but inoperative, is vibration tested to determine if the panel response accelerations as measured by accelerometers installed at the device attachment locations are less than the levels at which the devices were qualified. Note that the purpose of installing the nonoperating devices is to assure that the panel has the structural characteristics it will have when in use. If the acceleration levels at the device locations are found to be less than the levels to which the device is qualified, then the total assembly is considered qualified. Otherwise, either the panel is redesigned to reduce the acceleration level to the device locations and retested, or the devices is requalified to the higher levels.

3.9.3.2.5.2 Documentation

All of the preceding requirements (Subsection 3.9.3.2.5.1) are satisfied to demonstrate that functionality is assured for active valves. The documentation is prepared in a format that clearly shows that each consideration has been properly evaluated and tests have been validated by a designated quality assurance representative. The analysis is included as a part of the certified stress report for the assembly.

3.9.3.3 Design and Installation of Pressure Relief Devices

3.9.3.3.1 Main Steam Safety/Relief Valves

SRV lift in a main steam (MS) piping system results in a transient that produces momentary unbalanced forces acting on the MS and SRV discharge piping system for the period from opening of the SRV until a steady discharge flow from the reactor pressure vessel to the suppression pool is established. This period includes clearing of the water slug from the end of the discharge piping submerged in the suppression pool. Pressure waves traveling through the main steam and discharge piping following the relatively rapid opening of the SRV cause this piping to vibrate.

The analysis of the MS and discharge piping transient due to SRV discharge consists of a stepwise time-history solution of the fluid flow equation to generate a time history of the fluid properties at numerous locations along the pipe. The fluid transient properties are calculated based on the maximum set pressure specified in the steam system specification and the value of ASME Code flow rating increased by a factor to account for the conservative method of establishing the rating. Simultaneous discharge of all valves in a MS line is assumed in the analysis because simultaneous discharge is considered to induce maximum stress in the piping. Reaction loads on the pipe are determined at each location corresponding to the position of an elbow. These loads are composed of pressure-times-area, momentum-change, and fluid-friction terms.

The method of analysis applied to determine response of the MS piping system including the SRV discharge line, to relief valve operation is time-history integration. The forces are applied at locations on the piping system where fluid flow changes direction thus causing momentary reactions. The resulting loads on the SRV, the main steamline, and the discharge piping are combined with loads due to other effects as specified in Subsection 3.9.3.1. In accordance with Tables 3.9-1 and 3.9-2, the Code stress limits for service levels corre- sponding to load combination classification as normal, upset, emergency, and faulted are applied to the main steam and discharge pipe.

3.9.3.3.2 Other Safety/Relief Valves

An SRV is identified as a pressure relief valve or vacuum breaker. SRVs in the reactor components and subsystems are described and identified in Subsection 5.4.13.

The operability assurance program discussed in Subsection 3.9.3.2.5 applies to safety/relief valves. The qualification of active relief valves is specifically outlined in Subsection 3.9.3.2.5.1.2.2.

ABWR safety/relief valves (safety valves with auxiliary actuating devices and pilot operated valves) are designed and manufactured in accordance with the ASME Code, Section III, Division 1 requirements. Specific rules for pressure relieving devices are as specified in Article NB-7000, and NB-3500 (pilot operated and power actuated pressure relief valves).

The design of ABWR SRVs incorporates SRV opening and pipe reaction load considerations required by ASME III, Appendix O, and including the additional criteria of SRP, Section 3.9.3, Paragraph II.2 and those identified under Subsection NB-3658 for pressure and structural integrity. Safety/relief valve operability is demonstrated either by dynamic testing or analysis of similarly tested valves or a combination of both in compliance with the requirements of SRP Subsection 3.9.3. 210.38

3.9.3.3.3 Rupture Disks

There are no rupture disks in the ABWR plant design, that must function during and after a dynamic event (SSE including other RBV loads).

3.9.3.4 Component Supports

The design of bolts for component supports is specified in the ASME Code Section III, Subsection NF. Stress limits for bolts are given in NF-3225. The rules and stress limits which must be satisfied are those given in NF-3324.6 multiplied by the appropriate stress limit factor for the particular service loading level and stress category specified in Table NF-3225.2-1.

Moreover, on equipment which is to be, or may be, mounted on a concrete support, sufficient holes for anchor bolts are provided to limit the anchor bolt stress to less than 10,000 psi on the nominal bolt area in shear or tension.

Concrete anchor bolts which are used for pipe support base plates will be designed to the applicable factors of safety which are defined in I&E Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," Revision 1 dated June 21,1979.

3.9.3.4.1 Piping

Supports and their attachments for essential ASME Code Section III, Class 1,2, and 3 piping are designed in accordance with Subsection NF* up to the interface of the building structure. The building structure component supports are designed in accordance with ANSI/AISC N690, Nuclear Facilities-Steel Safety-Related Structures for Design, Fabrication and Erection or AISC specification for the Design, Fabrication, and Erection of Structural Steel for buildings.

*Augmented by the following: (1) application of Code Case N-476, Supplement 89.1 which governs the design of single angle members of ASME Class 1,2,3 and MC linear component supports; and (2) when eccentric loads or other torsional loads are not accommodated by designing the load to act through the shear center or meet "Standard for Steel Support Design", analyses will be performed in accordance with torsional analysis methods such as: "Torsional Analysis of Steel Members, USS Steel Manual", Publication T114-2/83.

correspond to those used for design of the supported pipe. The component loading combinations are discussed in Subsection 3.9.3.1. The stress limits are per ASME III, Subsection NF and Appendix F. Supports are generally designed either by load rating method per paragraph NF-3260 or by the stress limits for linear supports per paragraph NF-3231. The critical buckling loads for the Class 1 piping supports subjected to faulted loads that are more severe than normal, upset and emergency loads, are determined by using the methods discussed in Appendices F and XVII of the Code. To avoid buckling in the piping supports, the allowable loads are limited to two thirds of the determined critical buckling loads.

The design of all supports for non-nuclear piping satisfies the requirements of ANSI B31.1, Paragraphs 120 and 121.

For the major active valves identified in Subsection 3.9.3.2.4, the valve operators are not used as attachment points for piping supports.

The design criteria and dynamic testing requirements for the ASME III piping supports are as follows:

(1) Piping Supports - All piping supports are designed, fabricated, and assembled so that they cannot become disengaged by the movement of the supported pipe of equipment after they have been installed. All piping supports are designed in accordance with the rules of Subsection NF of the ASME Code up to the building structure interface as defined in the project design specifications.

(2) Spring Hangers - The operating load on spring hangers is the load caused by dead weight. The hangers are calibrated to ensure that they support the operating load at both their hot and cold load settings. Spring hangers provide a specified down travel and up travel in excess of the specified thermal movement.

(3) Snubbers - The operating loads on snubbers are the loads caused by dynamic events (e.g., seismic, RBV due to LOCA and SRV discharge, discharge through a relief valve line or valve closure) during various operating conditions. Snubbers restrain piping against response to the vibratory excitation and to the associated differential movement of the piping system support anchor points. The criteria for locating snubbers and ensuring adequate load capacity, the structural and mechanical performance parameters used for snubbers and the installation and inspection considerations for the snubbers are as follows:

Required Load Capacity and Snubber Location

The entire piping system including valves and support system between anchor points is mathematically modeled for complete piping structural analysis. In the dynamic analysis, the snubbers are modeled as a spring with a given spring stiffness depending on the snubber size. The analysis determines the forces and moments acting on each piping components and the forces acting on the snubbers due to all dynamic loading and operating conditions defined in the piping design specification. The forces on snubbers are operating loads for various operating conditions. The calculated loads cannot exceed the snubber design load capacity for various operating conditions, i.e., design, normal, upset, emergency and faulted.

Snubbers are generally used in situations where dynamic support is required because thermal growth of the piping prohibits the use of rigid supports. The snubber locations and support directions are first decided by estimation so that the stresses in the piping system will have acceptable values. The snubber locations and support directions are refined by performing the dynamic analysis of the piping and support system as described above in order that the piping stresses and support loads meet the Code requirements.

The pipe support design specification requires that snubbers be provided with position indicators to identify the rod position. This indicator facilitates the checking of hot and cold settings of the snubber, as specified in the installation manual, during plant preoperational and startup testing.

(b) Inspection, Testing, Repair and/or Replacement of Snubbers

The pipe support design specification requires that the snubber supplier prepare an installation instruction manual. This manual is required to contain complete instructions for the testing, maintenance, and repair of the snubber. It also contains inspection points and the period of inspection.

The pipe support design specification requires that hydraulic snubbers be equipped with a fluid level indicator so that the level of fluid in the snubber can be ascertained easily.

The spring constant achieved by the snubber supplier for a given load capacity snubber is compared against the spring constant used in the piping system model. If the spring constants are the same, then the snubber location and support direction become confirmed. If the spring constants are not in agreement, they are brought in agreement, and the system analysis is redone to confirm the snubber loads. This iteration is continued until all snubber load capacities and spring constants are reconciled.

(c) Snubber Design and Testing

To assure that the required structural and mechanical performance characteristics and product quality are achieved, the following requirements for design and testing are imposed by the design specification:

(i) The snubbers are required by the pipe support design specification to be designed in accordance with all of the rules and regulations of the ASME Code Section III, Subsection NF. This design requirement includes analysis for the normal, upset, emergency, and faulted loads. These calculated loads are then compared against the allowable loads to make sure that the stresses are below the code allowable limit.

(ii) The snubbers are tested to insure that they can perform as required during the seismic and other RBV events, and under anticipated operational transient loads or other mechanical loads associated with the design requirements for the plant. The following test requirements are included:

o Snubbers are subjected to force or displacement versus time loading at frequencies within the range of significant modes of the piping system;

- Displacements are measured to determine the performance characteristics specified;

- Tests are conducted at various temperatures to ensure operability over the specified range;

- Peak test loads in both tension and compression are required to be equal to or higher than the rated load requirements; and

- The snubbers are tested for various abnormal environmental conditions. Upon completion of the abnormal environmental transient test, the snubber is tested dynamically at a frequency within a specified frequency range. The snubber must operate normally during the dynamic test.

(d) Snubber Installation Requirements

An installation instruction manual is required by the pipe support design specification. This manual is required to contain instructions for storage, handling, erection, and adjustments (if necessary) of snubbers. Each snubber has an installation location drawing which contains the installation location of the snubber on the pipe and structure, the hot and cold settings, and additional information needed to install the particular snubber.

(e) Snubber Pre-service Examination

The pre-service examination plan of all snubbers covered by the Chapter 16 technical specifications will be prepared. This examination will be made after snubber installation but not more than 6 months prior to initial system pre-operational testing. The pre-service examination will verify the following:

(i) There are no visible signs of damage or impaired operability as a result of storage, handling, or installation.

(ii) The snubber location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifications.

(iii) Snubbers are not seized, frozen or jammed.

(iv) Adequate swing clearance is provided to allow snubber movements.

(v) If applicable, fluid is to be recommended level and not be leaking from the snubber system.

(vi) Structural connections such as pins, fasteners and other connecting hardware such as lock nuts, tabs, wire, cotter pins are installed correctly.

If the period between the initial pre-service examination and initial system pre-operational tests exceeds 6 months because of unexpected situations, reexamination of Items 1, 4, and 5 will be performed. Snubbers which are installed incorrectly or otherwise fail to meet the above requirements will be repaired or replaced and re-examined in accordance with the above criteria.

(4) Struts - The design load on struts includes those loads caused by dead weight, thermal expansion, seismic forces (i.e., OBE and SSE), other RBV loads,

system anchor displacements, and reaction forces caused by relief valve discharge or valve closure, etc.

Struts are designed in accordance with ASME Code Section III, Subsection NF-3000 to be capable of carrying the design loads for various operating conditions. As in case of snubbers, the forces on struts are obtained from an analysis, which are assured not to exceed the design loads for various operating conditions.

3.9.3.4.2 Reactor Pressure Vessel Support Skirt

The ABWR RPV support skirt is designed as an ASME Code Class 1 component per the requirements of ASME Code Section III, Subsection NF*. The loading conditions and stress criteria are given in Tables 3.9-1 and 3.9-2, and the calculated stresses meet the Code allowable stresses in the critical support areas for various plant operating conditions. The stress level margins assure the adequacy of the RPV support skirt. An analysis for buckling shows that the support skirt complies with Subparagraph F-1332.5 of ASME III, Appendix F, and the loads do not exceed two thirds of the critical buckling strength of the skirt. The permissible skirt loads at any elevation, when simultaneously applied, are limited by the following interaction equation:

$$(P/P_{crit}) + (q/q_{crit}) + (\tau/\tau_{crit}) < (1/S.F.)$$

where:

q = longitudinal load
P = external pressure
τ = transverse shear stress
S.F. = safety factor
= 3.0 for design, testing, service levels A & B
= 2.0 for Service Level C
= 1.5 for Service Level D.

3.9.3.4.3 Reactor Pressure Vessel Stabilizer

The RPV stabilizer is designed as a Safety Class 1 linear type component support in accordance with the requirements of ASME Boiler and Pressure Vessel Code Section III, Subsection NF. The stabilizer provides a reaction point near the upper end of the RPV to resist horizontal loads due to effects such as earthquake, pipe rupture and RBV. The design loading conditions, and stress criteria are given in Tables 3.9-1 and 3.9-2, and the calculated stresses meet the Code allowable stresses in the critical support areas for various plant operating conditions.

3.9.3.4.4 Floor-Mounted Major Equipment (Pumps, Heat Exchangers, and RCIC Turbine)

Since the major active valves are supported by piping and not tied to building structures, valve "supports" do not exist (See Subsection 3.9.3.4.1).

The HPCF, RHR, RCIC, SLC, FPCCU, SPCU, and CUW pumps; RMC, RHR, RWCU, and FPCCU heat exchangers; and RCIC turbine are all analyzed to verify the adequacy of their support structure under various plant operating conditions. In all cases, the load stresses in the critical support areas are within ASME Code allowables.

Seismic Category I active pump supports are qualified for dynamic (seismic and other RBV) loads by testing when the pump supports

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together with the pump meet the following test conditions:

(1) simulate actual mounting conditions;

(2) simulate all static and dynamic loadings on the pump;

(3) monitor pump operability during testing;

(4) the normal operation of the pump during and after the test indicates that the supports are adequate (any deflection or deformation of the pump supports which precludes the operability of the pump is not accepted); and

(5) supports are inspected for structural integrity after the test. Any cracking or permanent deformation is not accepted.

Dynamic qualification of component supports by analysis is generally accomplished as follows:

(1) Stresses at all support elements and parts such as pump holddown and baseplate holddown bolts, pump support pads, pump pedestal, and foundation are checked to be within the allowable limits as specified in the ASME Code Section III, Subsection NF.

(2) For normal and upset conditions, the deflections and deformations of the supports are assured to be within the elastic limits, and to not exceed the values permitted by the designer based on design verification tests. This ensures the operability of the pump.

(3) For emergency and faulted plant conditions, the deformations do not exceed the values permitted by the designer to ensure the operability of the pump. Elastic/plastic analysis are performed if the deflections are above the elastic limits. 210.40

3.9.3.5 Other ASME III Component Supports

The ASME III component supports and their attachments (other than those discussed in preceding subsection) are designed in accordance with Subsection NF of the ASME Code Section III* up to
210.39 the interface with the building structure. The building structure component supports are designed in accordance with the AISC Specification for [illegible] Design, Fabrication, and Erection of Structural Steel for Buildings. The loading combinations for the various operating conditions correspond to those used to design the supported component. The component loading combinations are discussed in Subsection 3.9.3.1. Active component supports are discussed in Subsection 3.9.3.2. The stress limits are per ASME III, Subsection NF and Appendix F. The supports are evaluated for buckling in accordance with ASME III.

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*Augmented by the following: (1) application of Code Case N-476, Supplement 89.1 which governs the design of single angle members of ASME Class 1,2,3 and MC linear component supports; and (2) when eccentric loads or other torsional loads are not accommodated by designing the load to act through the shear center or meet "Standard for Steel Support Design", analyses will be performed in accordance with torsional analysis methods such as: "Torsional Analysis of Steel Members, USS Steel Manual", Publication T114-2/83.

3.9.4 Control Rod Drive System (CRDS)

[illegible]ontrol rod drive system CRDS)in an ABWR [illegible] is equipped with an electro-hydraulic fine motion control rod drive (FMCRD) system, which includes the control rod drive (CRD) mechanism, the hydraulic control unit (HCU), the condensate supply system, and power for FMCRD motor, and extends inside RPV to the coupling interface with the control rod blades.

3.9.4.1 Descriptive Information on CRDS

Descriptive inform[illegible] on the CRDs as well as the entire control [illegible] drive system is contained in Section 4.6.

3.9.4.2 Applicable CRDS Design Specification

CRDS is designed to meet the functional design criteria outlined in Section 4.6 and consists of the following:

(1) fine motion control rod drive;

(2) hydraulic control unit;

(3) hydraulic power supply (pumps);

(4) electric power supply (for FMCRD motors)

(5) interconnecting piping;

(6) flow and pressure and isolation valves; and

(7) instrumentation and electrical controls.

Those components of the CRDS forming part of the primary pressure boundary are designed according to ASME Code Section III, Class 1 requirements.

The quality group classification of the components of the CRDS is outlined in Table 3.2-1 and they are designed to the codes and standards, per Table 3.2-2, in accordance with their individual quality groups.

Pertinent aspects of the design and qualification of the CRDS components are discussed in the following locations: transients in Subsection 3.9.1.1, faulted conditions in Subsection 3.9.1.4, seismic testing in Subsection 3.9.2.2.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations

The ASME III Code components of the CRDS have been evaluated analytically and the design loading conditions, and stress criteria are as given in Tables 3.9-1 and 3.9-2, and the calculated stresses meet the Code allowable stresses. For the non-Code components, the ASME III Code requirements are used as guidelines and experimental testing is used to determine the CRD performance under all possible conditions as described in Subsection 3.9.4.4.

3.9.4.4 CRD Performance Assurance Program

The CRD test program consists of these tests:

(1) development tests;

(2) factory quality control tests;

(3) Five-year maintenance life tests;

(4) 1.5X design life tests;

(5) operational tests;

(6) acceptance tests; and

(7) surveillance tests.

All of the tests except (3) and (4) are discussed in Section 4.6. A discussion of tests (3) and (4) follows:

(3) Five-Year Maintenance Life Tests - Four control rod drives are normally picked at random from the production stock each year and subjected to various tests under simulated reactor conditions and 1/6th of the service life cycles.

Upon completion of the test program, control rod drives must meet or surpass the minimum specified performance requirements.

(4) 1.5X Design Life Tests - When a significant design change is made to the components of the drive, the drive is subjected to a series of tests equivalent to 1.5 times the service life cycles.

3.9.5 Reactor Pressure Vessel Internals

This subsection identifies and discusses the structural and functional integrity of the major reactor pressure vessel (RPV) internals, including core support structures.

3.9.5.1 Design Arrangements

The core support structures and reactor vessel internals (exclusive of fuel, control rods, and incore nuclear instrumentation) are:

(1) Core Support Structures

Shroud;

Shroud support (including the internal pump deck);

Core plate (and core plate hardware);

Top guide;

Fuel supports (orificed fuel supports and peripheral fuel supports);

Control rod guide tubes; and

(2) Reactor Internals

*Shroud head and *steam separators assembly;

*Steam dryers assembly;

Feedwater spargers;

RHR/ECCS low pressure flooder spargers;

ECCS high pressure core flooder spargers and piping;

RPV vent and head spray assembly;

Core and *internal pump differential pressure lines;

In-core guide tubes and stabilizers;

*Surveillance sample holders;

A general assembly drawing of the important

* *These are non-nuclear safety (or "other") category components as defined in Subsection 3.2.5.1. In Subsection 3.9.5, such components are called non-safety class components, and the safet-related internals (Safety Class 3) are called safety class components.*

reactor components is shown in Figure 5.3-2.

The floodable inner volume of the reactor pressure vessel can be seen in Figure 3.9-2. It is the volume up to the level of the core flooder sparger.

The design arrangement of the reactor internals, such as the shroud, steam separators and guide tubes, is such that one end is unrestricted and thus free to expand.

The ECCS core flooder couplings incorporate vertically-oriented slip-fit joints to allow free thermal expansion.

3.9.5.1.1 Core Support Structures

The core support structures consist of those items listed in Subsection 3.9.5.1(1) and are Safety Class 3 as defined in Section 3.2. These structures form partitions within the reactor vessel to sustain pressure differentials across the partitions, direct the flow of the coolant water, and laterally locate and support the fuel assemblies. Figures 3.9-2 and 3.9-3 show the reactor vessel internal flow paths.

3.9.5.1.1.1 Shroud

The shroud support, shroud, and top guide make up a stainless steel cylindrical assembly that provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. This partition separates the core region from the downcomer annulus. The volume enclosed by this assembly is characterized by three regions. The upper portion surrounds the core discharge plenum which is bounded by the shroud head on top and the top guide plate below. The central portion of the shroud surrounds the active fuel and forms the longest section of the assembly.

This section is bounded at the top by the top guide plate and at the bottom by the core plate. The lower portion, surrounding part of the lower plenum, is welded to the reactor pressure vessel shroud support. The shroud provides the horizontal support for the core by supporting the core plate and top guide.

3.9.5.1.1.2 Shroud Support

The RPV shroud support is designed to support the shroud, and includes the internal pump deck that locates and supports the pumps. The pump discharge diffusers penetrate the deck to introduce the coolant to the inlet plenum below the core. The RPV shroud support is a horizontal structure welded to the vessel wall to provide support to the shroud, pump diffusers, and core and pump deck differential pressure lines. The structure is a ring plate welded to the vessel wall and to a vertical cylinder supported by vertical stilt legs from the bottom head.

3.9.5.1.1.3 Core Plate

The core plate consists of a circular stainless steel plate with round openings and is stiffened with a rim and beam structure. The core plate provides lateral support and guidance for the control rod guide tubes, in-core flux monitor guide tubes, peripheral fuel supports, and startup neutron sources. The last two items are also supported vertically by the core plate.

The entire assembly is bolted to a support ledge in the lower portion of the shroud.

3.9.5.1.1.4 Top Guide

The top guide consists of a circular plate with square openings for fuel with a cylindrical side forming an upper shroud extension and having a top flange for attaching the shroud head. Each opening provides lateral support and guidance for four fuel assemblies or, in the case of peripheral fuel, less than four fuel assemblies. Holes are provided in the bottom of the support intersections to anchor the in-core flux monitors and startup neutron sources. The top guide is mechanically attached to the top of the shroud.

3.9.5.1.1.5 Fuel Supports

The fuel supports (Figure 3.9-4) are of two basic types: peripheral supports and orificed fuel supports. The peripheral fuel supports are located at the outer edge of the active core and are not adjacent to control rods. Each peripheral fuel support supports one fuel assembly and contains an orifice designed to assure proper coolant flow to the peripheral fuel assembly. Each orificed fuel support supports four fuel assemblies vertically upward and horizontally and is provided with orifice plates to assure proper coolant flow distribution to each rod-controlled fuel assembly. The orificed fuel supports rest on the top of the control rod guide tubes which are supported laterally by the core plate. The control rods pass through cruciform openings in the center of the orificed fuel support. A control rod and the four adjacent fuel assemblies represent a core cell (Section 4.4).

3.9.5.1.1.6 Control Rod Guide Tubes

The control rod guide tubes located inside the vessel extend from the top of the control rod drive housings up through holes in the core plate. Each guide tube is designed as the guide for the lower end of a control rod and as the support for an orificed fuel support. This locates the four fuel assemblies surrounding the control rod. The bottom of the guide tube is supported by the control rod drive housing, which in turn transmits the weight of the guide tube, fuel support, and fuel assemblies to the reactor vessel bottom head. The control rod guide tubes also contain holes, hear the top of the control rod guide tube and below the core plate, for coolant flow to the orificed fuel supports.

3.9.5.1.2 Reactor Internals

The reactor internals consist of those items listed in Subsection 3.9.5.1(2), and are Safety Class 3 or non-safety class as noted. These components direct and control coolant flow through the core or support safety-related and nonsafety related function.

3.9.5.1.2.1 Shroud Head and Steam Separators Assembly

The shroud head and standpipes/steam separators are non-safety class internal components. The assembly is discussed here to describe the coolant flow paths in the reactor pressure vessel. The shroud head and steam separators assembly includes the upper flanges and bolts, and forms the top of the core discharge mixture plenum together with the separators and their connecting standpipes. The discharge plenum provides a mixing chamber for the steam/water mixture before it enters the steam separators. Individual stainless steel axial flow steam separators are supported on and attached to the top of standpipes that are welded into the shroud head. The steam separators have no moving parts. In each separator, the steam/water mixture rising through the standpipe passes vanes that impart a spin to establish a vortex separating the water from the steam. The separated water flows from the lower portion of the steam separator into the downcomer annulus. The assembly is removable from the reactor pressure vessel as a single unit on a routine basis.

3.9.5.1.2.2 Reactor Internal Pump (RIP)/Diffusers

The pump assemblies are non-safety class components and are discussed here to describe coolant flow paths (Figure 3.9-3) in the vessel. The pump provides a means for forced circulation of the reactor coolant through the core, including the mixing of feedwater and annulus water from the steam separators and distribution of this fluid to the vessel lower plenum and up through the lower grid to the core.

The pump assemblies are mounted vertically into pump nozzles arranged in an equally-spaced ring pattern on the bottom head of the RPV and are located inside the downcomer annulus between the core shroud and the reactor vessel wall. The design and performance of the pump assemblies is covered in detail in Subsection 5.4.1. Each pump consists of three major hardware sections: an internal pump (IP) section; a recirculation motor (RM) section; and a stretch tube section (Figure 5.4-1).

The IP section of the RIP is located inside the RPV, in an opening through the RPV pump deck--the latter being the horizontal ring-plate enclosing the bottom of the downcomer annulus and thus separating the lower pressure annulus region from the higher-pressure lower plenum region. The IP, in turn, is comprised of a vertical axis single-stage, mixed-flow impeller driven from underneath by a pump shaft, with the impeller being encircled by a diffuser shroud assembled into the pump deck opening.

The RM section of the RIP is located underneath, and at the periphery of, the RPV bottom head inside a pressure retaining housing termed the motor casing. The motor casing itself is not part of the RM, but is instead a part of and welded into an RPV nozzle (pump nozzle). The motor casing thus comprises part of the reactor coolant pressure boundary and is a Safety Class 1 component.

The principal element of the stretch tube section is a thin-walled tube configured as a hollow bolt fitting around the pump shaft and within the pump nozzle. It has an external lip (bolt head) at its upper end and an external threaded section at this lower end. The stretch tube function is to achieve tight clamping of the IP diffuser to the gasketed, internal-mount end of the RPV pump nozzle, at all extremes of thermal transients and pump operating conditions.

3.9.5.1.2.3 Steam Dryer Assembly

The steam dryer assembly is a non-safety class component. It is discussed here to describe coolant flow paths in the vessel. The steam dryer removes moisture from the wet steam leaving the steam separators. The extracted moisture flows down the dryer vanes to the collecting troughs, then flows through tubes into the downcomer annulus.

The steam dryer assembly consists of multiple banks of dryer units mounted on a common structure which is removable from the reactor pressure vessel as an integral unit. The assembly includes the dryer banks, dryer supply and discharge ducting, drain collecting trough, drain piping, and a skirt which forms a water seal extending below the separator reference zero elevation. Upward and radial movement of the dryer assembly under the action of blowdown and seismic loads are limited by reactor vessel internal stops which are arranged to permit differential expansion growth of the dryer assembly with respect to the reactor pressure vessel. The assembly is arranged for removal from the vessel as an integral unit on a routine basis.

3.9.5.1.2.4 Feedwater Spargers

These are Safety Class 2 components. They are discussed here to describe coolant flow paths in the vessel and their safety function. Each of two feedwater lines is connected to three spargers via three RPV nozzles. One line is utilized by the RCIC system; the other by the RHR shutdown cooling system. During the ECCS mode, the two groups of spargers support diverse type of flooding of the vessel. The RCIC system side supports high pressure flooding and the RHR system side supports low pressure flooding, as required during the ECCS operation.

The feedwater spargers are stainless steel headers located in the mixing plenum above the downcomer annulus. A separate sparger in two halves is fitted to each feedwater nozzle via a tee and is shaped to conform to the curve of the vessel wall. The sparger tee inlet is connected to the RPV nozzle safe end by a double thermal sleeve arrangement, with all connections made by full penetration welds. Sparger end brackets are pinned to vessel brackets to support the spargers. Feedwater flow enters the center of the spargers and is discharged radially inward to mix the cooler feedwater with the downcomer flow from the steam separators and steam dryer before it contacts the vessel wall. The feed- water also serves to condense steam in the region above the downcomer annulus and to subcool water flowing to the recirculation internal pumps.

3.9.5.1.2.5 RHR/ECCS Low Pressure Flooder Spargers

These are Safety Class 2 components. The design features of these two spargers of the RHR shutdown cooling system are similar to those of the six feedwater spargers, three of which belonging to one feedwater line support additionally the same RHR (and ECCS) function. During the ECCS mode, these spargers support low pressure flooding of the vessel. The feedwater spargers are described in Subsection 3.9.5.1.2.4.

Two lines of RHR shutdown cooling system enter the reactor vessel through the two diagonally opposite nozzles and connect to the spargers. The sparger tee inlet is connected to the RPV nozzle safe end by a thermal sleeve arrangement with all connections made by full penetration welds.

3.9.5.1.2.6 ECCS High Pressure Core Flooder Spargers and Piping

21045 The core flooder spargers and piping are Safety Class 2. The spargers and piping are the means for directing high pressure ECCS flow to the upper end of the core during accident conditions.

Each of two high pressure core flooder (HPCF) system lines enters the reactor vessel through a diagonally opposite nozzle in the same manner as an RHR low pressure flooder line, except that the curved sparger including the connecting tee is routed around the inside of and is supported by the cylindrical portion of the top guide. A flexible coupling is interposed between the sparger tee inlet and the sleeved inlet connector inside the nozzle. The two spargers are supported so as to accommodate thermal expansion.

3.9.5.1.2.7 RPV Vent and Head Spray Assembly

This is designed as a Safety Class 1 component. However, only the nozzle portion of the assembly is a reactor coolant pressure boundary, and the assembly function is not a safety-related operation. The reactor water cleanup return flow to the reactor vessel, via feedwater lines, can be diverted partly to a spray nozzle in the reactor head in preparation for refueling cooldown. The spray maintains saturated conditions in the reactor vessel head volume by condensing stream being generated by the hot reactor vessel walls and internals. The head spray subsystem is designed to rapidly cooldown the reactor vessel head flange region for refueling and to allow installation of steam line plugs before vessel floodup for refueling.

The head vent side of the assembly passes steam and noncondensable gases from the reactor head to the steamlines during startup and operation. During shutdown and filling for hydrotesting, steam and noncondensable gases may be vented to the drywell equipment sump while the connection to the steamline is blocked. When draining the vessel during shutdown, air enters the vessel through the vent.

3.9.5.1.2.8 Core and Internal Pump Differential Pressure Lines

These lines comprise the core flow measurement subsystem of the recirculation flow control system (RFCS) and provide two methods of measuring the ABWR core flow rates. The core DP lines (Safety Class 3) and internal pump DP lines (non-safety class) enter the reactor vessel separately through reactor bottom head penetrations. Four pairs of the core DP lines enter the head in four quadrants through four penetrations and terminate immediately above and below the core plate to sense the pressure in the region outside the bottom of the fuel assemblies and below the core plate during normal operation.

Similarly, four pairs of the internal pump DP lines terminate above and below the pump deck and are used to sense the pressure across the pump during normal pump operation. Each pair is routed concentrically through a penetration and upward along a shroud support leg in the lower plenum.

3.9.5.1.2.9 In-Core Guide Tubes and Stabilizers

These are Safety Class 3 components. The guide tubes protect the in-core instrumentation from flow of water in the bottom head plenum and provide a means of positioning fixed detectors in the core as well as a path for insertion and withdrawal of the calibration monitors (ATIP, automated traversing incore probe subsystem). The in-core flux monitor guide tubes extend from the top of the in-core flux monitor housing to the top of the core plate. The power range detectors for the power range monitoring units and the dry tubes for the startup range neutron monitoring and average power range monitoring (SRNM/APRM) detectors are inserted through the guide tubes.

Two levels of stainless steel stabilizer latticework of clamps, tie bars, and spacers give lateral support and rigidity to the guide tubes. The stabilizers are connected to the shroud and shroud support. The bolts are tack-welded after assembly to prevent loosening during reactor operation.

3.9.5.1.2.10 Surveillance Sample Holders

This a non-safety class component. The surveillance sample holders are welded baskets containing impact and tensile specimen capsules. The baskets hang from the brackets that are attached to the inside of the reactor vessel wall and extend to mid height of the active core. The radial positions are chosen to expose the specimens to the same environment and maximum neutron fluxes experienced by the reactor vessel itself.

3.9.5.2 Loading Conditions

3.9.5.2.1 Events to be Evaluated

Examination of the spectrum of conditions for which the safety design bases (Subsection 3.9.5.3.1) must be satisfied by core support structures and safety-related internal components reveals four significant faulted events:

(1) Feedwater Line Break - A break in a feedwater line between the reactor vessel and the primary containment penetration; (the accident results in significant annulus pressurization and reactor building vibration due to suppression pool dynamics);

(2) Steam Line Break Accident - A break in one main steam line between the reactor vessel nozzle and the main steam isolation valve; (the accident results in significant pressure differentials across some of the structures within the reactor and reactor building vibration due to suppression pool dynamics);

(3) Earthquake - subjects the core support structures and reactor internals to significant forces as a results of ground motion and consequent RBV; and

(4) Safety/relief valve discharge - RBV due to suppression pool dynamics and structural feedback

Analysis of other conditions existing during normal operation, abnormal operational transients, and accidents show that the loads affecting core support structures and other safety-related reactor internals are less severe than those affected by the four postulated events.

The faulted conditions for the reactor pressure vessel internals are discussed in Subsection 3.9.1.4. Loading combination and analysis for safety-related reactor internals including core support structures are discussed in Subsections 3.9.3.1, 3.9.5.3.5, and 3.9.5.3.6.

3.9.5.2.2 Pressure Differential During Rapid Depressurization

A digital computer code is used to analyze the transient conditions within the reactor vessel following the main steam line break between the vessel nozzle and main steam isolation valve. The analytical model of the vessel consists of nine nodes which are connected to the necessary adjoining nodes by flow paths having the required resistance and inertial characteristics. The program solves the energy and mass conservation equations for each node to give the depressurization rates and pressures in the various regions of the reactor. Figure 3.9-5 shows the nine reactor nodes. The computer code used is the General Electric Short-Term Thermal-Hydraulic Model described in Reference 4. This model has been approved for use in ECCS conformance evaluation under 10CFR50 Appendix K. In order to adequately describe the blowdown pressure effect on the individual assembly components, three features are included in the model that are not applicable to the ECCS analysis and are therefore not described in Reference 4. These additional features are as follows:

(1) The liquid level in the steam separator region and in the annulus between the dryer skirt and the pressure vessel is tracked to more accurately determine the flow and mixture quality in the steam dryer and in the steamline.

(2) The flow path between the bypass region and the shroud head is more accurately modeled since the fuel assembly pressure differential is influenced by flashing in the

guide tubes and bypass region for a steam-line break. In the ECCS analysis, the momentum equation is solved in this flow path but its irreversible loss coefficient is conservatively set at an arbitrary low value.

(3) The enthalpies in the guide tubes and the bypass are calculated separately since the fuel assembly pressure differential is influenced by flashing in these regions. In the ECCS analysis, these regions are lumped.

3.9.5.2.3 Feedwater Line and Main Steam Line Break

3.9.5.2.3.1 Accident Definition

Both a feedwater line break (the largest liquid line break) and a main steam line break (the largest steam line break) upstream of the main steam isolation valve are considered in determining the design basis accident for the safety related reactor internals including the core support structures.

The feedwater line break is the same as the design basis loss-of-coolant accident described in Subsection 6.2.1.1.3.3.1. A sudden, complete circumferential break is assumed to occur in one feedwater line. The pressure differentials on the reactor internals and core support structures are in all cases lower than those for the main steam line break.

The analysis for the main steam line break assumes a sudden, complete circumferential break of one main steam line at the reactor vessel nozzle, downstream of the limiting flow area. This is described in Subsection 6.2.1.1.3.3.2.

The steam line break accident produces significantly higher pressure differential across the reactor internal structures than does the feedwater line break. This results from the higher reactor depressurization rate associated with the steam line break. Therefore, the steam line break is the design basis accident for internal pressure differentials.

3.9.5.2.3.2 Effects of Initial Reactor Power and Core Flow

The maximum internal pressure loads can be considered to be composed of two parts: steady-state and transients pressure differentials. For a given plant, the core flow and power are the two major factors which influence the reactor internal pressure differentials. The core flow essentially affects only the steady-state part. For a fixed power, the greater the core flow, the larger will be the steady-state pressure differentials. On the other hand, the core power affects both the steady-state and the transient parts. As the power is decreased, there is less voiding in the core and consequently the steady-state core pressure differential is less. However, less voiding in the core also means that less steam is generated in the reactor pressure vessel and thus the depressurization rate and the transient part of the maximum pressure load is increased. As a result, the total loads on some components are higher at low power.

To ensure that calculated pressure differences bound those which could be expected if a steam line break should occur, an analysis is conducted at a low power high-recirculation flow condition in addition to the standard safety analysis condition at high power, rated recirculation flow. The power chosen for analysis is the minimum value permitted by the recirculation system controls at rated recirculation drive flow (i.e., the drive flow necessary to achieve rated core flow at rated power).

This condition maximizes those loads which are inversely proportional to power. It must be noted that this condition, while possible, is unlikely; first, because the reactor will generally operate at or near full power; second, because high core flow is neither required nor desirable at such a reduced power condition.

Table 3.9-3 summarizes the maximum pressure differentials. Case 1 is the safety analysis condition; Case 2 is the low power high-flow condition.

3.9.5.2.4 Seismic and Other Reactor Building Vibration Events

The loads due to earthquake and other reactor building vibration (RBV) acting on the structure within the reactor vessel are based on a dynamic analysis described in Sections 3.7, 3.8, and

Subsection 3.9.2.5. Dynamic analysis is performed by coupling the lumped-mass model of the reactor vessel and internals with the building model to determine the system natural frequencies and node shapes. The relative displacement, acceleration, and load response is then determined by either the time-history method or the response-spectrum method.

3.9.5.3 Design Bases

3.9.5.3.1 Safety Design Bases

The reactor internals including core support structures shall meet the following safety design bases:

(1) The reactor vessel nozzles and internals shall be so arranged as to provide a floodable volume in which the core can be adequately cooled in the event of a breach in the nuclear system process barrier external to the reactor vessel;

(2) Deformation of internals shall be limited to assure that the control rods and core standby cooling systems can perform their safety-related functions; and

(3) Mechanical design of applicable structures shall assure that safety design bases (1) and (2) are satisfied so that the safe shutdown of the plant and removal of decay heat are not impaired.

3.9.5.3.2 Power Generation Design Bases

The reactor internals including core support structures shall be designed to the following power generation design bases:

(1) The internals shall provide the proper coolant distribution during all anticipated normal operating conditions to full power operation of the core without fuel damage;

(2) The internals shall be arranged to facilitate refueling operations; and

(3) The internals shall be designed to facilitate inspection.

3.9.5.3.3 Design Loading Categories

The basis for determining faulted dynamic event loads on the reactor internals is shown in Sections 3.7, 3.8 and Subsections 3.9.2.5, 3.9.5.2.3 and 3.9.5.2.4. Table 3.9-2 shows the load combinations used in the analysis.

Core support structures and safety class internals stress limits are consistent with ASME Code Section III, Subsection NG. For these components, Level A, B, C, and D service limits are applied to the normal, upset, emergency, and faulted loading conditions, respectively, as defined in the design specification. Stress intensity and other design limits are discussed in Subsections 3.9.5.3.5 and 3.9.5.3.6

3.9.5.3.4 Response of Internals Due to Steam Line Break Accident

As described in Subsection 3.9.5.2.3.2, the maximum pressure loads acting on the reactor internal components result from steam line break upstream of the main steam isolation valve and, on some components, the loads are greatest with operation at the minimum power associated with the maximum core flow (Table 3.9-3, Case 2). This has been substantiated by the analytical comparison of liquid versus steam line breaks and by the investigation of the effects of core power and core flow.

It has also been pointed out that, although possible, it is not probable that the reactor would be operating at the rather abnormal condition of minimum power and maximum core flow. More realistically, the reactor would be at or near a full power condition and thus the maximum pressure loads acting on the internal components would be as listed under Case 1 in Table 3.9-3.

3.9.5.3.5 Stress and Fatigue Limits for Core Support Structures

The design and construction of the core support structures are in accordance with ASME Code Section III, Subsection NG.

3.9.5.3.6 Stress, Deformation, and Fatigue Limits for Safety Class and Other Reactor Internals (Except Core Support Structures)

For safety class reactor internals, the stress deformation and fatigue criteria listed in Tables 3.9-4 through 3.9-7 are based on the criteria established in applicable codes and standards for similar equipment, by manufacturers standards, or by empirical methods based on field experience and testing. For the quantity SF_{min} (minimum safety factor) appearing in those tables, the following values are used:

Service Level	Service Condition	SF_{min}
A	Normal	2.25
B	Upset	2.25
C	Emergency	1.5
D	Faulted	1.125

Components inside the reactor pressure vessel such as control rods which must move during accident condition have been examined to determine if adequate clearances exist during emergency and faulted conditions. No mechanical clearance problems have been identified. The forcing functions applicable to the reactor internals are discussed in Subsection 3.9.2.5.

The design criteria, loading conditions, and analyses that provide the basis for the design of the safety class reactor internals other than the core support structures meet the guidelines of NG-3000 and are constructed so as not to adversely affect the integrity of the core support structures (NG-1122).

The design requirements for equipment classified as non-safety (other) class internals (e.g., steam dryers and shroud heads) are specified with appropriate consideration of the intended service of the equipment and expected plant and environmental conditions under which it will operate. Where Code design requirements are not applicable, accepted industry or engineering practices are used.

3.9.6 Inservice Testing of Pumps and Valves

Inservice testing of safety-related pumps and valves will be performed in accordance with the requirements of Section XI, Subsection IWP and IWV, of the ASME Code. Table 3.9-8 lists the inservice testing parameters and frequencies for the safety-related pumps and valves. Valves having a containment isolation function are also noted in the listing. Code testing flexibility in the ASME/ANSI O&M Part 6 for pumps and Part 10 for valves produced no need for relief requests. A review of field experience for typical BWR testing problems also showed the Code encompassed common relief requests. Inservice inspection is discussed in Subsection 5.2.4 and Section 6.6. 210.47

Details of the inservice testing program, including test schedules and frequencies will be reported in the inservice inspection and testing plan which will be provided by the applicant referencing the ABWR design. The plan will integrate the applicable test requirements for safety-related pumps and valves including those listed in the technical specifications, Chapter 16, and the containment isolation valves, Subsection 6.2.4. An example is the periodic leak testing of the reactor coolant pressure isolation valves in Table 3.9-9 will be performed in accordance with Chapter 16 Surveillance Requirement 3.6.1.5.10. This plan will include baseline pre-service testing to support the periodic in-service testing of the components. Depending on the test results, the plan will provide a commitment to disassemble and inspect the safety related pumps and valves when limits of Subsection IWP or IWV are exceeded, as described in the following paragraphs. The primary elements of this plan, including the requirements of Generic Letter 89-10 for motor operated valves, are delineated in the subsections to follow. (See Subsection 3.9.7.3 for COL license information requirements). 210.49 210.50

3.9.6.1 Inservice Testing of Safety-Related Pumps

The ABWR safety-related pumps and piping configurations accommodate inservice testing at a flow rate at least as large as the maximum design flow for the pump. In addition, the 210.47

sizing of each minimum recirculation flow path is evaluated to assure that its use under all analyzed conditions will not result in degradation of the pump. The flow rate through minimum recirculation flow paths can also be periodically measured to verify that flow is in accordance with the design specification.

The safety-related pumps are provided with instrumentation to verify that the net positive suction head (NPSH) is greater than or equal to the NPSH required during all modes of pump operation. These pumps can be disassembled for evaluation when the Code Section XI testing results in a deviation which falls within the "required action range." The Code provides criteria limits for the test parameters indentified in Table 3.9-8. A program will be developed by the applicant referencing the ABWR design to establish the frequency and the extent of disassembly and inspection based on suspected degradation of all safety related pumps, including the basis for the frequency and the extent of each disassembly. The program may be revised throughout the plant life to minimize disassembly based on past disassembly experience. (See Subsection 3.9.7.3(1) for COL license information requirements.)

3.9.6.2 Inservice Testing of Safety-Related Valves

3.9.6.2.1 Check Valves

All ABWR safety-related piping systems incorporate provisions for testing to demonstrate the operability of the check valves under design conditions. In-service testing will incorporate the use of advance non-intrusive techniques to periodically assess degradation and the performance characteristics of the check valves. The Code Section XI tests will be performed, and check valves that fail to exhibit the required performance can be disassembled for evaluation. The Code provides criteria limits for the test parameters identified in Table 3.9-8. A program will be developed by the applicant referencing the ABWR design to establish the frequency and the extent of disassembly and inspection based on suspected degradation of all safety related pumps, including the basis for the frequency and the extent of each disassembly. The program may be revised throughout the plant life to minimize disassembly based on past disassembly experience. (See Subsection 3.9.7.3(1) for COL license information requirements.)

3.9.6.2.2 Motor Operated Valves

The motor operated valve (MOV) equipment specifications require the incorporation of the results of either in-situ or prototype testing with full flow and pressure or full differential pressure to verify the proper sizing and correct switch settings of the valves. Guidelines to justify prototype testing are contained in Generic Letter 98-10, Supplement 1, Questions 22 and 24 through 28. The applicant referencing the ABWR design will provide a study to determine the optimal frequency for valve stroking during in-service testing such that unnecessary testing and damage is not done to the valve as a result of the testing. (See Subsection 3.9.7.3 for COL license information requirements).

The concerns and issues identified in Generic Letter 89-10 for MOVs will be addressed prior to plant startup. The method of assessing the loads, the method of sizing the actuators, and the setting of the torque and limit switches will be specifically addressed. (See Subsection 3.9.7.3 for COL license information requirements).

The in-service testing of MOVs will rely on diagnostic tecniques that are consistent with the state of the art and which will permit an assessment of the performance of the valve under actual loading. Periodic testing will be conducted under adequate differential pressure and flow conditions that allow a justifiable demonstration of continuing MOV capability for design basis conditions, including recovery from inadvertent valve positioning. MOVs that fail the acceptance criteria, and are "declared inoperable," for stroke tests and leakage rate can be disassembled for evaluation. The Code provides criteria limits for the test parameters identified in Table 3.9-8. A program will be developed by the apllicant referencing the ABWR design to establish the frequency and the extent of disassembly and inspection based on suspected degradation of all safety related "MOV's", including the basis for the frequency and the extent of each disassembly. The program may be revised throughout the plant life to minimize disassembly based on past disassembly experience. (See Subsection 3.9.7.3(1) for COL license information requirements.)

3.9.6.2.3 Isolation Valve Leak Tests

The leak-tight integrity will be verified for each valve relied upon to provide a leak-tight function. These valves include:

(1) pressure isolation valves - valves that provide isolation of pressure differential from one part of a system from another or between systems;

(2) temperature isolation valves - valves whose leakage may cause unacceptable thermal loading on supports or stratification in the piping and thermal loading on supports or whose leakage may cause steam binding of pumps; and

(3) containment isolation valves - valves that perform a containment isolation function in accordance with the Evaluation Against Criterion 54, Subsection 3.1.2.5.5.2, including valves that may be exempted from Appendix J, Type C, testing but whose leakage may cause loss of suppression pool water inventory.

Leakage rate testing of valves will be in accordance with the Code Section XI. An example is the fusible plug valves that provide a lower drywell flood for severe accidents described in Subsection 9.5.12. The valves are safety-related due to the function of retaining suppression pool water as shown in Figure 9.5-3. These special valves are noted here and not in Table 3.9-8. The fusible plug valve is a nonreclosing pressure relief device and the Code requires replacement of each at a maximum of 5 year intervals.

3.9.7 COL License Information

3.9.7.1 Reactor Internals Vibration Analysis, Measurement and Inspection Program

The first COL applicant referencing the ABWR design will provide, at the time of application, the results of the vibration assessment program for the ABWR prototype internals. These results will include the following information specified in Regulatory Guide 1.20.

R. G. 1.20	Subject
C.2.1	Vibration Analysis Program
C.2.2	Vibration Measurement Program
C.2.3	Inspection Program
C.2.4	Documentation of Results

NRC review and approval of the above information on the first applicants docket will complete the vibration assessment program requirements for prototype reactor internals.

In addition to the information tabulated above, the first applicant referencing the ABWR design will provide the information on the schedules in accordance with the applicable portions of position C.3 of Regulatory Guide 1.20 for non-prototype internals.

Subsequent COL applicants need only provide the information on the schedules in accordance with the applicable portions of position C.3 of Regulatory Guide 1.20 for non-prototype internals. (See Subsection 3.9.2.4 for interface requirements).

3.9.7.2 ASME Class 2 or 3 or Quality Group Components with 60 Year Design Life

COL applicants referencing the ABWR design will identify ASME Class 2 or 3 or Quality Group D components that are subjected to loadings which could result in thermal or dynamic fatigue and provide the analyses required by the ASME Code, Subsection NB. These analyses will include the appropriate operating vibration loads and for the effects of mixing hot and cold fluids. (See Subsection 3.9.3.1.

3.9.7.3 Pump and Valve Inservice Testing Program

COL applicants referncing the ABWR design will provide a plan for the detailed pump and valve inservice testing and inspection program. This plan will

(1) Include baseline pre-service testing to support the periodic in-service testing of the components required by technical specifications. Provisions are included to disassemble and inspect the pump, check valves, and MOVs within the Code and safety-related classification as necessary, depending on test results. (See Subsections 3.9.6, 3.9.6.1, 3.9.6.2.1 and 3.9.6.2.2)

(2) Provide a study to determine the optimal frequency for valve stroking during inservice testing. (See Subsection 3.9.6.2.2)

(3) Address the concerns and issues identified in Generic Letter 89-10; specifically the method of assessment of the loads, the method of sizing the actuators, and the setting of the torque and limit switches. (See Subsection 3.9.6.2.2)

3.9.7.4 Audit of Design Specification and Design Reports

COL applicants referencing the ABWR design will make available to the NRC staff design specification and design reports required by ASME Code for vessels, pumps, valves and piping systems for the purpose of audit. (See Subsection 3.9.3.1)

3.9.8 References

1. *BWR Fuel Channel Mechanical Design and Deflection,* NEDE-21354-P, September 1976.

2. *BWR/6 Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of Coolant Accident (LOCA) Loadings,* NEDE-21175-P, November 1976.

3. NEDE-24057-P (Class III) and NEDE-24057 (Class I) Assessment of Reactor Internals. Vibration in BWR/4 and BWR/5 Plants,

November 1977. Also NEDO-24057-P, Amendment 1, December 1978, and NEDE-2-P 24057 Amendment 2, June 1979.

4. *General Electric Company, Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K*, NEDE-20566P, Proprietary Document, November 1975.

5. *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*, NUREG-0619.

6. *General Electric Environmental Qualification Program*, NEDE-24326-1-P, Proprietary Document, January 1983.

7. *Functional Capability Criteria for Essential Mark II Piping*, NEDO-21985, September 1978, prepared by Battelle Columbus Laboratories for General Electric Company.

8. *Generic Criteria for High Frequency Cutoff of BWR Equipment*, NEDO-25250, Proprietary Document, January 1980.

Table 3.9-1

PLANT EVENTS

A. Plant Operating Events

		ASME Code Service Limit(10)	No. of Events(1)
1.	Boltup (1)	A	68
2.	Hydrostatic Test (two test cycles for each boltup cycle)	Testing	135
3.	Startup ($100^{\circ}F/hr$ Heatup Rate)(2)	A	390
4.	Daily and Weekly Reduction to 50% Power (1)	A	18,000
5.	Control Rod Pattern Change (1)	A	600
6.	Loss of Feedwater Heaters	B	120
7.	Scram:		
	a. Turbine Generator Trip, Feedwater On, and Other Scrams	B	188
	b. Loss of Feedwater Flow, Loss of Auxiliary Power	B	209
	c. Turbine Bypass, Single Safety or Relief Valve Blowdown	B	12
8.	Reduction to 0% Power, Hot Standby, Shutdown ($100^{\circ}F/hr$ Cooldown Rate) (2)	A	378
9.	Refueling Shutdown with Head Spray and Unbolt (1)	A	68
10.	Scram:		
	a. Reactor Overpressure with Delayed Scram (Anticipated Transient Without Scram, ATWS)	C	1(3)
	b. Automatic Blowdown	C	1(3)
11.	Improper or Sudden Start of Recirculation Pump with Cold Bottom Head or Hot Standby - Drain Shut Off - Pump Restart	C	1(3)

Table 3.9-1

PLANT EVENTS

B. Dynamic Loading Events[8]

		ASME Code Service Limit[10]	No. of Cycles/ Events[1]
12.	Operating Basis Earthquake (OBE) Event at Rated Power Operating Conditions	B	10 Cycles (4)
13.	Safe Shutdown Earthquake (SSE) (5) at Rated Power Operating Conditions	D(9)	1(3) Cycle
14.	Turbine Stop Valve Full Closure (TSVC)(6) During Event 7a and Testing	B	990 Cycles
15.	Safety Relief Valve (SRV) Actuation (One, Two Adjacent, All or Automatic Depressurization System) During Event 7a and 7b	B	396 Events(7)
16.	Loss of Coolant Accident (LOCA)		
	Small Break LOCA (SBL)	D(9)	1(3)
	Intermediate Break LOCA (IBL)	D(9)	1(3)
	Large Break LOCA (LBL)	D(9)	1(3)

NOTES:

(1) Some events apply to reactor pressure vessel (RPV) only. The number of events/cycles applies to RPV as an example.

(2) Bulk average vessel coolant temperature change in any one hour period.

(3) The annual encounter probability of a single event is $<10^{-2}$ for a Level C event and $<10^{-4}$ for a Level D event. See Subsection 3.9.3.1.1.5.

(4) 50 peak OBE cycles for piping, 10 peak OBE cycles for other equipment and components.

(5) One stress or load reversal cycle of maximum amplitude.

Table 3.9-1

PLANT EVENTS

B. Dynamic Loading Events

(Continued)

NOTES:

(6) Applicable to main steam piping system only.

(7) The number of reactor building vibratory load cycles on the reactor vessel and internal components is 29,400 cycles of varying amplitude during the 396 events of safety/relief valve actuation.

(8) Table 3.9-2 shows the evaluation basis combination of these dynamic loadings.

(9) Appendix F or other appropriate requirements of the ASME Code are used to determine the service Level D limits, as described in Subsection 3.9.1.4.

(10) These ASME Code Service Limits apply to ASME Code Class 1, 2 and 3 components, component supports and Class CS structures. Different limits apply to Class MC and CC containment vessels and components, as discussed in Section 3.8.

Table 3.9-2

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR SAFETY-RELATED, ASME CODE CLASS 1, 2 AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CLASS CS STRUCTURES

	Plant Event	Service Loading Combination(1),(3),(4)	ASME Service Level(2)	
1.	Normal Operation (NO)	N	A	
2.	Plant/System Operating Transients (SOT)	(a) N + TSVC (b) N + SRV(8)	B(5) B(5)	210.34
3.	NO + OBE	N + OBE	B(5)	210.34
4.	SOT + OBE	(a) N + TSVC + OBE (b) N + SRV(8) + OBE	B(5) B(5)	
5.	Infrequent Operating Transient (IOT), ATWS	N(10) + SRV(8)	C(5),(6),(10)	
6.	SBL	N + SRV (9) + SBL(11)	C(5),(6)	
7.	SBL or IBL + SSE	N + SBL (or IBL)(11) + SSE + SRV(9)	D(5),(6),(7)	
8.	LBL + SSE	N + LBL (14) + SSE	D(5),(6),(7)	
9.	NLF	N + SRV (8) + TSVC (12)	D(5)	210.34

NOTES:

(1) See Legend on the following pages for definition of terms. See Table 3.9-1 for plant events and cycles information.

The service loading combination also applies to Seismic Category I Instrumentation and electrical equipment (See Section 3.10).

(2) The service levels are as defined in appropriate subsection of ASME Section III, Division 1.

(3) For vessels and pumps, loads induced by the attached piping are included as identified in their design specification.

For piping systems, water (steam) hammer loads are included as identified in their design specification.

(4) The method of combination of the loads is in accordance with NUREG-0484, Revision 1.

(5) For active Class 1, 2 or 3 valves, the design pressure is specified equal to or greater than the pressure for which the valve must operate (open or close).

Table 3.9-2

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR SAFETY-RELATED, ASME CODE CLASS 1, 2 AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CLASS CS STRUCTURES (Continued)

NOTES:

(6) All ASME Code Class 1, 2 and 3 Piping Systems which are essential for safe shutdown under the postulated events are designed to meet the requirements of NEDO-21985 (Reference 7) and NRC's "Evaluation of Topical Report - Piping Functional Capability Criteria," by MEB dated July 17, 1980.

(7) For active Class 2 and 3 valves and pumps, the stresses are limited by criteria: $\sigma m \leq 1.2S$, and $(\sigma m \text{ or } \sigma L) + \sigma b \leq 1.8S$, where the notations are as defined in the ASME Code, Section III, subsections NC or ND, respectively.

(8) The most limiting load combination case among SRV(1), SRV(2) and SRV (ALL). For main steam and branch piping evaluation, additional loads associated with relief line clearing and blowdown into the suppression pool are included.

(9) The most limiting load combination case among SRV(1), SRV(2) and SRV (ADS). See Note (8) for main steam and branch piping.

(10) The reactor coolant pressure boundary is evaluated using in the load combination the maximum pressure expected to occur during ATWS.

(11) The piping systems that are qualified to the leak-before-break criteria of Subsection 3.6.3 are excluded from the pipe break events to be postulated for design against LOCA dynamic effects, viz., SBL, IBL and LBL.

(12) This applies only to the main steam lines and components mounted on it. The low probability that the TSVC and SRV loads can exist at the same time results in this combination being considered under service level D. 210.34

LOAD DEFINITION LEGEND:

Normal (N) - Normal and/or abnormal loads associated with the system operating conditions, including thermal loads, depending on acceptance criteria.

SOT System Operational Transient (see Subsection 3.9.3.1).

IOT Infrequent Operational Transient (see Subsection 3.9.3.1).

ATWS - Anticipated Transient Without Scram.

TSVC - Turbine stop valve closure induced loads in the main steam piping and components integral to or mounted thereon.

RBV Loads - Dynamic loads in structures, systems and components because of reactor building vibration (RBV) induced by a dynamic event.

OBE RBV loads induced by operational basis earthquake.

NLF - Non LOCA Fault 210.34

Table 3.9-2

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR SAFETY-RELATED, ASME CODE CLASS 1, 2 AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CLASS CS STRUCTURES (Continued)

LOAD DEFINITION LEGEND:

SSE - RBV loads induced by safe shutdown earthquake.

SRV(1), SRV(2) - RBV loads induced by safety/relief valve (SRV) discharge of one or two adjacent valves, respectively.

SRV (ALL) - RBV loads induced by actuation of all safety/relief valves which activate within milliseconds of each other (e.g., turbine trip operational transient).

SRV (ADS) - RBV loads induced by the actuation of safety/relief valves associated with automatic depressurization system which actuate within milliseconds of each other during the postulated small or intermediate break LOCA, or SSE.

LOCA - The loss of coolant accident associated with the postulated pipe failure of a high-energy reactor coolant line. The load effects are defined by $LOCA_1$ through $LOCA_7$. LOCA events are grouped in three categories, SBL, IBL or LBL, as defined here.

$LOCA_1$ - Pool swell (PS) drag/fallback loads on essential piping and components located between the main vent discharge outlet and the suppression pool water upper surface.

$LOCA_2$ - Pool swell (PS) impact loads acting on essential piping and components located above the suppression pool water upper surface.

$LOCA_3$ - (a) Oscillating pressure induced loads on submerged essential piping and components during main vent clearing (VLC), condensation oscillations (CO), or chugging (CHUG), or

(b) Jet impingement (JI) load on essential piping and components as a result of a postulated IBL or LBL event.

Piping and components are defined essential, if they are required for shutdown of the reactor or to mitigate consequences of the postulated pipe failure without offsite power (see introduction to Subsection 3.6).

$LOCA_4$ - RBV load from main vent clearing (VLC).

$LOCA_5$ - RBV loads from condensation oscillations (CO).

$LOCA_6$ - RBV loads from chugging (CHUG).

Table 3.9-2

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR SAFETY-RELATED, ASME CODE CLASS 1, 2 AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CLASS CS STRUCTURES (Continued)

LOAD DEFINITION LEGEND:

$LOCA_7$ - Annulus pressurization (AP) loads due to a postulated line break in the annulus region between the RPV and shieldwall. Vessel depressurization loads on reactor internals (see Subsection 3.9.2.5) and other loads due to reactor blowdown reaction and jet impingement and pipe whip restraint reaction from the broken pipe are included with the AP loads.

SBL - Loads induced by small break LOCA (see Subsections 3.9.3.1.1.3 and 3.9.3.1.1.4); the loads are: $LOCA_3$(a), $LOCA_4$ and $LOCA_6$. See Note (11).

IBL - Loads induced by intermediate break LOCA (see Subsection 3.9.3.1.1.4); the loads are: $LOCA_3$(a) or $LOCA_3$(b), $LOCA_4$, $LOCA_5$ and $LOCA_6$. See Note (11).

LBL - Loads induced by large break LOCA (see Subsection 3.9.3.1.1.4); the loads are: $LOCA_1$ through $LOCA_7$. See Note (11).

Table 3.9-3

PRESSURE DIFFERENTIALS ACROSS REACTOR VESSEL INTERNALS

Reactor Component[3]	Maximum Pressure Differences Occurring During a Steam Line Break (psid) Case 1[1]	Case 2[2]
1. Core plate and guide tube	26.7	23.5
2. Shroud support ring and lower shroud (beneath the core plate)	35.1	37.8
3. Shroud head (at marked elevation)	11.3	21.7
4. Upper shroud (just below top guide)	13.1	22.1
5. Core averaged power fuel bundle (bulge at bottom of bundle)	14.2	13.0
5. Core averaged power fuel bundle (collapse at bottom of top guide)	11.8	11.5
6. Maximum power fuel bundle (bulge at bottom of bundle)	16.2	14.0
7. Top guide	6.2	9.4
8. Steam Dryer	6.9	10.8
- Shroud head to water level, from points (a) to (b), irreversible pressure drop	13.4	23.2
- Shroud head to water level, from points (a) to (b), elevation pressure drop	1.5	2.2

NOTES:

(1) Instantaneous break initiated at 102% rated core power, 102.4% rated steam flow, and 111.1% rated recirculation flow.

(2) Instantaneous break initiated at 54.5% rated core power, 49.8% rated steam flow, and 114.8% rated recirculation flow.

(3) Item numbers in this column correspond to the location (node) numbers identified in Figure 3.9-5.

Table 3.9-4

DEFORMATION LIMIT FOR SAFETY CLASS REACTOR INTERNAL STRUCTURES ONLY

	Either One of (Not Both)	General Limit	
a.	$\dfrac{\text{Permissible Deformation, DP}}{\text{Analyzed Deformation Causing Loss of Function, DL}}$	$\leq \dfrac{0.9}{SF_{min}}$	
b.	$\dfrac{\text{Permissible Deformation, DP}}{\text{Experiment Deformation Causing Loss of Function, DE}}$	$\leq \dfrac{1.0}{SF_{min}}$	(Note 1) 21046

Where:

DP = Permissible deformation under stated conditions of Service levels A, B, C or D (normal, upset, emergency or fault)

DL = Analyzed deformation which could cause a system loss of functions(2)

DE = Experimentally determined deformation which could cause a system loss of function

SF_{min} = Minimum safety factor (see Subsection 3.9.5.3.6)

21046 (1) Equation b will not be used unless supporting data are provided to the NRC by General Electric.

(2) "Loss of Function" can only be defined quite generally until attention is focused on the component of interest. In cases of interest, where deformation limits can affect the function of equipment and components, they will be specifically delineated. From a practical viewpoint, it is convenient to interchange some deformation condition at which function is assured with the loss of function condition if the required safety margins from the functioning conditions can be achieved. Therefore, it is often unnecessary to determine the actual loss of function condition because this interchange procedure produces conservative and safe designs. Examples where deformation limits apply are: control rod drive alignment and clearances for proper insertion, reactor internal pump wear, or excess leakage of any component.

Table 3.9-5

PRIMARY STRESS LIMIT FOR SAFETY CLASS REACTOR INTERNAL STRUCTURES ONLY

	Any One Of (No More Than One Required)	General Limit	
a.	$\frac{\text{Elastic evaluated primary stresses, PE}}{\text{Permissible primary stresses, PN}}$	$\leq \frac{2.25}{SF_{min}}$	
b.	$\frac{\text{Permissible load, LP}}{\text{Largest lower bound limit load, CL}}$	$\leq \frac{1.5}{SF_{min}}$	
c.	$\frac{\text{Elastic evaluated primary stress, PE}}{\text{Conventional ultimate strength at temperature, US}}$	$\leq \frac{0.75}{SF_{min}}$	
d.	$\frac{\text{Elastic-plastic evaluated nominal primary stress, EP}}{\text{Conventional ultimate strength at temperature, US}}$	$\leq \frac{0.9}{SF_{min}}$	
e.	$\frac{\text{Permissible load, LP}}{\text{Plastic instability load, PL}}$	$\leq \frac{0.9}{SF_{min}}$	(Note 1)
f.	$\frac{\text{Permissible load, LP}}{\text{Ultimate load from fracture analysis, UF}}$	$\leq \frac{0.9}{SF_{min}}$	(Note 1)
g.	$\frac{\text{Permissible load, LP}}{\text{Ultimate load or loss of function load from test, LE}}$	$\leq \frac{1.0}{SF_{min}}$	(Note 1)

21C.46

where

PE = Primary stresses evaluated on an elastic basis. The effective membrane stresses are to be averaged through the load carrying section of interest. The simplest average bending, shear or torsion stress distribution which will support the external loading will be added to the membrane stresses at the section of interest.

PN = Permissible primary stress levels under service level A or B (normal or upset) conditions under ASME Boiler and Pressure Vessel Code, Section III.

LP = Permissible load under stated conditions of service level A, B, C or D (normal, upset, emergency or faulted).

Table 3.9-5

PRIMARY STRESS LIMIT
FOR SAFETY CLASS REACTOR INTERNAL STRUCTURES ONLY
(Continued)

where

CL = Lower bound limit load with yield point equal to 1.5 Sm where Sm is the tabulated value of allowable stress at temperature of the ASME III code or its equivalent. The "lower bound limit load" is here defined as that produced from the analysis of an ideally plastic (non-strain hardening) material where deformations increase with no further increase in applied load. The lower bound load is one in which the material everywhere satisfies equilibrium and nowhere exceeds the defined material yield strength using either a shear theory or a strain energy of distortion theory to relate multiaxial yield to the uniaxial case.

US = Conventional ultimate strength at temperature or loading which would cause a system malfunction, whichever is more limiting.

EP = Elastic plastic evaluated nominal primary stress. Strain hardening of the material may be used for the actual monotonic stress strain curve at the temperature of loading or any approximation to the actual stress curve which everywhere has a lower stress for the same strain as the actual monotonic curve may be used. Either the shear or strain energy of distortion flow rule may be used.

PL = Plastic instability loads. The "Plastic Instability Load" is defined here as the load at which any load bearing section begins to diminish its cross-sectional area at a faster rate than the strain hardening can accommodate the loss in area. This type analysis requires a true stress-true strain curve or a close approximation based on monotonic loading at the temperature of loading.

UF = Ultimate load from fracture analyses. For components which involve sharp discontinuities (local theoretical stress concentration) the use of a "Fracture Mechanics" analysis where applicable utilizing measurements of plane strain fracture toughness may be applied to compute fracture loads. Correction for finite plastic zones and thickness effects as well as gross yielding may be necessary. The methods of linear elastic stress analysis may be used in the fracture analysis where its use is clearly conservative or supported by experimental evidence. Examples where "Fracture Mechanics" may be applied are for fillet welds or end of fatigue life crack propagation.

LE = Ultimate load or loss of function load as determined from experiment. In using this method, account shall be taken of the dimensional tolerances which may exist between the actual part and the tested part or parts as well as differences which may exist in the ultimate tensile strength of the actual part and the tested parts. The guide to be used in each of these areas is that the experimentally determined load shall be adjusted to account for material property and dimension variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.

SF_{min} = Minimum safety factor (see Subsection 3.9.5.3.6).

21046

(1) Do not use unless supporting data are provided to the NRC by General Electric.

Table 3.9-6

BUCKLING STABILITY LIMIT FOR SAFETY CLASS REACTOR INTERNAL STRUCTURES ONLY

	Any One Of (No More Than one Required)	General Limit	
a.	$\dfrac{\text{Permissible load, LP}}{\text{Service level A (normal) permissible load, PN}}$	$\le \dfrac{2.25}{SF_{min}}$	
b.	$\dfrac{\text{Permissible load, LP}}{\text{Stability analysis load, SL}}$	$\le \dfrac{0.9}{SF_{min}}$	
c.	$\dfrac{\text{Permissible load, LP}}{\text{Ultimate buckling collapse load from test, SET}}$	$\le \dfrac{1.0}{SF_{min}}$	(Note 1) 21046

where

LP = Permissible load under stated conditions of service levels A, B, C or D (normal, upset, emergency or faulted).

PN = Applicable service level A (normal) event permissive load.

SL = Stability analysis load. The ideal buckling analysis is often sensitive to otherwise minor deviations from ideal geometry and boundary conditions. These effects shall be accounted for in the analysis of the buckling stability loads. Examples of this are ovality in externally pressurized shells or eccentricity on column members.

SET = Ultimate buckling collapse load as determined from experiment. In using this method, account shall be taken of the dimensional tolerances which may exist between the actual part and the tested part. The guide to be used in each of these areas is that the experimentally determined load shall be adjusted to account for material property and dimension variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.

SF_{min} = Minimum safety factor (see Subsection 3.9.5.3.6)

(1) Equation C will not be used unless supporting data are provided to the NRC by General Electric. 21046

Table 3.9-7

FATIGUE LIMIT
FOR SAFETY CLASS REACTOR INTERNAL STRUCTURES ONLY

Summation of fatigue damage usage following Minor hypotheses[1]:

Cumulative Damage in Fatigue	Limit for Service Levels A&B (Normal and Upset Conditions)
Design fatigue cycle usage from analysis using the method of the ASME Code	≤ 1.0

NOTE

(1) Miner, M.A., *Cumulative Damage in Fatigue*, Journal of Applied Mechanics, Vol. 12, ASME, Vol. 67, pp A159-A164, September 1945.

Table 3.9-8

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

This table responds to NRC Questions 210.47, 210.48 and 210.49 regrading provisions for inservice testing of safety-related pumps and valves within the scope of the ABWR Standard Plant in accordance with the ASME Code. The information is presented separately for each system for the MPL numbers given below.*

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

MPL	SYSTEM	PUMP PAGE	VALVE PAGE
P52	Instrument Air		3.9-58.25
P54	High Pressure Nitrogen Gas Supply		3.9-58.25
T22	Standby Gas Treatment		3.9-58.26
T31	Atmospheric Control		3.9-58.27
T49	Flammability Control		3.9-58.29
U41	Heating, Ventilating and Air Conditioning		3.9-58.30

* See end of table for notes.

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

System Pumps

No.	Qty	Description	Safety Class (a)	Test Param (b)	Test Freq. (f)	SSAR Fig.
B31-C001	10	Reactor Recirc Sys (RPS) Reactor Internal Pump	1	E10		5.4-4a
C41-C001	2	Standby Liquid Control System pump	2	P,Vv	3 mo	9.3-1
				Q	2 yrs	
E11-C001	3	Residual Heat Removal System Pump	2	DP,Q,Vv	3 mo	5.4-10c,d,f
E11-C002	3	Residual Heat Removal System fill pump	2	E10		5.4-10c,d,f
E22-C001	2	High Pressure Core Flooder pump	2	DP,Q,Vv	3 mo	6.3-7b
E51-C001	1	Reactor Core Isolation Cooling pump	2	Q,N,DP, Vd,Vv	3 mo	5.4-8a
P21-C001	6	Reactor Building Cooling Water pump	3	E10		9.2-1a,d,g
P25-C001	4	HVAC Emergency Cooling Water Sys pump	3	E10		9.2-3a,b
P41-C001	6	Reactor Service Water System pump	3	E10		

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

B21 Nuclear Boiler System Valves

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq (f)	SSAR Fig.
F001	2	Feedwater line Motor-Operated Valve (MOV)	2	A	I,A	L,P,S	2 yrs	5.1-3d
F002	2	Upstream (First) FW line check valve	2	C	A	S	2 yrs	5.1-3d
F003	2	FW line outboard check valve-Air-Operated (AO)	1	C	I,A	L,P,S	2 yrs	5.1-3d
F004	2	FW line inboard check valve	1	C	I,A	L,S	2 yrs	5.1-3d
F005	2	FW line inboard maintenance valve	1	B	P	P	2 yrs	5.1-3d
F006	2	RWCU (or CUW) System injection line check valve	2	C	A	S	2 yrs	5.1-3d
F007	2	RWCU (or CUW) System injection line MOV	2	A	I,A	L,P	2 yrs	5.1-3d
						S	3 mo	
F008	4	Inboard Main Steam Iso. Vlv. (MSIV)	1	A	I,A	L,P	2 yrs	5.1-3c
						S	3 mo	
F009	4	Outboard Main Steam Iso. Vlv (MSIV)	1	A	I,A	L,P	2 yrs	5.1-3c
						S	3 mo	
F010	18	Safety/Relief Valve (SRV)	1	C	A	L	5 yrs	5.1-3b
						S(ADS)	2 yrs	
F011	1	MSL bypass/drain line inb. iso. vlv	1	A	I,A	L,P	2 yrs	5.1-3c
						S	3 mo	
F012	1	MSL bypass/drain line outb. iso. vlv	1	A	I,A	L,P	2 yrs	5.1-3c
						S	3 mo	
F018	1	RPV non-condensible gas removal line	1	B	P			5.1-3b
F019	1	RPV head vent inboard shutoff valve	1	A	P	L, P	2 yrs	5.1-3b
F020	1	RPV head vent outboard shutoff valve	1	A	P	L, P	2 yrs	5.1-3b
F021	18	SRV discharge line vacuum breaker	3	C	A	S	2 yrs	5.1-3b
F022	18	SRV discharge line vacuum breaker	3	C	A	S	2 yrs	5.1-3b
F024	4	Inboard MSIV air supply line check valve	3	C	A	L, S	2 yrs	5.1-3c
F025	4	Outboard MSIV air supply line check valve	3	C	A	L, S	2 yrs	5.1-3c
F026	8	SRV ADS pnuematic supply line check valve	3	C	A	L, S	2 yrs	5.1-3b
F031	2	Inboard valve on the outb. FW line check valve test line	2	B	I,P	E1		5.1-3d
F033	4	Inboard shutoff valve on the outboard MSIV test line	2	B	I,P	E1		5.1-3c
F035	1	Inboard test line valve for the MSL bypass/drain valve	2	B	I,P	E1		5.1-3c
F039	2	Inboard test line valve for the inboard FW line check valve	2	B	P	E1		5.3-3c
F040	2	Outboard test line valve for the FW line check valve	2	B	P	E1		5.1-3d
F500	2	Inboard drain line test valve for the first FW line check valve	2	B	P	E1		5.1-3d
F503	2	Outboard drain line valve for the FW line check valve	2	B	P	E1		5.1-3d

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

B21 Nuclear Boiler System Valves (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq (f)	SSAR Fig.
F508	4	Inboard MSIV accumulator vent line valve	3	B	P	E1		5.1-3c
F509	4	Outboard MSIV accumulator vent line valve	3	B	P	E1		5.1-3c
F510	8	SRV ADS accumulator vent line valve	3	B	P	E1		5.1-3b
F700	4	Root valve - RPV reference leg water level instrument line	2	B	P	E1		5.1-3e,f
F701	4	Isolation valve - RPV reference leg water level instrument line	2	C	I,A	L,S	2 yrs	5.1-3e,f
F702	4	Root valve - RPV narrow range water level instrument line	2	B	P	E1		5.1-3e,f
F703	4	Isolation valve - RPV narrow range water level instrument line	2	C	I,A	L,S	2 yrs	5.1-3e,f
F704	4	Root valve - RPV wide range water level instrument line	2	B	P	E1		5.1-3e,f
F705	4	Isolation valve - RPV wide range water level instrument line	2	C	I,A	L,S	2 yrs	5.1-3e,f
F706	1	Root valve - Reactor well water level instrument line	2	B	P	E1		5.1-3e
F709	1	Root valve - RPV head vent line instrument line	2	B	P	E1		5.1-3b
F710	1	Isolation valve - RPV head vent line instrument line	2	C	I,A	L,S	2 yrs	5.1-3b
F711	1	Root valve - RPV head seal leakage instrument line	2	B	P	E1		5.1-3h
F712	1	Isolation valve to RPV head seal leakage instrument line	2	C	I,A	L,S	2 yrs	5.1-3h
F713	4	Root valve - RPV above pump deck instrument line	2	B	P	E1		5.1-3g
F714	4	Isolation valve - RPV above pump deck instrument line	2	C	I,A	L,S	2 yrs	5.1-3g
F715	4	Root valve - RPV below pump deck instrument line	2	B	P	E1		5.1-3g
F716	4	Isolation valve - RPV below pump deck instrument line	2	C	I,A	L,S	2 yrs	5.1-3g
F717	4	Root valve - RPV above core plate instrument line	2	B	P	E1		5.1-3g
F718	4	Isolation valve - RPV above core plate instrument line	2	C	I,A	L,S	2 yrs	5.1-3g
F719	4	Root valve - RPV below core plate instrument line	2	B	P	E1		5.1-3g
F720	4	Isolation valve - RPV below core plate instrument line	2	C	I,A	L,S	2 yrs	5.1-3g

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

B21 Nuclear Boiler System Valves (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F723	4	Root valve - MSL flow restrictor instrument line	2	B	P	E1		5.1-3b
F724	4	Isoaltion valve - MSL flow restrictor instrument line	2	C	I,A	L,S	2 yrs	5.1-3b
F725	4	Root valve - MSL flow restrictor instrument line	2	B	P	E1		5.1-3b
F726	4	Isolation valve - MSL flow restrictor instrument line	2	C	I,A	L,S	2 yrs	5.1-3b

B31 Reactor Recirculation Internal Pump Valves

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F008	10	RIP pump motor purge water line outboard isoaltion valve	2	A	I,A	L	2 yrs	5.4-4b
F009	10	RIP pump motor purge water line inboard isolation valve	2	A	I,A	L	2 yrs	5.4-4b
F010	10	RIP pump motor purge water supply line valve	3	B	P	E1		5.4-4a
F011	10	RIP inflatable pressurized water line inboard valve	3	B	P	E1		5.4-4a
F013	10	RIP seal equalizing line valve	3	B	P	E1		5.4-4a
F500	10	RIP cooling water HX vent line inboard valve	3	B	P	E1		5.4-4a
F502	10	RIP drain line inboard valve	3	B	P	E1		5.4-4a
F505	10	RIP cooling water HX shell drain line inboard valve	3	B	P	E1		5.4-4a

C12 Control Rod Drive System Valves

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F719	4	Root valve charging line header pressure instrument line	2	B	P	E1		4.6-8b
F720	4	Root valve charging line header pressure instrument line	2	B	P	E1		4.6-8b

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

C41-Standby Liquid Control System Valves

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F001	2	SLCS storage tank outlet line MOV	2	B	A	S	3 mo.	9.3-1
						P	2 yrs	
F002	2	SLCS pump suction line maintenance valve	2	B	P	E1		9.3-1
F003	2	SLCS pump discharge line relief valve	2	C	P	P,S	5yrs	9.3-1
F004	2	SLCS pump discharge line check valve	2	C	A	S	3 mo.	9.3-1
F005	2	SLCS pump discharge line maintenance valve	2	B	P	E1		9.3-1
F006	2	SLCS pump discharge line MOV	2	A	I,A	L, P	2 yrs	9.3-1
						S	3 mo	
F007	1	SLCS injection line outboard check valve	2	A,C	I,A	L,S	2 yrs	9.3-1
F008	1	SLCS injection line inboard check valve	2	A,C	I,A	L,S	2 yrs	9.3-1
F010	1	SLCS test tank return line inboard shutoff valve	2	B	P	E1		9.3-1
F012	1	SLCS test tank outlet line shutoff valve	2	B	P	E1		9.3-1
F014	1	SLCS pump suct line demin water supply line	2	B	P	E1		9.3-1
F018	1	SLCS storage tank sample line inboard shutoff valve	2	B	P	E1		9.3-1
F020	1	SLCS pump suction line demin water supply line bypass line	2	B	P	E1		9.3-1
F025	1	SLCS injection line test/vent line inb vlv	2	B	P	E1		9.3-1
F026	1	SLCS pump suction line relief valve	2	C	P	L,P	5 yrs	9.3-1
F500	1	SLCS pump suction line drain line	2	B	P	E1		9.3-1
F501	2	SLCS pump discharge line drain line valve	2	B	P	E1		9.3-1
F700	2	SLCS test tank return line instr line valve	2	B	P	E1		9.3-1

C51 Neutron Monitoring (ATIP) System Valves

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
J004	3	Isolation valve assembly	2	A,C,D	P	L,P	2 yrs	7.6-1c
J011	3	Purge isolation valve	2	A,C	P	L,P	2 yrs	7.6-1c
J012	3	Manual gate vale	2	A	P	E1		7.6-1c

D23 Containment Atmosphere Monitoring System Valves

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F001	2	CAMS drywell pressure instrument line outboard isolation valve	2	A	I,A	L	3 mo	7.6-7c
F004	2	CAMS drywell sample line outboard containment isolation valve	2	A	I,A	L,P	3 mo	7.6-7c
F005	2	CAMS drywell return line outboard containment isolation valve	2	A	I,A	L,P	3 mo	7.6-7c

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

D23 Containment Atmosphere Monitoring System Valves (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F006	2	CAMS wetwell sample line outboard containment isolation valve	2	A	I,A	L,P	3 mo	7.6-7c
F007	2	CAMS wetwell return line outboard containment isolation valve	2	A	I,A	L,P	3 mo	7.6-7c
F008	2	CAMS rack drain line outboard containment isolation valve	2	A	I,A	L,P	3 mo	7.6-7c
F009	2	CAMS drywell pressure instrument line outboard isolation valve	2	A	I,P	S	3 mo	7.6-7c
F010	2	CAMS drywell sample line outboard containment isolation valve	2	A	I,P	S	3 mo	7.6-7c
F011	2	CAMS drywell return line outboard containment isolation valve	2	A	I,P	S	3 mo	7.6-7c
F012	2	CAMS wetwell sample line outboard containment isolation valve	2	A	I,P	S	3 mo	7.6-7c
F013	2	CAMS wetwell return line outboard containment isolation valve	2	A	I,P	S	3 mo	7.6-7c
F014	2	CAMS rack drain line outboard containment isoaltion valve	2	A	I,P	S	3 mo	7.6-7c
F100	2	CAMS rack drywell sample line maint. valve	3	B	P	E2		7.6-7d
F101	2	CAMS rack wetwell sample line maint. valve	3	B	P	E2		7.6-7d
F102	2	CAMS rack accident sample booster pump inlet valve	3	B	P	E2		7.6-7d
F103	2	CAMS rack accident sample booster pump outlet valve	3	B	P	E2		7.6-7d
F104	2	CAMS rack accident sample booster pump bypass line check valve	3	C	A	E2		7.6-7d
F105	2	CAMS rack accident sample booster pump line solenoid valve	3	B	A	E2		7.6-7d
F106	2	CAMS rack booster pumps discharge line pressure control valve	3	B	A	E2		7.6-7d
F107	2	CAMS rack sample pumps inlet press cont. vlv	3	B	A	E2		7.6-7d
F108	2	CAMS rack sample pump bypass line sol. vlv	3	B	P	E2		7.6-7d
F112	2	CAMS rack sample return line to drywell (DW)/wetwell (WW)	3	B	P	E2		7.6-7d
F116	2	CAMS rack sample return line to drywell (DW)/wetwell (WW)	3	B	P	E2		7.6-7d
F117	2	CAMS rack sample return line to drywell (DW)/wetwell (WW)	3	C	A	E2		7.6-7d
F118	2	CAMS rack steam separator condensate line to DW/WW drain line	3	B	P	E2		7.6-7d
F121	2	CAMS rack steam separator condensate line to DW/WW	3	B	P	E2		7.6-7d

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

D23 Containment Atmosphere Monitoring System Valves (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F128	2	CAMS rack line form the CAMS Gas Calibration Rack check valve	3	C	A	E2		7.6-7d
F190	2	CAMS rack normal sample pump inlet solenoid valve	3	B	A	E2		7.6-d
F191	2	CAMS rack normal sample pump discharge solenoid valve	3	B	A	E2		7.6-7d
F193	2	CAMS rack accident sample pump discharge line solenoid valve	3	B	A	E2		7.6-7d
F195	2	CAMS rack normal sample booster pump outlet line solenoid valve	3	B	A	E2		7.6-7d
F197	2	CAMS rack normal sample booster pump outlet line solenoid valve	3	B	A	E2		7.6-7d
F201	2	CAMS rack drywell sample line admis. valve	3	B	A	E2		7.6-7d
F202	2	CAMS rack drywell sample line admis. valve	3	B	A	E2		7.6-7d
F510	2	CAMS rack steam separator condensate line exit AO valve	3	B	A	E2		7.6-7d
F512	2	CAMS rack drain line needle valve	3	B	P	E2		7.6-7d
F513	2	CAMS rack drain line Air-Operated Valve	3	B	A	E2		7.6-7d
F515	2	CAMS rack dehumidifier condensate line Air-Operated Valve	3	B	A	E2		7.6-7d
F520	2	CAMS rack drain line maintenance valve	3	B	P	E2		7.6-7d

E11 Residual Heat Removal System Valves

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F001	3	Suppression pool suction valve	2	A	I,A	P	2 yrs	5.4-10c,d,f
						S	3 mo	
F002	3	RHR pump discharge line check valve	2	C	A	S	3 mo	5.4-10c,d,f
F003	3	RHR pump disharge line maintainence valve	2	B	P	E1		5.4-10c,d,f
F004	3	Heat Exchanger flow control valve	2	B	A	P	2 yrs	5.4-10c,d,f
						S	3 mo	
F005	1	RPV injection valve	2	B	A	P	2 yrs	5.4-10c
						S	CS	
F005	2	RPV injection valve	1	A	I,A	L,P	2 yrs	5.4-10e,g
						S	CS	
F006	1	RPV injection line check valve	2	B	A	P	2 yrs	5.5-10c
						S	CS	
F006	2	RPV injection line check valve	1	A	I,A	L,P	2 yrs	5.4-10e,g
						S	CS	
F007	2	RPV injection line inboard maint. valve	1	B	P	E1		5.4-10e,g

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

E11 Residual Heat Removal System Valves (Continued)

No.	Quan	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F008	3	Suppression pool return line MOV	2	B	I,A	P	2 yrs	5.4-10c,d,f
						S	3 mo	
F009	3	Shutdown Cooling suct. line maint. vlv	1	B	P	E3		5.4-10b
F010	3	Shutdown Cooling suct. line inb. iso. vlv	1	A	I,A	L,P	2 yrs	5.4-10b
						S	CS	
F011	3	Shutdown Cooling suct line outb iso. vlv	1	A	I,A	L,P	2 yrs	5.4-10b
						S	CS	
F012	3	Shutdown Cooling suction line adm. vlv	2	B	A	P	2 yrs	5.4-10c,d,f
						S	3 mo	
F013	3	Heat exchanger bypass flow control vlv	2	B	A	P	2 yrs	5.4-10c,d,f
						S	3 mo	
F014	2	Fuel Pool Cooling return line inb MOV	2	B	P	P	2 yrs	5.4-10e,g
F015	2	Fuel Pool Cooling return line outb MOV	2	B	P	P	2 yrs	5.4-10e,g
F016	2	Gate vlv-line from Fuel Pool Clg (FPC)	2	B	P	P	2 yrs	5.4-10b
F017	2	Drywell spray line inboard valve	2	B	I,A	L,P	2 yrs	5.4-10e,g
						S	RO	
F018	2	Drywell spray line outboard valve	2	B	I,A	L,P	2 yrs	5.4-10e,g
						S	RO	
F019	2	Wetwell spray line MOV	2	B	I,A	L,P	2 yrs	5.4-10e,g
						S	RO	
F020	3	RHR pump min flow bypass line check vlv	2	B,C	P	P	2 yrs	5.4-10c,d,f
F021	3	RHR pump min flow bypass line MOV	2	B	I,A	P	2 yrs	5.4-10c,d,f
						S	3 mo	
F022	3	Discharge line fill pump suction line valve	2	B	P	P	2 yrs	5.4-10c,d,f
F023	3	Fill pump discharge line check valve	2	B, C	A	S	CS	5.4-10c,d,f
F024	3	Fill pump discharge line stop check valve	2	B, C	A	S	CS	5.4-10c,d,f
F025	3	Fill pump minimum flow line globe valve	2	B	P	P	2 yrs	5.4-10c,d,f
F026	3	RHR pump suction to High Conductivity Waste (HCW)	2	B	P	E1		5.4-10c,d,f
F027	3	Bypass line around the check valve MPL E11-F002	2	B	P	E1		5.4-10c,d,f
F028	3	Heat exchanger outlet line relief valve	2	B,C	A			5.4-10c,d,f
F029	3	Inboard reactor well drain line valve	2	B	P	E1		5.4-10c,d,f
F030	3	Drain to radwaste valve	2	B	P	E1		5.4-10c,d,f
F031	3	Outb reactor well drain line valve (to SP)	2	B	I,P	E1		5.4-10c,d,f
F032	3	Shutoff valve - line from MUWC	2	B	P	E1		5.4-10c,f,g
F033	3	Check valve in the line from MUWC	2	B,C	A	E1		5.4-10c,f,g
F034	2	RPV injection line vent/test line outb vlv	2	B	P	E1		5.4-10e,g
F036	1	Press equal valve around chk vlv E11-F006	2	A	P	E1		5.4-10c
F036	2	Press equal valve around chk vlv E11-F006	1	A	P	E1		5.4-10e,g

Table 3.9-8 (Coninued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

E11 Residual Heat Removal System Valves (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F037	3	Shutdown cooling suction line test line	1	A	P	E1		5.4-10b
F039	3	Relief vlv around the MOV MPL E11-F011	1	A,C	A	E1		5.4-10b
F040	3	Shutoff valve - line from MUWC	2	B	P	E1		5.4-10b
F041	3	Check valve - line from Make-Up Water Condenser (MUWC)	2	B,C	P	E1		5.4-10b
F042	3	Shutdown Cooling Mode suction line relief valve	2	B,C	A	E1		5.4-10c,d,f
F043	3	HX outlet to the Sampling System (SS) test inboard valve	2	B	P	E1		5.4-10c,f,g
F045	1	HX outlet to the PASS - inboard valve	2	B	A	P	2 yrs	5.4-10c
				B		S	3 mo	
F049	2	Drywell spray line vent & test line inboard valve	2	B	P	E1		5.4-10e,g
F051	3	Fill pump discharge line relief valve	2	B	A	E1		5.4-10c,d,f
F052	1	Drain line for the suppression pool	2	B	P	E1		5.4-10d
F102	1	AC independeat water addition inp[illegible] vlv	2	B	A	S	3 mo	5.4-10g
F500	3	Heat exchanger inlet drain line inboard valve	2	B	P	E1		5.4-10c,d,f
F502	3	HX outlet line drain line inboard vlv	2	B	P	E1		5.4-10c,d,f
F504	3	RPV injection line vent line inb vlv	2	B	P	E1		5.4-10c,f,g
F506	1	RPV injection line drain line inb vlv	2	B	P	E1		5.4-10c
F506	2	RPV injection line drain line inb vlv	1	B	P	E1		5.4-10e,g
F508	3	Shutdown Cooling suct line vent line vlv	2	B	P	E1		5.4-10b
F511	2	Drywell spray line inboard drain line vlv	2	B	P	E1		5.4-10e,g
F513	2	Drywell spray line inboard drain line vlv	2	B	P	E1		5.4-10e,g
F515	2	Wetwell spray line inboard drain line vlv	2	B	P	E1		5.4-10e,g
F517	3	RHR pump min flow line drn line inb vlv	2	B	P	E1		5.4-10c,d,f
F700	3	RHR pump suction line pressure instr line	2	B	P	E1		5.4-10c,d,f
F701	3	RHR pump suction line pressure instr line	2	B	P	E1		5.4-10c,d,f
F702	3	RHR pump discharge line press. instr line	2	B	P	E1		5.4-10c,d,f
F704	3	RHR pump discharge line press. instr line	2	B	P	E1		5.4-10c,d,f
F706	3	RHR pump discharge line press. instr line	2	B	P	E1		5.4-10c,d,f
F707	3	RHR pump discharge line press. instr line	2	B	P	E1		5.4-10c,d,f
F708	3	FT MPL E11-FT008 instr line inb root vlv	2	B	P	E1		5.4-10c,d,f
F709	3	FT MPL E11-FT008 instr line outb root vlv	2	B	P	E1		5.4-10c,d,f
F710	3	FT MPL E11-FT008 instr line inb root vlv	2	B	P	E1		5.4-10c,d,f
F711	3	FT MPL E11-FT008 instr line outb root vlv	2	B	P	E1		5.4-10c,d,f
F712	3	Shutdown Cooling Mode suction line pressure instrument line	2	B	P	E1		5.4-10c,d,f
F713	3	Fill pump suction line instrument line valve	2	B	P	E1		5.4-10c,d,f
F714	1	Discharge to radwaste flow instr line	2	B	P	E1		5.4-10d
F716	1	Discharge to radwaste flow instr line	2	B	P	E1		5.4-10d

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

E22 High Pressure Core Flooder System Valves

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq.	SSAR Fig.
F001	2	Condensate Storage Pool (CSP) suction line MOV	2	B	A	P S	2 yrs 3 mo	6.3-7b
F002	2	CSP suction line check valve	2	B,C	A	S	3 mo	6.3-7b
F003	2	HPCF System injection valve	1	A	I,A	L,P S	2 yrs CS	6.3-7a
F004	2	HPCF System inboard check valve	1	A,C	I,A	L,P S	2 yrs 3 mo	6.3-7a
F005	2	Pump discharge line inboard maint valve	1	B	P	E1		6.3-7a
F006	2	Suppression pool suction line MOV	2	A	I,A	P S	2 yrs 3 mo	6.3-7b
F007	2	Suppression pool suction line check valve	2	B,C	A	S	3 mo	6.3-7b
F008	2	Test return line inboard valve	2	B	A	P S	2 yrs 3 mo	6.3-7b
F009	2	Test return line outboard valve	2	A	I,A	P S	2 yrs 3 mo	6.307b
F010	2	Pump minimum flow bypass line MOV	2	A	I,A	P S	2 yrs 3 mo	6.3-7b
F011	2	Bypass line shutoff valve around check valve E22-F0[illegible]	2	B	P	E1		6.3-7b
F012	2	HPCI pump suction line drain line to HCW	2	B	P	E1		6.3-7b
F015	2	Pump discharge line fill line check vlv	2	B,C	A	S	RO	6.3-7a
F017	2	Pump discharge line test and vent line inboard valve	1	A	P	E1		6.3-7a
F019	2	Pressure equalizing valve around check valve E22-F004	1	A	P	E1		6.3-7a
F020	2	Suppression pool suction line relief valve	2	B,C	A			6.3-7b
F500	2	Pump discharge line high point vent inboard valve	2	B	P	E1		6.3-7a
F700	2	Pump suction line pressure instrument line root valve	2	B	P	E1		6.3-7b
F701	2	Pump suction line pressure instrument line root valve	2	B	P	E1		6.3-7b
F702	2	Pump discharge line pressure instrument line inboard valve	2	B	P	E1		6.3-7b
F704	2	Pump discharge line pressure instrument line inboard valve	2	B	P	E1		6.3-7b
F705	2	Pump discharge line pressure instrument line outboard valve	2	B	P	E1		6.3-7b
F706	2	Pump discharge line flow instrument line inboard valve	2	B	P	E1		6.3-7a

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

E22 High Pressure Core Flooder System Valves (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F707	2	Pump discharge line flow instrument line outboard valve	2	B	P	E1		6.3-7a
F708	2	Pump discharge line flow instrument line inboard valve	2	B	P	E1		6.3-7a
F709	2	Pump discharge line flow instrument line outboard valve	2	B	P	E1		6.3-7a

E31 Leak Detection and Isolation System Valves

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F001	1	Drywell fission product monitoring line maintenance valve	2	B	P	S	3 mo	5.2-8i
F002	1	Drywell fission product monitoring line inboard isolation valve	2	A	I,A	L,P	3 mo	5.2-8i
F003	1	Drywell fission product monitoring line outboard isolation valve	2	A	I,A	L,P	3 mo	5.2-8i
F004	1	Drywell fission product monitoring line outboard isolation valve	2	A	I,A	L,P	3 mo	5.2-8i
F005	1	Drywell fission product monitoring line inboard isolation valve	2	A	I,A	S	3 mo	5.2-8i
F006	1	Drywell fission product monitoring line maintenance valve	2	B	P	S	3 mo	5.2-8i
F009	1	Drywell cooler condensate sampling line vlv	2	A	I,P	L	3 mo	5.2-8h
F010	1	Drywell cooler condensate sampling line vlv	2	A	I,P	L	3 mo	5.2-8h
F701	4	RCIC instrument line isolation valve	2	A	I,P	S	3 mo	5.2-8f
F702	4	RCIC instrument line isolation valve	2	A	I,P	S	3 mo	5.2-8f
F703	4	RCIC instrument line isolation valve	2	A	I,P	S	3 mo	5.2-8f
F704	4	RCIC instrument line isolation valve	2	A	I,P	S	3 mo	5.2-8f

E51 Reactor Core Isolation System Valves

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F001	1	Condensate Storage Pool (CSP) suction line MOV	2	B	A	P,S	3 mo	5.4-8a
F002	1	CSP suction line check valve	2	C	A	P,S	3 mo	5.4-8a
F003	1	RCIC pump discharge line check valve	2	C	A	P,S	3 mo	5.4-8a
F004	1	RCIC System injection valve	2	A	A	L P,S	2 yrs 3 mo	5.4-8a
F005	1	RCIC System discharge line testable check valve	2	C	A	L P,S	2 yrs 3 mo	5.4-8a

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

E51 Reactor Core Isolation Cooling System (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F006	1	Suppression Pool (CSP) suction line MOV	2	A	I,A	L P,S	2 yrs 3 mo	5.4-8a
F007	1	Suppression Pool (CSP) suction line check vlv	2	C	A	P,S	3 mo	5.4-8a
F008	1	RCIC Sys suppr pool test return line MOV	2	B	A	P,S	3 mo	5.4-8a
F009	1	RCIC Sys suppr pool test return line MOV	2	B	I,A	L P,S	2 yrs 3 mo	5.4-8a
F010	1	RCIC Sys minimum flow bypass line check vlv	2	C	A	P,S	3 mo	5.4-8a
F011	1	RCIC Sys minimum flow bypass line MOV	2	B	I,A	L P,S	2 yrs 3 mo	5.4-8a
F012	1	RCIC turbine accessories cooling water line MOV	2	B	A	P,S	3 mo	5.4-8c
F013	1	RCIC turbine accessories cooling water line PCV	2	B	A	E1		5.4-8c
F015	1	Barometric condenser condensate pump discharge line valve	2	B	P	E1		5.4-8c
F016	1	Barometric condenser condensate pump discharge line check valve	2	C	P	P,S	3 mo	5.4-8c
F017	1	RCIC pump suction line relief valve	2	C	A	L,S	2 yrs	5.4-8a
F018	1	Valve in the bypass line around check valve E51-F003	2	B	P	E1		5.4-8a
F019	1	Pump discharge line test line valve	2	B	P	E1		5.4-8a
F020	1	Pump discharge line test line valve	2	B	P	E1		5.4-8a
F021	1	Pump discharge line fill line shutoff valve	2	B	P	E1		5.4-8a
F022	1	Pump discharge line fill line check valve	2	C	A	P,S	3 mo	5.4-8a
F023	1	Pump discharge line fill line check valve	2	C	A	P,S	3 mo	5.4-8a
F024	1	Pump discharge line test line valve	2	B	P	E1		5.4-8a
F025	1	Pump discharge line test line valve	2	B	P	E1		5.4-8a
F026	1	Valve in pressure equalizing line around E51-F005	2	B	P	E1		5.4-8a
F027	1	Suppression Pool (S/P) suction line test line valve	2	B	P	E1		5.4-8a
F028	1	Minimum flow bypass line test line valve	2	B	P	E1		5.4-8a
F029	1	Minimum flow bypass line test line valve	2	B	P	E1		5.4-8a
F030	1	Turbine accessories cooling water line relief valve	2	C	A	L,S	2 yrs	5.4-8c
F031	1	Barometric condenser condensate discharge line AOV to HCW	2	B	P	E1		5.4-8c
F032	1	Barometric condenser condensate discharge line AOV to HCW	2	B	P	E1		5.4-8c
F033	1	Discharge line fill line bypass line shutoff valve	2	B	P	E1		5.4-8a

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

E51 Reactor Core Isolation Cooling System (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F034	1	Barometric condenser condensate pump discharge line test line valve	2	B	P	E1		5.4-8c
F035	1	Steam supply line isolation valve	1	A	I,A	L P,S	2 yrs 3 mo	5.4-8b
F036	1	Steam supply line isolation valve	1	A	I,A	L P,S	2 yrs 3 mo	5.4-8b
F037	1	Steam admission valve	2	B	A	P,S	3 mo	5.4-8a
F038	1	Turbine exhaust line check valve	2	C	I,A	L P,S	2 yrs 3 mo	5.4-8a
F039	1	Turbine exhaust line MOV	2	A	I,A	L P,S	2 yrs 3 mo	5.4-8a
F040	1	Steam supply line drain pot drain line AOV	2	B	P			5.4-8b
F041	1	Steam supply line drain pot drain line AOV	2	B	P			5.4-8b
F044	1	Steam admission valve bypass line maint-tenance valve	2	B	P			5.4-8b
F045	1	Steam admission valve bypass line MOV	2	B	A	P,S	3 mo	5.4-8b
F046	1	Barometric condenser vacuum pump discharge line check valve	2	C	A	L P,S	2 yrs 3 mo	5.4-8a
F047	1	Barometric condenser vacuum pump discharge line MOV	2	A	I,A	L P,S	2 yrs 3 mo	5.4-8a
F048	1	Steam supply line warm-up line valve	1	A	I,A	L P,S	2 yrs 3 mo	5.4-8b
F049	1	Steam supply line test line valve	2	B	P	E1		5.4-8b
F050	1	Steam supply line test line valve	2	B	P	E1		5.4-8b
F051	1	Turbine exhaust line drain line valve	2	B	P	E1		5.4-8c
F052	1	Turbine exhaust line drain line valve	2	B	P	E1		5.4-8c
F053	1	Turbine exhaust line test line valve	2	B	P	E1		5.4-8a
F054	1	Turbine exhaust line vacuum breaker	2	C	A	P,S	3 mo	5.4-8a
F055	1	Turbine exhaust line vacuuum breaker	2	C	A	P,S	3 mo	5.4-8a
F056	1	Steam supply line drain pot drain line test line valve	2	B	P	E1		5.4-8b
F057	1	Steam supply line drain pot drain line test drain line	2	B	P	E1		5.4-8b
F059	1	Baromemtric condener vacuum pump dis-charge line test line valve	2	B	P	E1		5.4-8a
F500	1	Pump discharge line vent line valve	2	B	P	E1		5.4-8a
F501	1	Pump discharge line vent line valve	2	B	P	E1		5.4-8a
F502	1	Pump discharge line drain line valve	2	B	P	E1		5.4-8a
F503	1	Pump discharge line drain line valve	2	B	P	E1		5.4-8a
F700	1	Pump suction line pressure instru-mentation instrument root valve	2	B	P	E1		5.4-8a

3D.4 PIPING

3D.4.1 Piping Analysis Program--PISYS

PISYS is a computer code for analyzing piping systems subjected to both static and dynamic piping loads. Stiffness matrices representing standard piping components are assembled by the program to form a finite element model of a piping system. The piping elements are connected to each other via nodes called pipe joints. It is through these joints that the model interacts with the environment, and loading of the piping system becomes possible. PISYS is based on the linear elastic analysis in which the resultant deformations, forces, moments and accelerations at each joint are proportional to the loading and the superposition of loading is valid.

PISYS has a full range of static dynamic load analysis options. Static analysis includes dead weight, uniformly-distributed weight, thermal expansion, externally-applied forces, moments, imposed displacements and differential support movement (pseudo-static load case). Dynamic analysis includes mode shape extraction, response spectrum analysis, and time-history analysis by modal combination or direct integration. In the response spectrum analysis, i.e. uniform support motion response spectrum analysis (USMA) or independent support motion response spectrum analysis (ISMA), the user may request modal response combination in accordance with NRC Regulatory Guide 1.92. In the ground motion (uniform motion) or independent support time-history analysis, the normal mode solution procedure is selected. In analysis involving time-varying nodal loads, the step by step direct integration method is used.

The PISYS program has been benchmarked against Nuclear Regulatory Commission piping models. The results are documented in a report to the Commission, "PISYS Analysis of NRC Benchmark Problems", NEDO-24210, August 1979 for mode shapes and USMA options. The ISMA option has been validated against NUREG/CR-1677, "Piping Benchmark Problems Dynamic Analysis Independent Support Motion Response Spectrum Method," published in August 1985.

3D.4.2 Component Analysis--ANSI7

ANSI7 is a computer code for calculating stresses and cumulative usage factors for Class 1, 2 and 3 piping components in accordance with articles NB, NC and ND-3650 of the ASME Code, Section III. ANSI7 is also used to combine loads and calculate combined service level A, B, C and D loads on piping supports and pipe mounted equipment.

3D.4.3 Area Reinforcement--NOZAR

The computer program NOZAR (Nozzle Area reinforcement Program) performs an analysis of the required reinforcement area for openings. The calculations performed by NOZAR are in accordance with the rules of the ASME Code, Section III, 1974 edition.

3D.4.4 Dynamic Forcing Functions

3D.4.4.1 Relief Valve Discharge Pipe Forces Computer Program--RVFOR

The relief valve discharge pipe connects the pressure-relief valve to the suppression pool. When the valve is opened, the transient fluid flow causes time dependent forces to develop on the pipe wall. This computer program computes the transients fluid mechanics and the resultant pipe forces using the method of characteristics.

3D.4.4.2 Turbine Stop Valve Closure--TSFOR

TSFOR program computes the time-history forcing function in the main steam piping due to turbine stop valve closure. The program utilizes the method of characteristics to compute fluid momentum and pressure loads at each change in pipe section or direction.

3D.4.5 Response Spectra Generation

3D.4.5.1 ERSIN Computer Program

ERSIN is a computer code used to generate response spectra for pipe mounted equipment and for floor mounted equipment. ERSIN provides direct generation of local or global acceleration response spectra.

3D.4.5.2 RINEX Computer Program

RINEX is a computer code used to interpolate and extrapolate amplified response spectra used in the response spectrum method of dynamic analysis. RINEX is also used to generate response spectra with nonconstant model damping. The nonconstant model damping analysis option can calculate spectral acceleration at the discrete eigenvalues of a dynamic system using either the strain energy weighted modal damping or the ASME Code Class N-411-1 damping values.

3D.4.6 Piping Dynamic Analysis Program--PDA

The pipe whip dynamic analysis is performed using the PDA computer program, as described in Subsection 3.6.2.2.2. PDA is a computer program used to determine the response of a pipe subjected to the thrust force occurring after a pipe break. It also is used to determine the pipe whip restraint design and capacity.

The program treats the situation in terms of generic pipe break configuration, which involves a straight, uniform pipe fixed at one end and subjected to a time-dependent thrust force at the other end. A typical restraint used to reduce the resulting deformation is also included at a location between the two ends. Nonlinear and time-independent stress-strain relations are used to model the pipe and the restraint. Using a plastic hinge concept, bending of the pipe is assumed to occur only at the fixed end and at the location supported by the restraint.

Effects of pipe shear deflection are considered negligible. The pipe-bending moment-deflection (or rotation) relation used for these locations is obtained from a static nonlinear cantilever beam analysis. Using moment-angular rotation relations, nonlinear equations of motion are formulated using energy considerations and the equations are numerically integrated in small time steps to yield the time-history of the pipe motion.

3D.4.7 Deleted

3D.4.8 Thermal Transient Program--LION

The LION program is used to compute radial and axial thermal gradients in piping. The program calculates a time-history of ∇T_1, ∇T_2, Ta, and Tb (defined in the ASME Code, Section III, Subsection NB) for uniform and tapered pipe wall thickness.

3D.4.9 Deleted

3D.4.10 Engineering Analysis System--ANSYS

The ANSYS computer program is a large scale general purpose program for the solution of several classes of Engineering Analysis problems. Analysis capabilities include static and dynamic; plastic, creep and swelling; small and large deflections; and other applications.

This program will accommodate a complete model and an enhanced capacities in input, output and graphic interface. Locations of interest for stresses and displacements can be obtained by this nonlinear analysis. It is served as a verification work for the PDA program.

Other program of the same capacities with periodical improvement is also applicable to this analysis.

SECTION 3D.5

CONTENTS

3D.5 PUMPS AND MOTORS

Following are the computer programs used in the dynamic analysis to assure the structural and functional integrity of the pump and motor assemblies, such as those used in the ABWR ECCS systems.

3D.5.1 Structural Analysis Program--SAP4G07

SAP4G07 is used to analyze the structural and functional integrity of the pump/motor systems. This program is also identified in Subsections 4.1.4.1.2 , 3D.3 and 3D.6. This is a general structural analysis program for static and dynamic analysis of linear elastic complex structures. The finite element displacement method is used to solve the displacement and stresses of each element of the structure. The structure can be composed of unlimited number of three-dimensional truss, beam, plate, shell, solid, plane strain-plane stress and spring elements that are axisymmetric. The program can treat thermal and various forms of mechanical loading. The dynamic analysis includes mode superposition, time-history, and response spectrum analysis. Seismic loading and time-dependent pressure can be treated. The program is versatile and efficient in analyzing large and complex structural systems. The output contains displacement of each nodal point as well as stresses at the surface of each element.

3D.5.2 Effects of Flange Joint Connections--FTFLG01

The flange joints connecting the pump bowl casings are analyzed using the FTFLG01 program. This program uses the local forces and moments determined by SAP4G07 to perform flat flange calculations in accordance with the rules set forth in the ASME Code, Section III, Appendices XI and L.

SECTION 3D.6

CONTENTS

3D.6 HEAT EXCHANGERS

The following computer programs are used in dynamic and static analysis to determine structural and functional integrity of the heat exchangers, such as those used in the ABWR RHR system.

3D.6.1 Structural Analysis Program--SAP4G07

The structural integrity of the heat exchanger is evaluated using SAP4G07. This program is described in Subsection 3D.5.1.

3D.6.2 Calculation of Shell Attachment Parameters and Coefficients--BILDR01

BILDR01 is used to calculate the shell attachment parameters and coefficients used in the stress analysis of the support to shell junction. The method per Welding Research Council Bulletin 107 is implemented in BILDR01 to calculate local membrane stress due to the support reaction loads on the heat exchanger shell.

SECTION 3D.7

CONTENTS

3D.7 SOIL-STRUCTURE INTERACTION

3D.7.1 A System For Analysis of Soil-Structure Interaction--SASSI01S

This program consists of a number of interrelated computer program modules which can be used to solve a wide range of dynamic soil-structure interaction (SSI) problems in two or three dimensions. This program is used to obtain enveloped seismic design loads based on the finite element method using substructuring technique, as described in Section 3A.5 of Appendix 3A of this document. A description of this program is included in Subsection 4.1.4.1.9.

The computer program SASSI was developed by the University of California, Berkeley, under the technical direction of Prof. John Lysmer. The Bechtel version of the program was obtained from the University of California, Berkeley, under a license agreement with the University. During the course of installation, testing, and validation of the Bechtel version of the program on the CDC CRAY System, some modifications and enhancements were made to the program to improve the performance. These include correcting the motion phases in Rayleigh wave calculation, replacing the plate element, modifying the spring element to include damping capability, and providing the option for local end release condition in beam element. The CRAY version provided to GE, identified as GE ECP SASSI01S, contains the same modifications and enhancements made to the Bechtel CRAY version to date. The program was verified against benchmark results reported by various investigators in the technical literature.

3D.7.2 Continuum Linear Analysis of Soil-Structure Interaction--CLASSI/ASD

This computer program is used in analyzing limited comparative cases to comply with the dual (finite-element and half-space) soil-structure analysis requirements, as described in Attachment A to Appendix 3A of this document. The program is a linear analysis program using the substructure approach based upon continuum mechanics for half-space.

The program CLASSI is comprised of a series of computer codes developed to calculate the three-dimensional soil-structure interaction response of surface-founded structures using a frequency-dependent continuum impedance approach. The basic version of the CLASSI family of computer programs was developed by Professor J.E. Luco of the University of California at San Diego, and Professor H.L. Wong of the University of Southern California. Additional development effort was contributed by Dr. R.J. Apsel of the University of California at San Diego.

In the CLASSI methodology, the continuum mechanics approach is used to characterize the site-foundation system and the incident seismic waves in terms of complex, frequency-dependent impedance matrices and driving force vectors. The superstructure is represented in terms of its fixed-force vectors. The superstructure is represented in terms of its fixed-base mass matrix, mode shapes, and frequencies, and its modal damping coefficients. These structural dynamic properties can be calculated using any standard finite-element formulation. Compatibility and dynamic equilibrium requirements at the superstructure-foundation interface are then used to determine the three-dimensional response of the complete superstructure-foundation system.

The program CLASSI/ASD is an improved version of the CLASSI family of computer codes, which is developed by ASD International, Inc. This version is verified in accordance with the ASD's Quality Assurance Program and requirements of 10CFR50, Appendix B. Results from the program are verified by benchmark results obtained by various investigators and published in the technical literature.

3D.7.3 Free-Field Response Analysis--SHAKE

This program is used to perform the free-field site response analysis required in the seismic SSI analysis (see Subsection 3A.6).

SHAKE is a computer program developed at the University of California, Berkeley, by Schnable, Lysmer and Seed. (See Reference 5 of Subsection 3A.10). The program uses the principle of one-dimensional propagation of shear waves in the vertical direction for a system of horizontal, visco-elastic soil layers to compute soil responses in the free-field. The nonlinearities in soil shear modulus and damping are accounted for by the use of equivalent linear soil properties using an iterative procedure to obtain values for modulus and damping compatible with the effective shear strains in each layer. The final iterated, strain-compatible properties are used as equivalent linear soil properties in seismic SSI analysis.

APPENDIX 3E

GUIDELINES FOR LBB APPLICATIONS

APPENDIX 3E

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APPENDIX 3E

GUIDELINES FOR LBB APPLICATION

3E.1 INTRODUCTION

As discussed in Subsection 3.6.3, this appendix provides detailed guidelines for the COL applicant's use in applying for NRC's approval of LBB for specific piping systems. Also included in this appendix are the fracture mechanics properties of ABWR piping materials and analysis methods, including the leak rate calculation methods. Table 3E.1-1 gives a list of piping systems inside and outside the containment that are preliminary candidates for LBB application. As noted on Table 3E.1-1, most candidate piping systems are carbon steel piping. Therefore, this appendix deals extensively with the evaluation of carbon steel piping.

Piping qualified by LBB would be excluded from the non-mechanistic postulation requirements of double-ended guillotine break (DEGB) specified in Subsection 3.6.3. The LBB qualification means that the through-wall flaw lengths that are detectable by leakage monitoring systems (see Subsection 5.2.5) are significantly smaller than the flaw lengths that could lead to pipe rupture or instability.

Section 3E.2 addresses the fracture mechanics properties aspects required for evaluation in accordance with Subsection 3.6.3. Section 3E.3 describes the fracture mechanics techniques and methods for the determination of critical flaw lengths and evaluation of flaw stability. Explained in Section 3E.4 is the determination of flaw lengths for detectable leakages with margin. A brief discussion on the leak detection capabilities is presented in Section 3E.5. Finally, Section 3E.6 provides general guidelines for the preparation of LBB justification reports by providing two examples.

Material selection and the deterministic LBB evaluation procedure are discussed in this section.

3E.1.1 Material Selection Guidelines

The LBB approach is applicable to piping systems for which the materials meet the following criteria: (1) low probability of failure from the effects of corrosion (e.g., intergrannular stress corrosion cracking) and (2) adequate margin before susceptibility to cleavage type fracture over the full range of consequences.

The ABWR plant design specifies use of austenitic stainless steel piping made of material (e.g., nuclear grade or low carbon type) that is recognized as resistant to IGSCC. The carbon steel or ferritic steels specified for the reactor pressure boundary are described in 3E.2.2. These steels are assured to have adequate toughness to preclude a fracture at operating temperatures. A COL applicant is expected to supply a detailed justification in the LBB evaluation report considering system temperature, fluid velocity and environmental conditions.

3E.1.2 Deterministic Evaluation Procedure

The following deterministic analysis and evaluation are performed as an NRC-approved method to justify applicability of the LBB concept.

(1) Use the fracture mechanics and the leak rate computational methods that are accepted by the NRC staff, or are demonstrated accurate with respect to other acceptable computational procedures or with experimental data.

(2) Identify the types of materials and materials specifications used for base metal, weldments and safe ends, and provide the materials properties including toughness and tensile data, long-term effects such as thermal aging, and other limitations.

(3) Specify the type and magnitude of the loads applied (forces, bending and torsional moments), their source(s) and method of combination. For each pipe size in the functional system, identify the location(s) which have the least favorable combination of stress and material properties for base metal, weldments and safe ends.

(4) Postulate a throughwall flaw at the location(s) specified in (3) above. The size of the flaw should be large enough so that the leakage is assured detection with sufficient margin using the installed leak detection capability when pipes are subjected to normal operating loads. If auxiliary leak detection systems are relied on, they should be described. For the estimation of leakage, the normal operating loads (i.e., deadweight, thermal expansion, and pressure) are to be combined based on the algebraic sum of individual values.

Using fracture mechanics stability analysis or limit load analysis based on (11) below, and normal plus SSE loads, determine the critical crack size for the postulated throughwall crack. Determine crack size margin by comparing the selected leakage size crack to the critical crack size. Demonstrate that there is a margin of 2 between the leakage and critical crack sizes. The same load combination method selected in (5) below is used to determine the critical crack size.

(5) Determine margin in terms of applied loads by a crack stability analysis. Demonstrate that the leakage size cracks will not experience unstable crack growth if 1.4 times the normal plus SSE loads are applied. Demonstrate that crack growth is stable and the final crack is limited such that a double-ended pipe break will not occur. The dead-weight, thermal expansion, pressure, SSE (inertial), and seismic anchor motion (SAM) loads are combined based on the same method used for the primary stress evaluation by the ASME Code. The SSE (inertial) and SAM loads are combined by square-root-of-the-sum-of-the-squares (SRSS) method.

(6) The piping material toughness (J-R curves) and tensile (stress-strain curves) properties are determined at temperatures near the upper range of normal plant operation.

(7) The specimen used to generate J-R curves is assured large enough to provide crack extensions up to an amount consistent with J/T condition determined by analysis for the application. Because practical specimen size limitations exist, the ability to obtain the desired amount of experimental crack extension may be restricted. In this case, extrapolation techniques is used as described in NUREG-1601, Volume 3, or in NUREG/CR-4575. Other techniques can be used if adequately justified.

(8) The stress-strain curves are obtained over the range from the preoperational limit to maximum load.

(9) Preferably, the materials tests should be conducted using archival materials for the pipe being evaluated. If archival material is not available, plant specific or industry wide generic material data bases are assembled and used to define the required material tensile and toughness properties. Test material includes base and weld metals.

(10) To provide an acceptable level of reliability, generic data bases are reasonable lower bounds for compatible sets of material tensile and toughness properties associated with materials at the plant. To assure that the plant specific generic data base is adequate, a determination is made to demonstrate that the generic data base represents the range of plant materials to be evaluated. This determination is based on a comparison of the plant material propertires identified in (2) above with those of the materials used to develop the generic data base. The number of material heats and weld procedures tested are adequate to cover the strength and toughness range of the actual plant materials. Reasonable lower bound tensile and toughness properties from the plant specific generic data base are to be used for the stability analysis of individual materials, unless otherwise justified.

Industry generic data bases are reviewed to provide a reasonable lower bound for the population of material tensile and toughness properties associated with any individual specification (e.g., A106, Grade B), material type (e.g., austenitic steel) or welding procedures.

The number of material heats and weld procedures tested should be adequate to cover the range of the strength and tensile properties expected for specific material specifications or types. Reasonable lower bound tensile and toughness properties from the industry generic data base are used fro the stability analysis of individual materials.

If the data are being developed from an archival heat of material three stress-strain curves and three J-resistance curves from that one heat of material is sufficient. The tests should be conducted at temperatures near the upper range of normal plant operation. Tests should also be conducted at a lower temperature, which may represent a plant condition (e.g., hot standby) where pipe break would present safety concerns similar to normal operation. These tests are intended only to determine if there is any significant dependence of toughness on temperature over the temperature range of interest. The lower toughness should be used in the fracture mechanics evaluation. One J-R curve and one stress- strain curve for one base metal and weld metal are considered adequate to determine temperature dependence.

(11) There are certain limitations that currently preclude generic use of limit load analyses to evaluate leak-before-break conditions deterministically. However, a modified limit-load analysis can be used for austenitic stainless steel piping to demonstrate acceptable margins as described in Subsection 3E.3.3.

Table 3E.1-1

LEAK BEFORE BREAK CANDIDATE PIPING SYSTEM

System	Location	Description	Diameter (mm)
Main Steam (4 lines)	PC	RPV to RCCV	[illegible]
Feedwater (2 lines/6 risers)	PC	RPV to RCCV	550/300
RCIC Steam	PC	MS line to RCCV	150
HPCF	PC	RPV to first check valve	200
RHR/LPFL	PC	RPV to first check valve	250
RHR/Suction	PC	RPV to first closed gate valve	350
CUW	PC	RHR suction to RCCV	200
Main Steam (4 lines)	Steam Tunnel	RCCV to turbine building	700
Feedwater (2 lines)	Steam Tunnel	RCCV to turbine bulding	550
RHR Div. A Suction	Steam Tunnel	FW line A to check valve	250
RCIC Steam	SC	RCCV to turbine shutoff valve	150
RCIC Supply	SC	FW line to first check valve	200
CUW Suction	SC	RCCV to heat exchanger discharge	200
CUW Discharge	SC	Heat exchanger discharge to FW suction	200/150

Note: All piping in primary and secondary containment (including steam tunnel) are carbon steel piping, except the in-containment CUW piping which is stainless steel.

Legend: PC: Primary Containment
SC: Secondary Containment
FW: Feedwater
MS: Main Steam

SECTION 3E.2

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SECTION 3E.2

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3E.2 MATERIAL FRACTURE TOUGHNESS CHARACTERIZATION

This subsection describes the fracture toughness properties and flow stress evaluation for the ferritic and austenitic steel materials used in ABWR plant piping, as required for evaluation according to Section 3E.1.2.

3E.2.1 Fracture Toughness Characterization

When the elastic-plastic fracture mechanics (EPFM) methodology or the J-T methodology is used to evaluate the leak-before-break conditions with postulated through-wall flaws, the material toughness property is characterized in the form of J-integral resistance curve (or J-R curve) [1, 2, 3]. The J-R curve, schematically shown in Figure 3E.2-1a, represents the material's resistance to crack extension. The onset of crack extension is assumed to occur at a critical value of J. Where the plane strain conditions are satisfied, initiation J is denoted by J_{IC}. Plane strain crack conditions, achieved in test specimen by side grooving, generally provide a lower bound behavior for material resistance to stable crack growth.

Once the crack begins to extend, the increase of J with crack growth is measured in terms of slope or the nondimensional tearing modulus, T, expressed as:

$$T = \frac{E}{\sigma_f^2} \cdot \frac{dJ}{da} \qquad \text{(E.2-1)}$$

The flow stress, σ_f, is a function of the yield and ultimate strength, and E is the elastic modulus. Generally, σ_f is assumed as the average of the yield and ultimate strength. The slope $\frac{dJ}{da}$ of the material J-R curve is a function of crack extension a. Generally, $\frac{dJ}{da}$ decreases with crack extension thereby giving a convex upward appearance to the material J-R curve in Figure 3E.2-1a.

To evaluate the stability of crack growth, it is convenient to represent the material J-R curve in the J-T space as shown in Figure 3E.2-1b. The resulting curve is labeled as J-T material. Crack instability is predicted at the intersection point of the J/T material and J/T applied curves.

The crack growth invariably involves some elastic unloading and distinctly nonproportional plastic deformation near the crack tip. J-integral is based on the deformation theory of plasticity [4, 5] which inadequately models both of these aspects of plastic behavior. In order to use J-integral to characterize crack growth (i.e. to assure J-controlled crack growth), the following sufficiency condition in terms of a nondimensional parameter proposed by Hutchinson and Paris [6], is used:

$$\omega = \frac{b}{J} \cdot \frac{dJ}{da} >> 1 \qquad \text{(E.2-2)}$$

Where b is the remaining ligament. Reference 7 suggests that $\omega > 10$ would satisfy the J-controlled growth requirements. However, if the requirements of this criteria are strictly followed, the amount of crack growth allowed would be very small in most test specimen geometries. Use of such a material J-R curve in J/T evaluation would result in grossly underpredicting the instability loads for large-diameter pipes where considerable stable crack growth is expected to occur before reaching the instability point. To overcome this difficulty, Ernst [8] proposed a modified J-integral, J_{mod}, which was shown to be effective even when limits on 1 were grossly violated. The Ernst correction essentially factors-in the effect of crack extension in the calculated value of J. This correction can be determined experimentally by measuring the usual parameters: load, displacement and crack length.

The definition of J_{mod} is:

$$J_{mod} = J - \int_{a_o}^{a} \left(\frac{\partial (J-G)}{\partial a}\right)_{\delta_{pl}} da \qquad \text{(E.2-3)}$$

Where

- J is based on deformation theory of plasticity
- G is the linear elastic Griffith energy release rate or elastic J, J_{el}.
- δ_{pl} is the nonlinear part of the load-point displacement. (or simply the total minus the elastic

displacement).

a_o, a are the initial and current crack lengths respectively.

For the particular case of the compact tension specimen geometry, the preceding Equation and the corresponding rate take the form

$$J_{mod} = J + \int_{a_o}^{a} \gamma \cdot \frac{J_{pl}}{b} \cdot da \qquad (E.2\text{-}4)$$

where J_{pl} is the nonlinear part of the deformation theory J, b is the remaining ligament and γ is

$$\gamma = (1+0.76\, b/W) \qquad (E.2\text{-}5)$$

Consequently the modified material tearing modulus T_{mod} can be defined as:

$$T_{mod} = T_{mat} + \frac{E}{\sigma_f^2} \frac{\gamma}{b} \cdot J_{pl} \qquad (E.2\text{-}6)$$

Since in most of the test J-R curves the $\omega > 10$ limit was violated, all of the material J-T data were recalculated in the J_{mod}, T_{mod} format. The J_{mod}, T_{mod} calculations were performed up to crack extension $\Delta a = 10\%$ of the original ligament in the test specimen. The J-T curves were then extrapolated to larger J values using the method recommended in NUREG 1061, Vol. 3 [9].

The J_{mod} - T_{mod} approach is used in this appendix for illustrative purposes. It should be adopted if justified based on its acceptability by the technical literature. A J_D - approach is another more justifiable approach.

3E.2.2 Carbon Steels and Associated Welds

The carbon steels used in the ABWR reactor coolant pressure boundary piping are: SA 106 Gr B, SA 333 Gr. 6 and SA 672, Gr. C70. The first specification covers seamless pipe and the second one pertains to both seamless and seam-welded pipe. The last one pertains to seam-welded pipe for which plate stock is specified as SA 516. Gr. 70. The corresponding material specifications used for carbon steel flanges, fittings and forgings are equivalent to the piping specifications.

While the chemical composition requirements for a pipe per SA 106 Gr. B and SA 333 Gr. 6 are identical, the latter is subjected to two additional requirements: (1) a normalizing heat treatment which refines the grain structure and, (2) a charpy test at -50°F with a specified minimum absorbed energy of 13 ft-lbs. The electrodes and filler metal requirements for welding carbon steel to carbon or low alloy steel are as specified in Table 3E.2-1.

A comprehensive test program was undertaken at GE to characterize the carbon steel base and weld material tougheness properties. The next section describes the scope and the results of this program. The purpose of the test program was two generate the necessary data for application in Section 3E.6 and to illustrate a general procedure of conducting the tests per requirements of Item (10) in Section 3E.1.2. The extent of the test program for NRC's approval of an application will depend upon the identified requirements.

3E.2.2.1 Fracture Toughness Test Program

The test program consisted of generating true stress-true strain curves, J-Resistance curves and the charpy V-notch tests. Two materials were selected : (1) SA333 Gr. 6, 16-inch diameter, Schedule 80 pipe and (2) SA516, Gr. 70, 1-inch thick plate. Table 3E.2-2 shows the chemical composition and mechanical property test information provided by the material supplier. The materials were purchased to the same specifications as those to be used in the ABWR applications.

To produce a circumferential butt weld, the pipe was cut in two pieces along a circumferential plane and welded back using the shielded metal arc process. The weld prep was of single V design with a backing ring. The preheat temperature was 200°F.

The plate material was cut along the longitudinal axis and welded back using the SAW process. The weld prep was of a single V type with one side as vertical and the other side at 45°. A backing plate was used during the welding with a clearance of 1/4 inch at the bottom of the V. The interpass temperature was maintained at less than 500^{o}F.

Both the plate and the pipe welds were X-rayed according to Code [11] requirements and were found to be satisfactory.

It is well-known that carbon steel base materials show considerable anisotropy in fracture toughness properties. The toughness depends on the orientation and direction of propagation of the crack in relation to the principal direction of mechanical working or gain flow. Thus, the selection of proper orientation of charpy and J-R curve test specimen is important. Figure 3E.2-2 shows the orientation code for rolled plate and pipe specimen as given in ASTM Standard E399 [12]. Since a through-wall circumferential crack configuration is of most interest from the DEGB point of view, the L-T specimen in a plate and the L-C specimen in a pipe provide the appropriate toughness properties for that case. On the other hand, T-L and C-L specimen are appropriate for the axial flaw case.

Charpy test data are reviewed first since they provide a qualitative measure of the fracture toughness.

3E.2.2.1.1 Charpy Tests

The absorbed energy or its complement, the lateral expansion measured during a Charpy V-notch test provides a qualitative measure of the material toughness. For example, in the case of austenitic stainless steel flux weldments, the observed lower Charpy energy relative to the base metal was consistent with the similar trend observed in the J-Resistance curves. The Charpy tests in this program were used as preliminary indicators of relative toughness of welds, HAZs and the base metal.

The carbon steel base materials exhibit considerable anisotropy in the Charpy energy as illustrated by Figure 3E.2-3 from Reference 13. This anisotropy is associated with development of grain flow due to mechanical working. The Charpy orientation C in Figure 3E.2-3 (orientations LC and LT in Figure 3E.2-2) is the appropriate one for evaluating the fracture resistance to the extension of a through-wall circumferential flaw. The upper shelf Charpy energy associated with axial flaw extension (orientation A in Figure 3E.2-3) is considerably lower than that for the circumferential crack extension.

A similar trend in the base metal charpy energies was also noted in this test program. Figures 3E.2-4a and b show the pipe and plate material Charpy energies for the two orientations as a function of temperature. The tests were conducted at six temperatures ranging from room temperature to 550^{o}F. From the trend of the Charpy energies as a function of temperature in Figures 3E.2-4a and b it is clear that even at room temperature the upper shelf conditions have been reached for both the materials.

No such anisotropy is expected in the weld metal since it does not undergo any mechanical working after its deposition. This conclusion is also supported by the available data in the technical literature. The weld metal charpy specimen in this test program were oriented the same way as the LC or LT orientations in Figure 3E.2-2. The HAZ charpy specimens were also oriented similarly.

Figure 3E.2-5 shows a comparison of the charpy energies from the 333 Gr. 6 base metal, the weld metal and the HAZ. In most cases two specimens were used. Considerable scatter in the weld and HAZ charpy energy values is seen. Nevertheless, the average energies fro the weld metal and the HAZ seem to fall at or above the average base metal values. This indicates that, unlike the stainless steel flux weldments, the fracture toughness of carbon steel weld and HAZ, as measured by the charpy tests, is at least equal to the carbon steel base metal.

The preceding results and the results of the stress-strain tests discussed in the next section or other similar data are used as a basis to choose between the base and the weld metal properties for use in the J-T methodology evaluation.

3E.2.2.1.2 Stress-Strain Tests

The stress-strain tests were performed at three temperatures: Room temperature, 350^{o}F and 550^{o}F. Base and weld metal from both the pipe and the plate were tested. The weld specimens were in the as-welded condition. The standard test data obtained from these tests are summarized in Table 3E.2-3.

An examination of Table 3E.2-3 shows that the measured yield strength of the weld metal, as expected, is considerably higher than that of the base metal. For example, the 550°F yield strength of the weld metal in Table 3E.2-3 ranges from 53 to 59 ksi, whereas the base metal yield strength is only 34 ksi. The impact of this observation in the selection of appropriate material (J/T) curve is discussed in later sections.

Figures 3E.2-6 a through d show the plots of the 550°F and 350°F stress-strain curves for both the pipe and the plate used in the test. As expected, the weld metal stress-strain curve in every case is higher than the corresponding base metal curve. The Ramberg-Osgood format characterization of these stress-strain curves is given in Section 3E.3.2 where appropriate values of and is also provided.

3E.2.2.1.3 J-R Curve Tests

The test temperatures selected for the J-R curve tests were: room temperature, 350°F and 550°F. Both the weld and the base metal were included. Due to the curvature, only the 1T plan compact tension (CT) specimens were obtained from the 16 inch diameter test pipe. Both 1T and 2T plan test specimens were prepared from the test plate. All of the CT specimens were side-grooved to produce plane strain conditions.

Table 3E.2-4 shows some details of the J-R curve tests performed in this test program. The J-R curve in the LC orientation of the pipe base metal and in the LT orientation of the plate base metal represent the material's resistance to crack extension in the circumferential direction. Thus, the test results of these orientations were used in the LBB evaluations. The orientation effects are not present in the weld metal. As an example of the J-R curve obtained in the test program, Figure 3E.2-7 shows the plot of J-R curve obtained from specimen OWLC-A.

3E.2.2.2 Material (J/T) Curve Selection

The normal operating temperatures for most of the carbon steel piping in the reactor coolant pressure boundary in the ABWR generally fall into two categories: 528-550°F and 420°F. The latter temperature corresponds to the operating temperature of the feedwater piping system. The selections of the appropriate material (J/T) curves for these two categories are discussed next.

3E.2.2.2.1 Material J/T Curve for 550°F

A review shows that 5 tests were conducted at 550°F. Two tests were on the weld metal, two were on the base metal and one was on the heat-affected zone. Figure 3E.2-8 shows the plot of material J_{mod}, T_{mod} values calculated from the J- Δa values obtained from the 550°F tests. The value of flow stress, σ_f, used in the tearing modulus calculation (Equation E.2-1) was 52.0 ksi based on data shown in Table 3E.2-3. To convert the deformation J and $\frac{dJ}{da}$ values obtained from the J-R into J_{mod}, T_{mod}, Equations E.2-4 and E.2-6 were used. Only the data from the pipe-weld (Specimen ID OWLC-A) and the plate base metal (Specimen ID BMLI-12) are shown in Figure 3E.2-8. A few unreliable data points were obtained in the pipe base metal (Specimen ID OBLC-2) J-R curve test due to a malfunction in the instrumentation. Therefore, the data from this test were not included in the evaluation. The J-R curves from the other two 550°F tests were evaluated as described in the next paragraph. For comparison purposes, Figure 3E.2-8 also shows the SA106 carbon steel J-T data obtained from the J-R curve reported by Gudas [14]. The curve also includes extrapolation to higher J values based on the method recommended in NUREG 1061, Vol. 3[9].

The J_{mod}-T_{mod} data for the plate weld metal and the plate HAZ were evaluated. A comparison shows that these data fall slightly below those for the plate base metal shown in Figure 3E.2-8. On the other hand, as noted in Subsection 3E.2.2.1.2, the yield strength of the weld metal and the HAZ is considerably higher than that of the base metal. The material stress-strain and J-T curves are the two key inputs in determining the instability load and flaw values by the (J/T) methodology. Calculations performed for representative through-wall flaw sizes showed that the higher yield strength of the weld metal more than compensates for the slightly lower J-R curve and, consequently, the instability load and flaw predictions based on base metal properties are smaller (i.e., conservative). Accordingly, it was concluded that the material (J-T) curve shown in Figure 3E.2-8 is the appropriate one to use in the LBB evaluations for carbon steel piping at 550°F.

3E.2.2.2.2 Material J/T Curve For 420°F

Since the test temperature of 350°F can be considered reasonably close to the 420°F, the test J-R curves for 350°F were used in this case. A review of the test matrix in Table 3E.2-4 shows that three tests were conducted at 350°F. The J_{mod}, T_{mod} data for all three tests were reviewed. The flow stress value used in the tearing modulus calculation was 54 ksi based on Table 3E.2-3. Also reviewed were the data on SA106 carbon steel at 300°F reported by Gudas [14].

Consistent with the trend of the 550°F data, the 350°F weld metal (J-T) data fell below the plate and pipe base metal data. This probably reflects the slightly lower toughness of the SAW weld in the plate. The (J/T) data for the pipe base metal fell between the plate base metal and the plate weld metal. Based on the considerations similar to those presented in the previous section, the pipe base metal J-T data, although they may lie above the weld J-T data, were used for selecting the appropriate (J-T) curve. Accordingly, the curve shown in Figure 3E.2-9 was developed for using the (J-T) methodology in evaluations at 420°F.

3E.2.3 Stainless Steels and Associated Welds

The stainless steels used in the ABWR reactor coolant pressure boundary piping are either Nuclear grade or low carbon Type 304 or 316. These materials and the associated welds are highly ductile and therefore, undergo considerable plastic deformation before failure can occur. Toughness properties of Type 304 and 316 stainless steels have been extensively reported in the open technical literature and are, thus, not discussed in detail in this section. Due to high ductility and toughness, modified limit load methods can be used to determine critical crack lengths and instability loads (see Subsection 3E.3.3).

3E.2.4 References

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Analysis, ASTM STP 803, C.F. Shih and J.P. Gudas. Eds., American Society for Testing and Materials, 1983, pp. I-191-I-213.

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10. Deleted

11. ASME Boiler & Pressure Vessel Code, Section III, Division 1, Nuclear Power Plant Components, American Society of Mechanical Engineers, 1980.

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13. Reynolds, M.B., *Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws,* General Electric Report No. GEAP-5620, April 1968.

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TABLE 3E.2-1

ELECTRODES AND FILLER METAL REQUIREMENTS FOR CARBON STEEL WELDS

Base Material	P-No.	Process	Electrode or Specification	Filler Metal Classification
Carbon Steel to	P-1 to	SMAW	SFA 5.1	E7018
Carbon Steel or	P-1, P-3			
Low Alloy Steel	P-4 or P-5	GTAW PAW	SFA 5.18	E70S-2, E70S-3
		GMAW	SFA 5.18 SFA 5.20	E70S-2,E70S-3,E70S-6 E70T-1
		SAW	SFA 5.17	F72EM12K, F72EL12

TABLE 3E.2-2

SUPPLIER PROVIDED CHEMICAL COMPOSITION AND MECHANICAL PROPERTIES INFORMATION

Material	Product Form	Chemical Composition				Mech. Property			
		C	Mn	P	S	Si	Sy(ksi)	Su(ksi)	Elongation (%)
SA 333 Gr.6 Heat #52339	16 In. Sch.80 Pipe	0.12	1.18	.01	.026	0.27	44.0	67.5	42.0
SA 516 Gr.70 Heat #E18767	1.0 In. Plate	0.18	0.98	0.017	0.0022	0.25	46.5	70.5	31.0

Note:

(1) Pipe was normalized at 1650°F. Held for 2 hrs. and air cooled.

(2) Plate was normalized at 1700°F for one hour and still air cooled.

TABLE 3E.2-3

STANDARD TENSION TEST DATA AT TEMPERATURE

SPEC. NO.	MATERIAL TEMP	TEST (ksi)	0.2% YS (ksi)	UTS (%)	Elong. %	RA
OW1	PIPE WELD	RT	66.1	81.6	32	77.2
OW2	PIPE WELD	550F	59.0	93.9	24	56.7
ITWL2	PLATE WELD	550F	53.0	91.4	34	51.3
IBL1	PLATE BASE	RT	44.9	73.7	38	51.3
IBL2	PLATE BASE	350F	37.9	64.2	34	68.9
IBL3	PLATE BASE	550F	34.1	69.9	29	59.4
OB1	PIPE BASE	RT	43.6	68.6	41	67.8
OB2	PIPE BASE	350F	42.2	74.9	21	55.4
OB3	PIPE BASE	550F	34.6	78.2	31	55.4

TABLE 3E.2-4

SUMMARY OF CARBON STEEL J-R CURVE TESTS

No.	Specimen ID	Size	Description	Temp.
(1)	OWLC-A	1T	Pipe Weld	550°F
(2)	OBCL-1	1T	Pipe Base C-L Orientation	RT
(3)	OBLC2	1T	Pipe Base L-C Orientation	550°F
(4)	OBLC3-B	1T	Pipe Base L-C Orientation	350°F
(5)	BML-4	1T	Plate Base Metal, L-T Orientation	RT
(6)	BML4-14	2T	Plate Base Metal, L-T Orientation	RT
(7)	BML2-6	2T	Plate Base Metal, L-T Orientation	350°F
(8)	BML1-12	2T	Plate Base Metal, L-T Orientation	550°F
(9)	WM3-9	2T	Plate Weld Metal	RT
(10)	XWM1-11	2T	Plate Weld Metal	350°F
(11)	WM2-5	2T	Plate Weld Metal	550°F
(12)	HAZ	(Non-standard) Width = 2.793"	Heat-Affected Zone, Plate	RT
(13)	O[illegible]C-7	1T	Pipe Weld	RT

Notes:

1. Pipe base metal, SA333 Gr.6
2. Plate base metal, SA516 Gr.70
3. Pipe weld made by shielded metal arc welding.
4. Plate weld made by submerged arc welding.

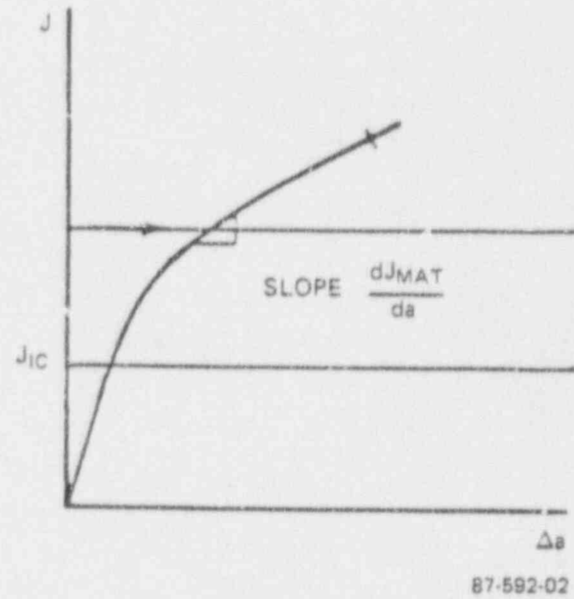


Figure 3E.2-1a SCHEMATIC REPRESENTATION OF MATERIAL J-INTEGRAL R CURVE

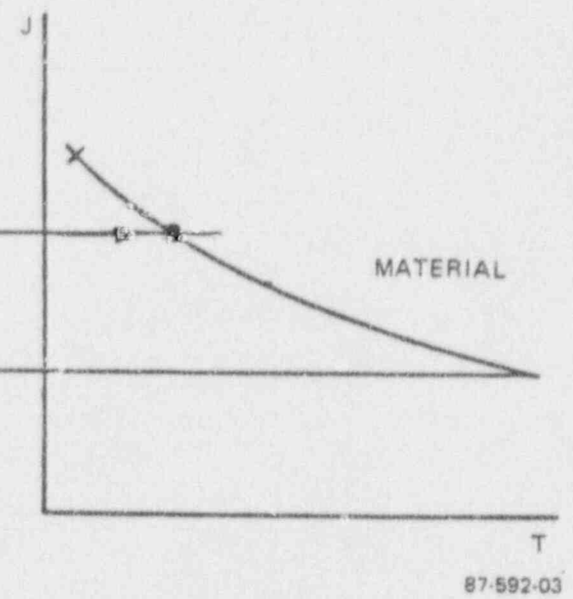
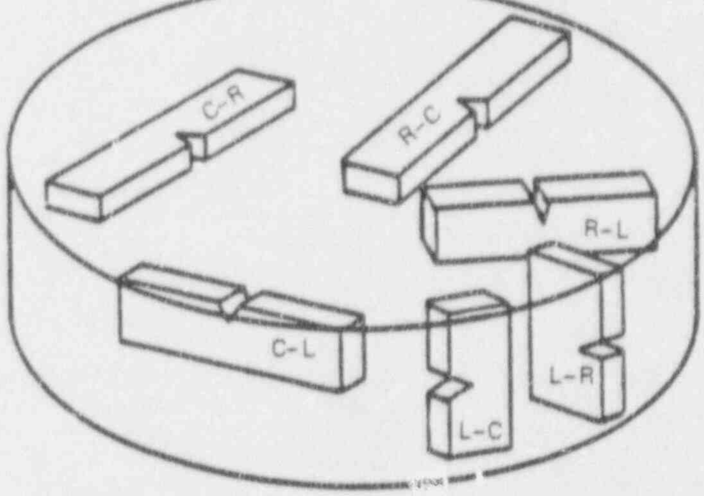
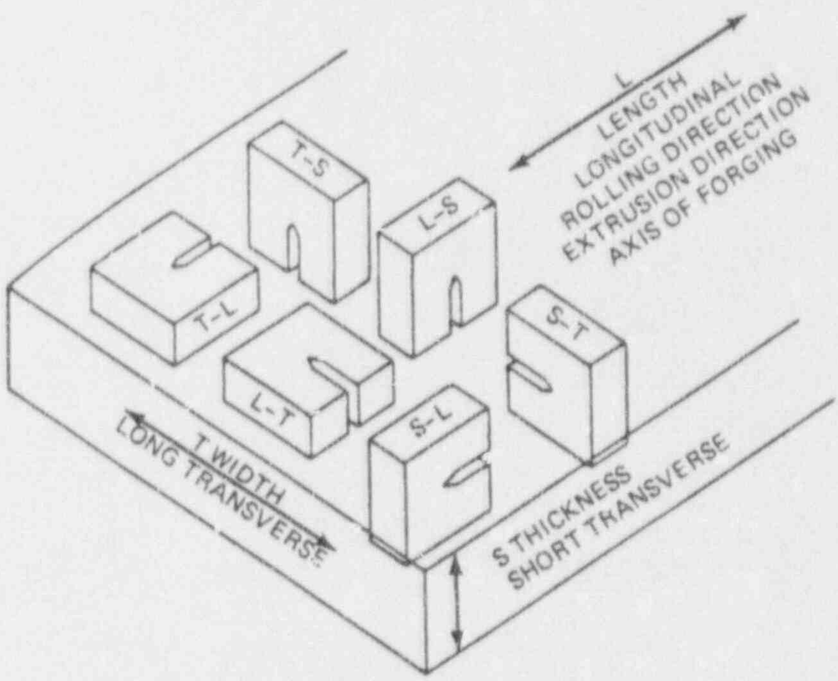


Figure 3E.2-1b SCHEMATIC REPRESENTATION OF MATERIAL J-T CURVE



CRACK PLANE ORIENTATION CODE FOR BAR AND HOLLOW CYLINDER



CRACK PLANE ORIENTATION CODE FOR RECTANGULAR SECTIONS

87-592-04

Figure 3E.2-2 CARBON STEEL TEST SPECIMEN ORIENTATION CODE

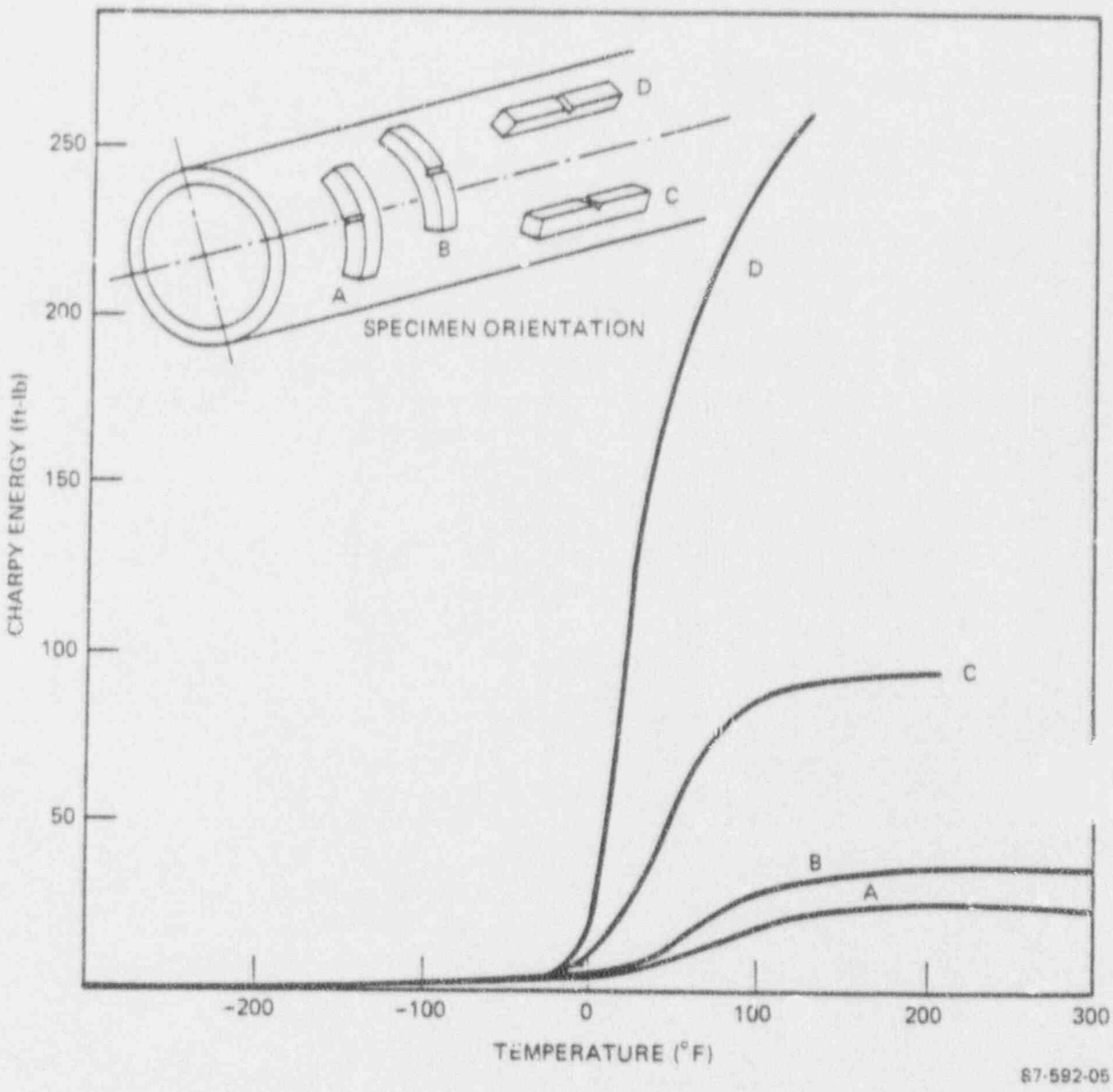


Figure 3E.2-3 TOUGHNESS ANISOTROPY OF ASTM 106 PIPE (6 in. Sch. 80)

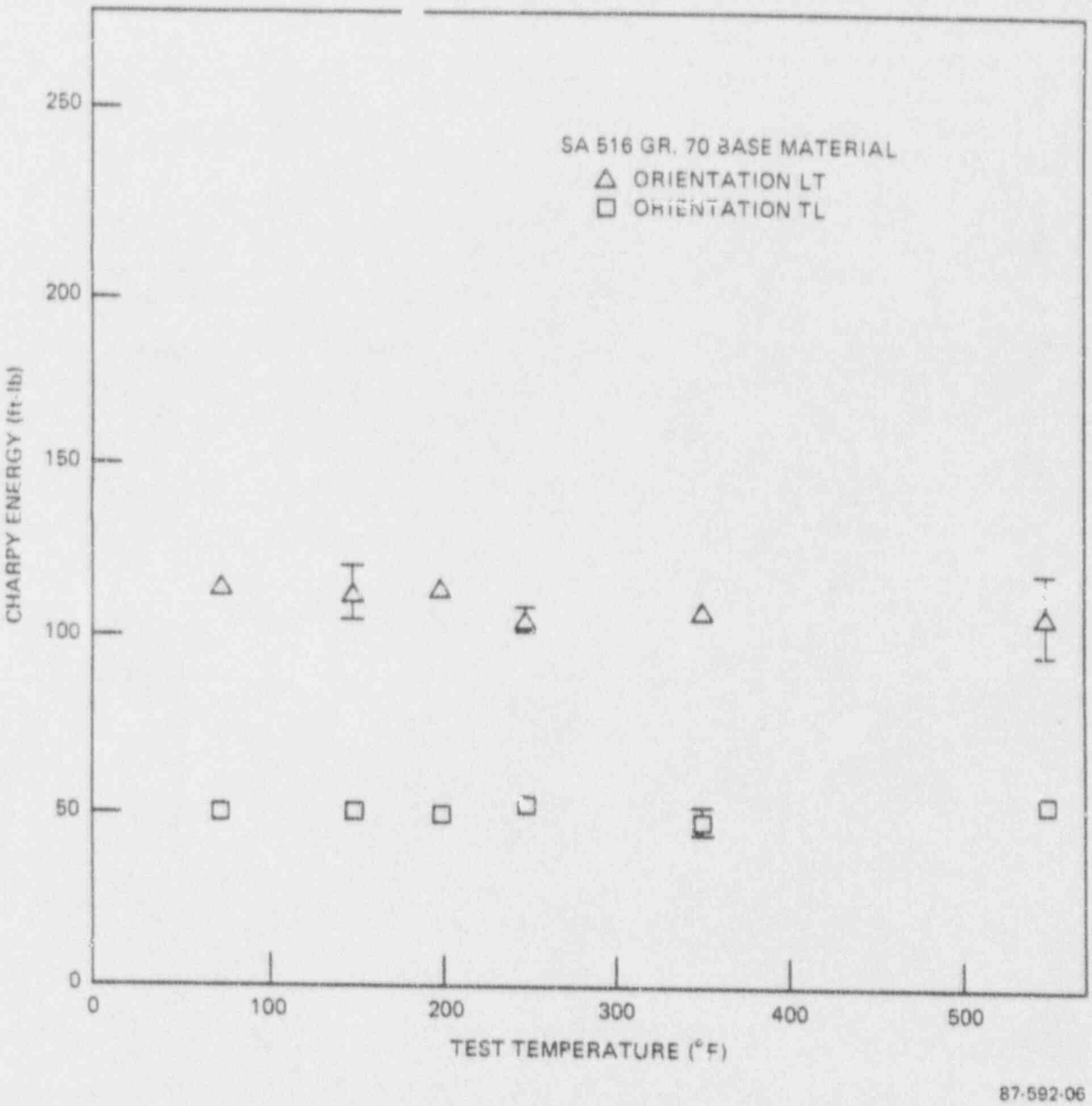


Figure 3E.2-4a CHARPY ENERGIES FOR PIPE TEST MATERIAL AS A FUNCTION OF ORIENTATION AND TEMPERATURE

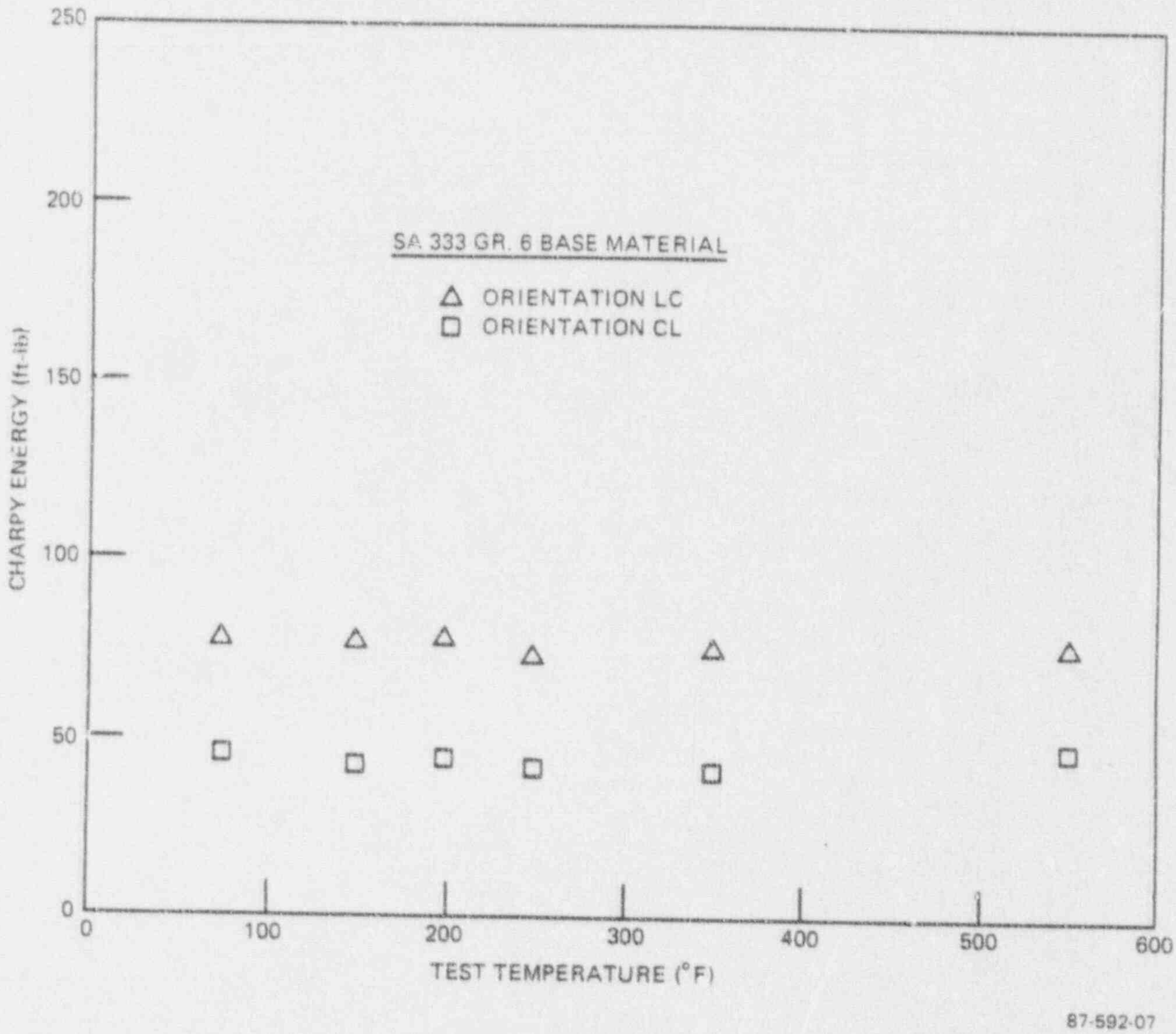


Figure 3E.2-4b CHARPY ENERGIES FOR PLATE TEST MATERIAL AS A FUNCTION OF ORIENTATION AND TEMPERATURE

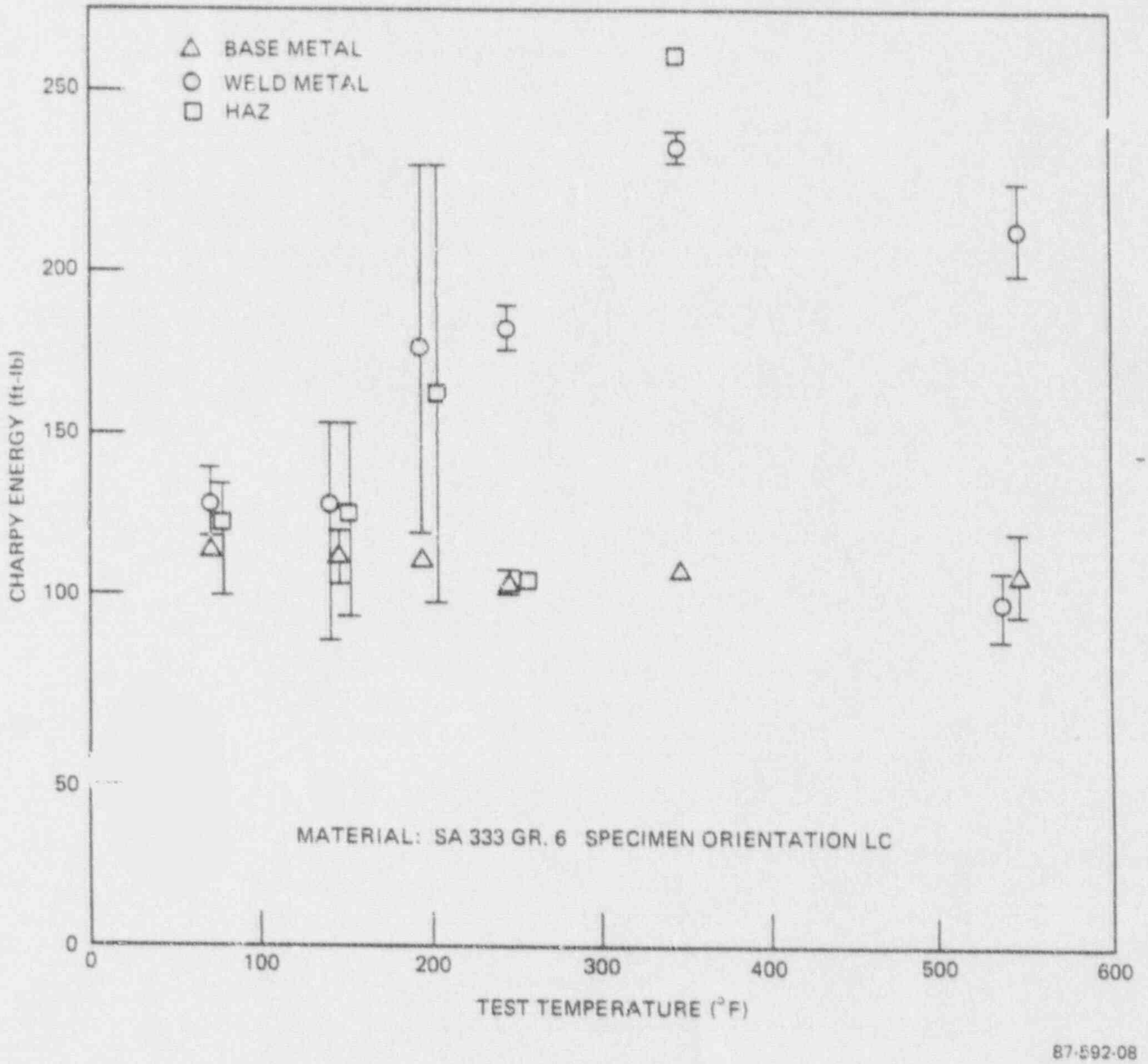


Figure 3E.2-5 COMPARISON OF BASE METAL, WELD AND HAZ CHARPY ENERGIES FOR SA 333 GR. 6

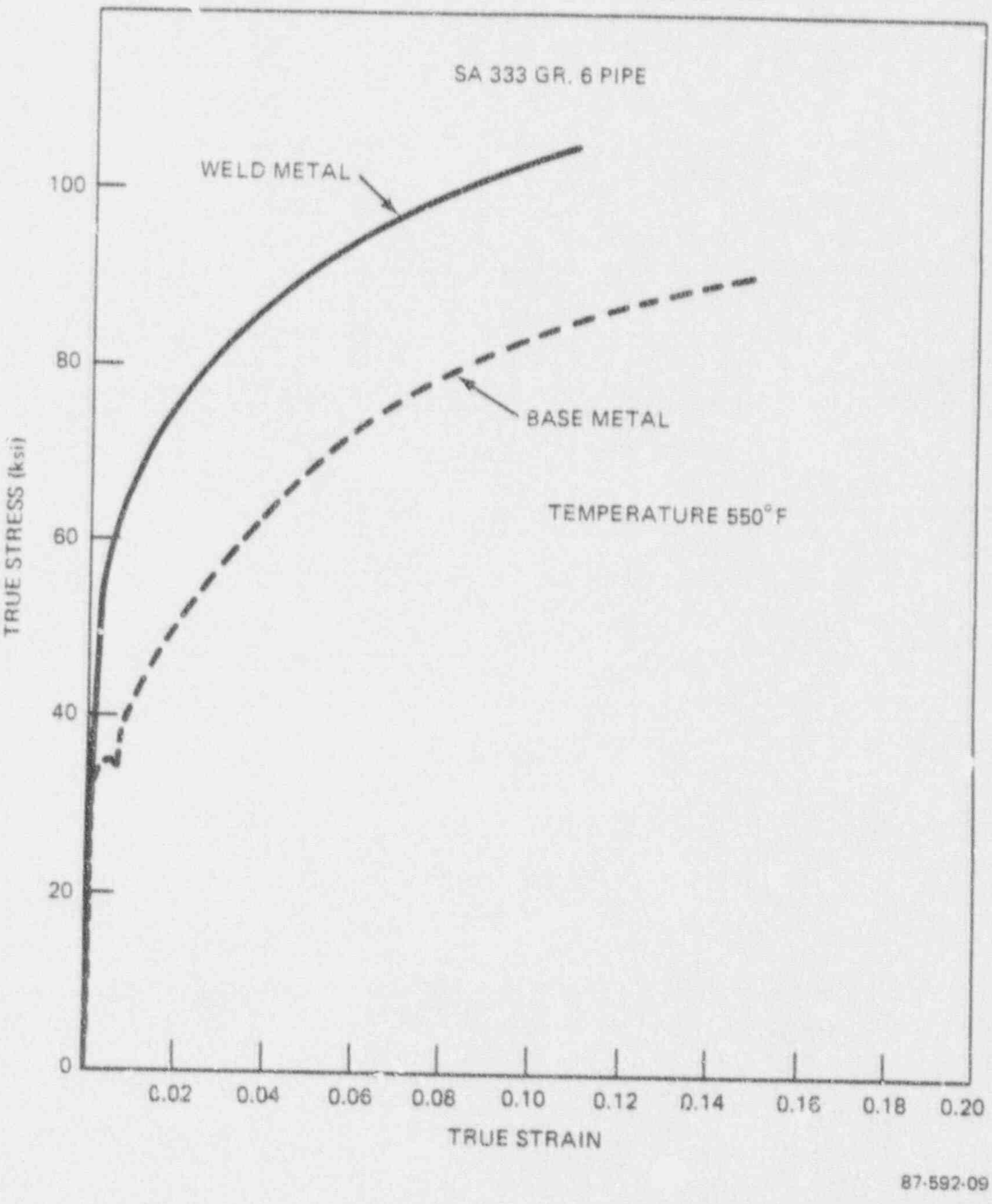


Figure 3E.2-6a PLOT OF 550°F TRUE STRESS-TRUE STRAIN CURVES FOR SA 333 GR. 6 CARBON STEEL

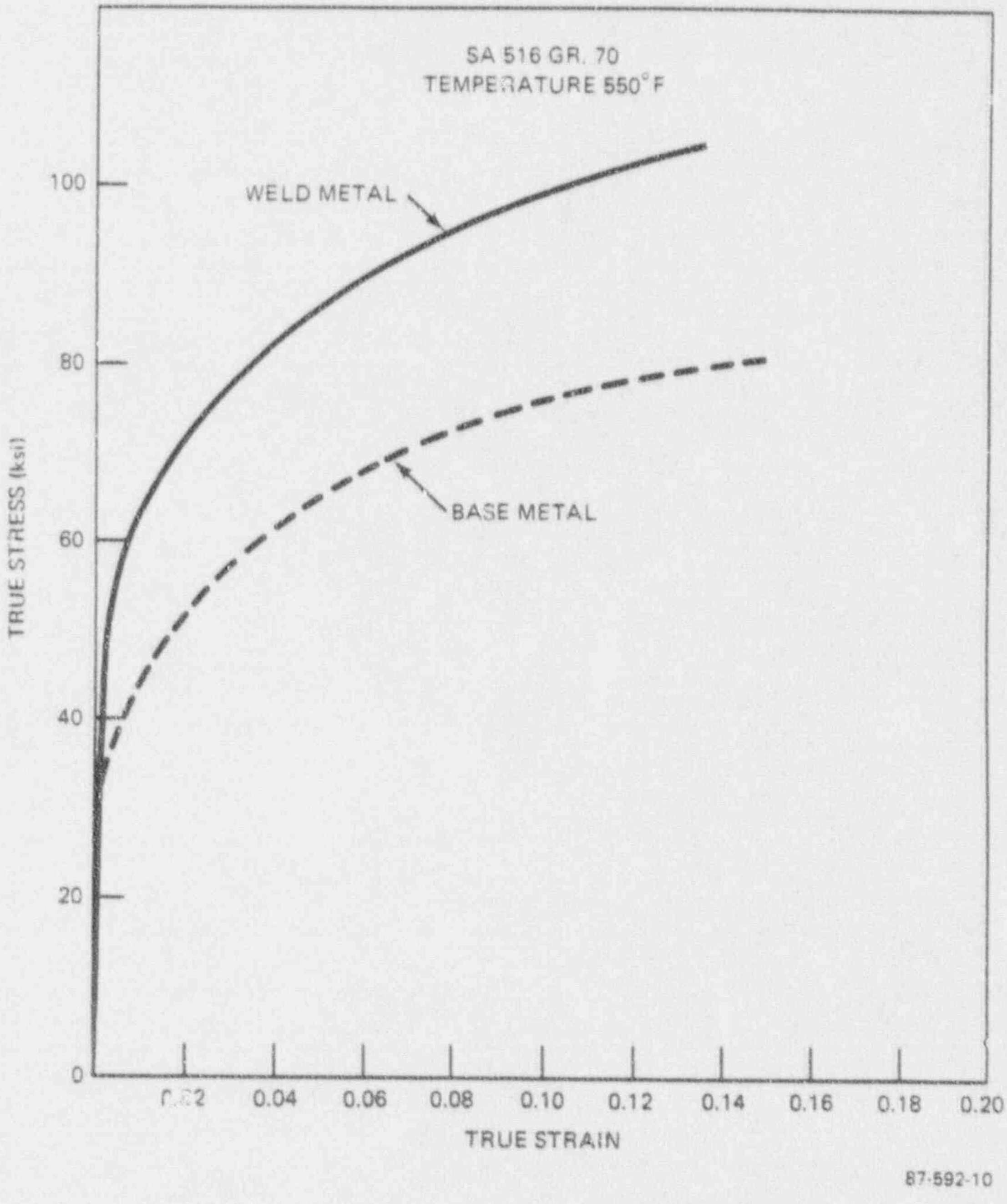


Figure 3E.2-6b PLOT OF 550°F TRUE STRESS–TRUE STRAIN CURVES FOR SA 516 GR. 70 CARBON STEEL

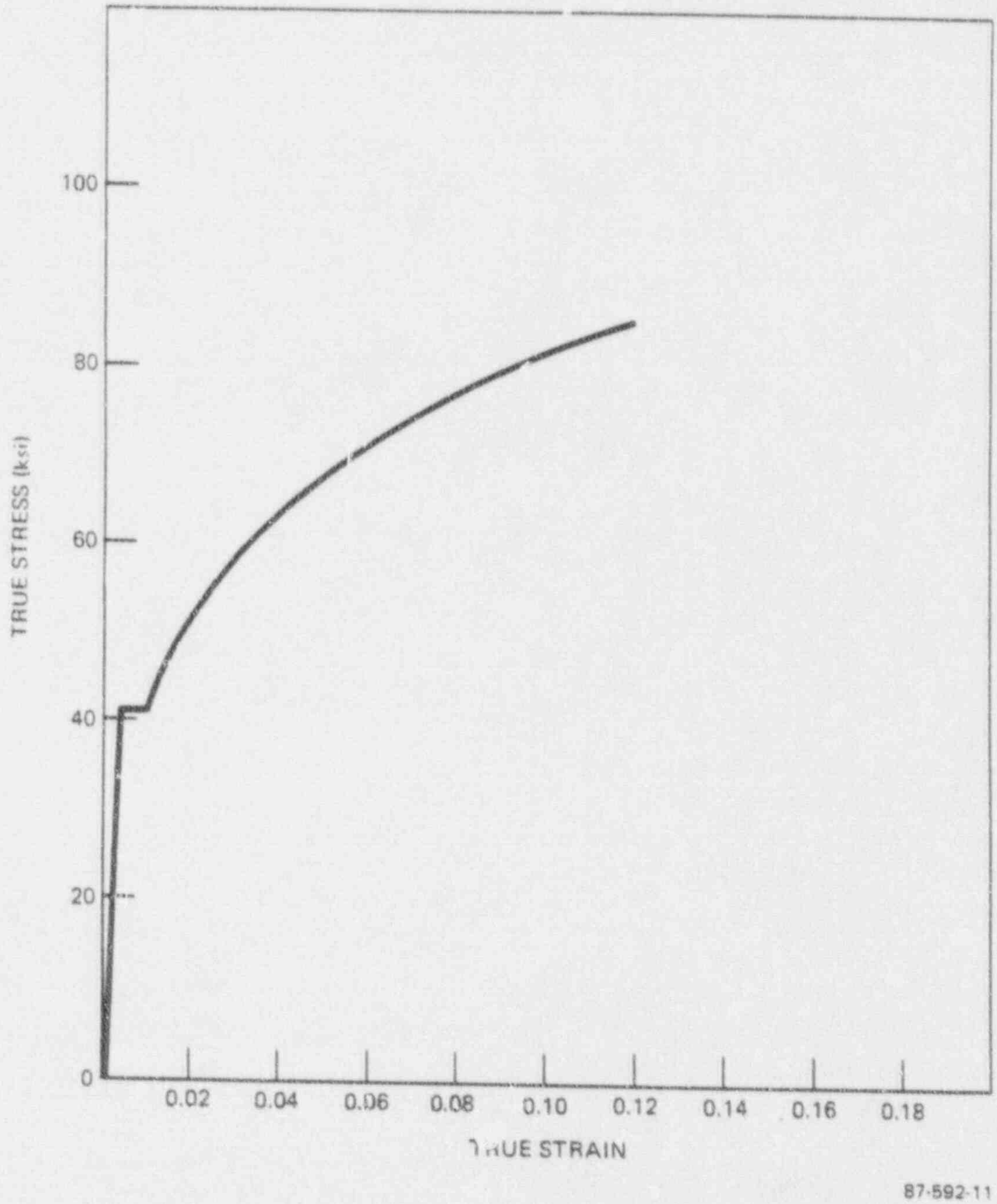


Figure 3E.2-6c PLOT OF 350°F TRUE STRESS-TRUE STRAIN CURVES FOR SA 333 GR. 6 CARBON STEEL

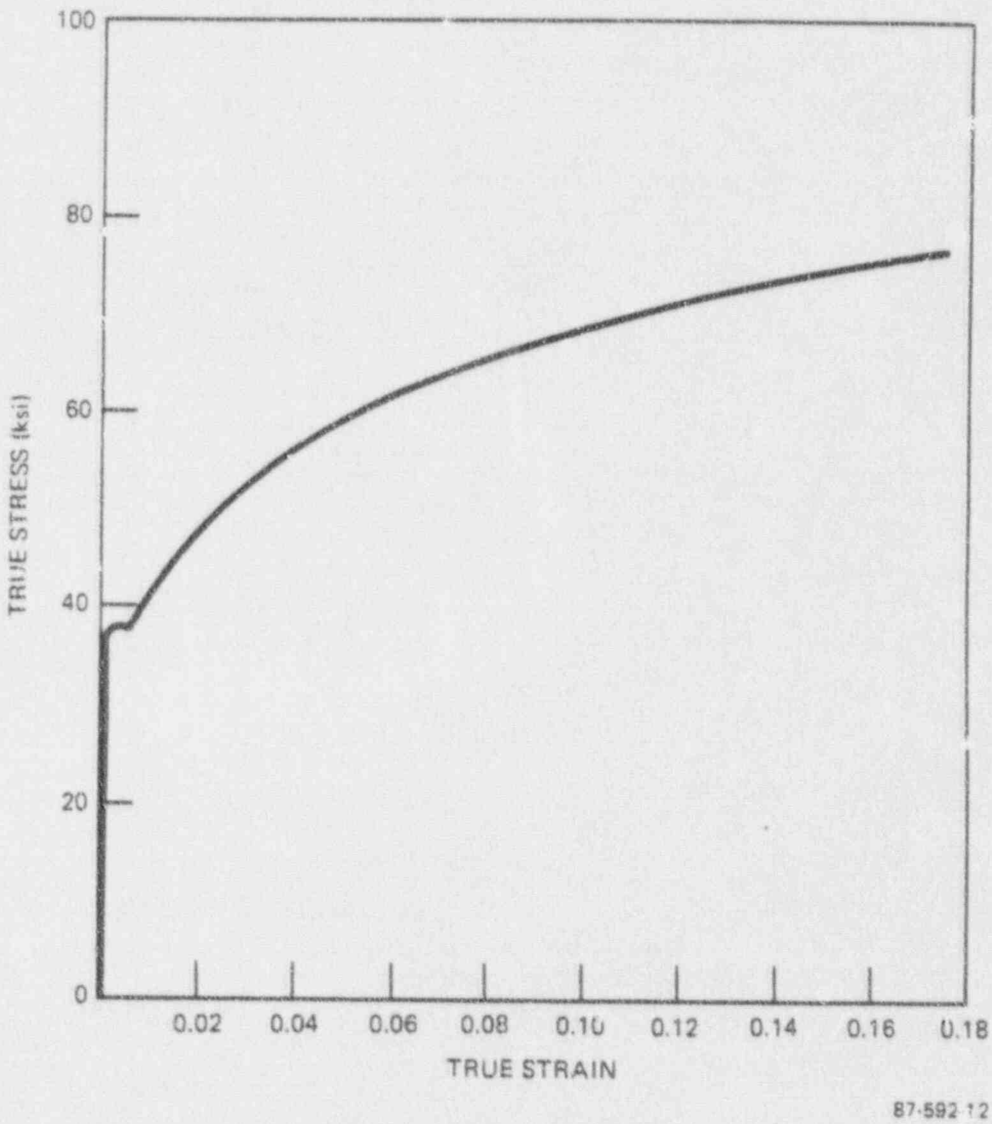


Figure 3E.2-6d PLOT OF 350°F TRUE STRESS-TRUE STRAIN CURVES FOR SA 516 GR. 70 CARBON STEEL

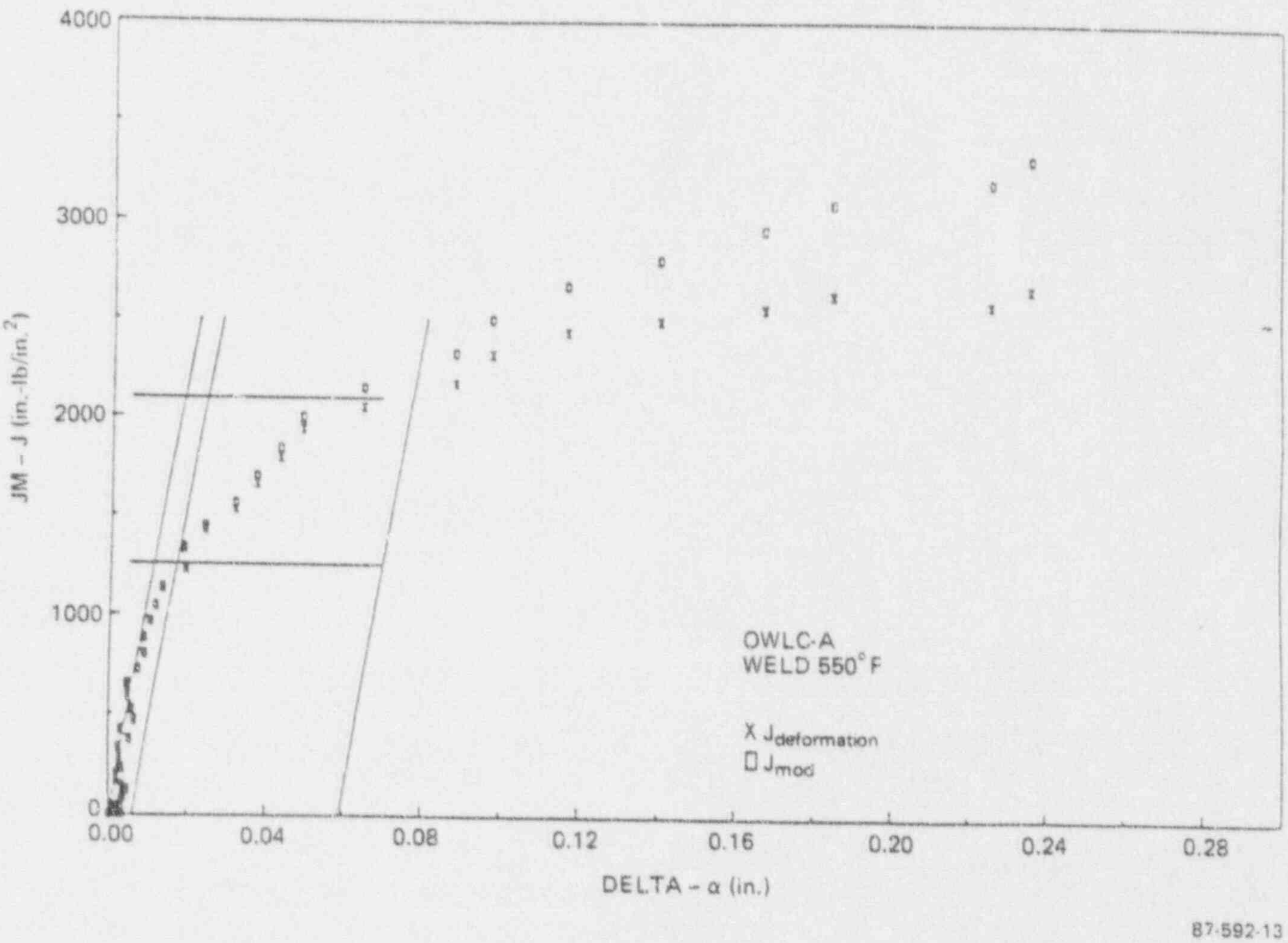


Figure 3E.2-7 PLOT OF 550° F TEST J–R CURVE FOR PIPE WELD

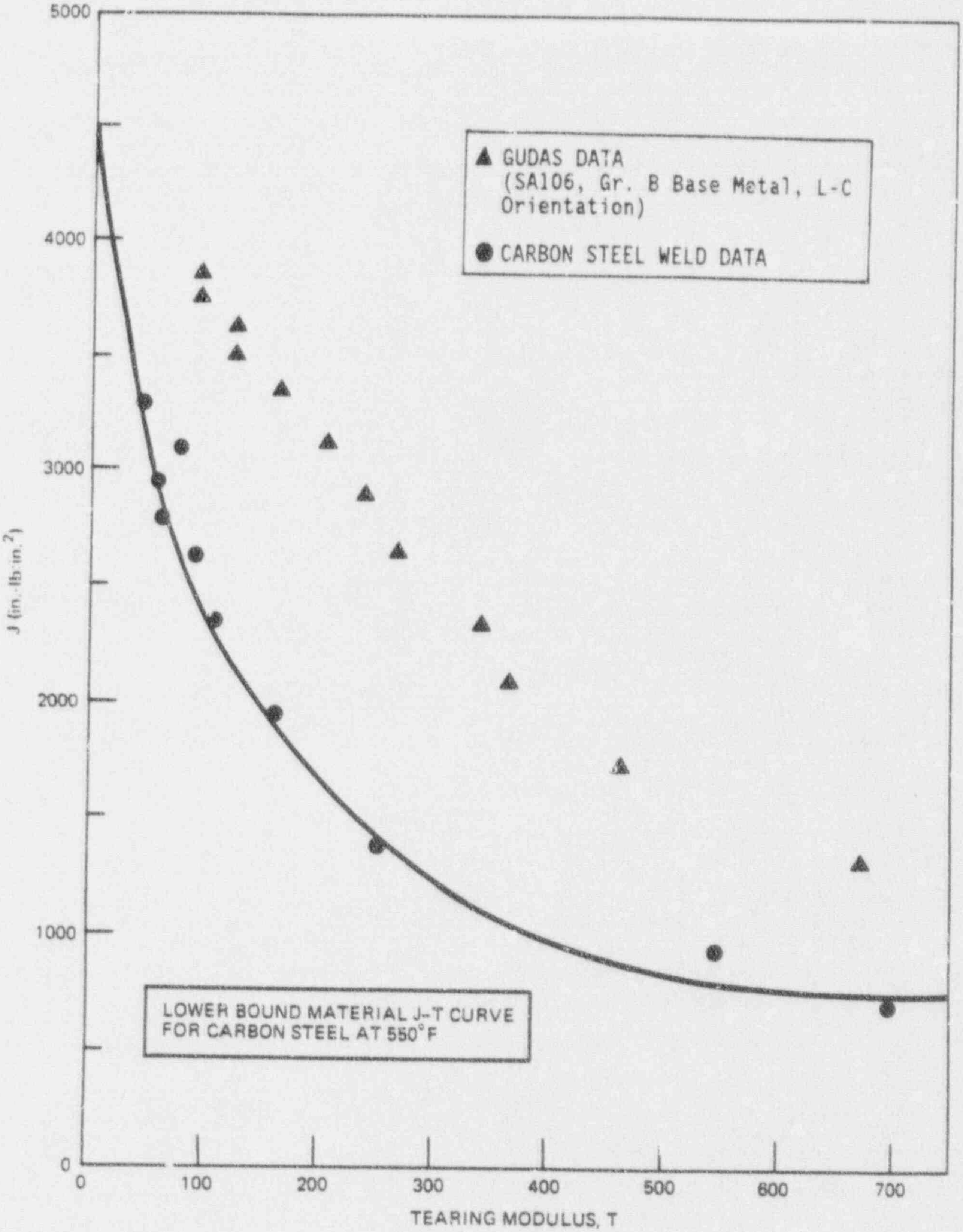


Figure 3E.2-8 PLOT OF 550°F J_{mod}, T_{mod} DATA FROM TEST J-R CURVE

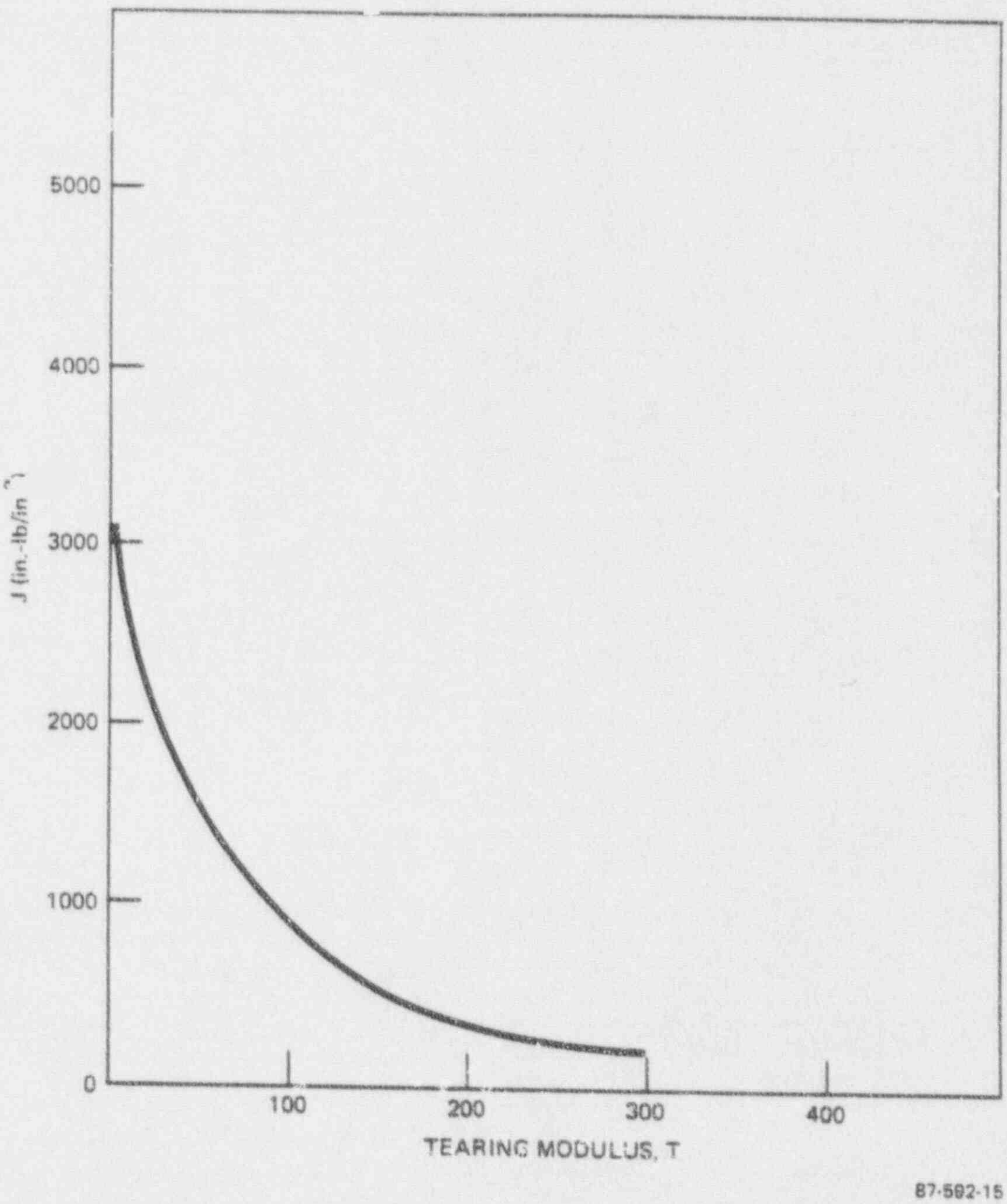


Figure 3E.2-9 CARBON STEEL J-T CURVE FOR 420°F

SECTION 3E.3

CONTENTS

SECTION 3E.3

ILLUSTRATIONS

3E.3 FRACTURE MECHANICS METHODS

This subsection deals with the fracture mechanics techniques and methods for the determination of critical flaw lengths and instability loads for materials used in ABWR. These techniques and methods comply with Criteria (5) through (11) described in Section 3E.1.2.

3E.3.1 Elastic-Plastic Fracture Mechanics or (J/T) Methodology

Failure in ductile materials such as highly tough ferritic materials is characterized by considerable plastic deformation and significant amount of stable crack growth. The EPFM approach outlined in this subsection considers these aspects. Two key concepts in this approach are: (1) J-integral [1, 2] which characterizes the intensity of the plastic stress-strain field surrounding the crack tip and (2) the tearing instability theory [3, 4] which examines the stability of ductile crack growth. A key advantage of this approach is that the material fracture toughness characteristic is explicitly factored into the evaluation.

3E.3.1.1 Basic (J/T) Methodology

Figure 3E.3-1 schematically illustrates the J/T methodology for stability evaluation. The material (J/T) curve in Figure 3E.3-1 represents the material's resistance to ductile crack extension. Any value of J falling on the material R-curve is denoted as J_{mat} and is a function solely of the increase in crack length Δa. Also defined in Figure 3E.3-1 is the 'applied' J, which for given stress-strain properties and overall component geometry, is a function of the applied load P and the current crack length, a. Hutchinson and Paris [4] also define the following two nondimensional parameters:

$$T_{applied} = \frac{E}{\sigma_f^2} \cdot \frac{\partial J_{applied}}{\partial a}$$

$$T_{mat} = \frac{E}{\sigma_f^2} \cdot \frac{dJ_{mat}}{da} \qquad (3E.3\text{-}1)$$

where E is Young' modulus and σ_f is an appropriate flow stress.

Intersection point of the material and applied (J/T) curves denotes the instability point. This is mathematically stated as follows:

$$J_{applied}(a,P) = J_{mat}(a) \qquad (3E.3\text{-}2)$$

$$T_{applied} < T_{mat} \text{ (stable)} \qquad (3E.3\text{-}3)$$

$$T_{applied} > T_{mat} \text{ (unstable)}$$

The load at instability is determined from the J versus load plot also shown schematically in Figure 3E.3-1. Thus, the three key curves in the tearing stability evaluation are: $J_{applied}$ versus $T_{applied}$, J_{mat} versus T_{mat} and $J_{applied}$ versus load. The determination of appropriate J_{mat} versus T_{mat} or the material (J/T) curve has been already discussed in subsection 3E.2.1. The $J_{applied}$ -$T_{applied}$ or the (J/T) applied curve can be easily generated through perturbation in the crack length once the $J_{applied}$ versus load information is available for different crack lengths. Therefore, only the methodology for the generation of $J_{applied}$ versus load information is discussed in detail.

3E.3.1.2 J Estimation Scheme Procedure

The $J_{applied}$ or J as a function of load was calculated using the GE/EPRI estimation scheme procedure [5, 6]. The J in this scheme is obtained as sum of the elastic and fully plastic contributions:

$$J = J_e + J_p \qquad (3E.3\text{-}4)$$

The material true stress-strain curve in the estimation scheme is assumed to be in the Ramberg-Osgood format:

$$\left(\frac{\varepsilon}{\varepsilon_0}\right) = \left(\frac{\sigma}{\sigma_0'}\right) + \alpha \left(\frac{\sigma}{\sigma_0}\right)^n \qquad (3E.3\text{-}5)$$

where, σ_0 is the material yield stress, $\varepsilon_0 = \frac{\sigma_0}{E}$, and α and n are obtained by fitting the preceding equation to the material true stress-strain curve.

The estimation scheme formulas to evaluate the J-integral for a pipe with a through-wall circumferential flaw subjected to pure tension or pure bending are as follows

Tension

$$J = f_1(a_e, \frac{R}{t}) \frac{P^2}{E} + \alpha \sigma_0 \epsilon_0 c (\frac{a}{b}) h_1 (\frac{a}{b}, n, \frac{R}{t}) \left[\frac{P}{P_0}\right]^{n+1} \quad (3E.3\text{-}6)$$

where,

$$f_1 (\frac{a}{b}, n, \frac{R}{t}) = \frac{a F^2(\frac{a}{b}, n, \frac{R}{t})}{4 \pi R^2 t^2}$$

$$P_0 = 2 \sigma_0 Rt [\pi - \gamma - 2 \arcsin (\frac{1}{2} \sin \gamma)]$$

Bending

$$J = f_1(a_e, \frac{R}{t}) \frac{M^2}{E} + \alpha \sigma_0 \epsilon_0 c (\frac{a}{b}) h_1 (\frac{a}{b}, n, \frac{R}{t}) \left[\frac{M}{M_0}\right]^{n+1} \quad (3E.3\text{-}7)$$

where,

$$f_1 (\frac{a}{b}, n, \frac{R}{t}) = \pi a (\frac{R}{I})^2 F^2 (\frac{a}{b}, n, \frac{R}{t})$$

$$M_0 = M_0 [\cos (\frac{\gamma}{2}) - \frac{1}{2} \sin (\gamma)]$$

The nondimensional functions F and h are given in Reference 6

While the calculation of J for given α, n, σ_0 and load type is reasonably straightforward, one issue that needs to be addressed is the tearing instability evaluation when the loading includes both the membrane and the bending stresses. The estimation scheme is capable of evaluating only one type of stress at a time. This aspect is addressed next.

3E.3.1.3 Tearing Instability Evaluation Considering Both the Membrane and Bending Stresses

Based on the estimation scheme formulas and the tearing instability methodology just outlined, the instability bending and tension stresses can be calculated for various through-wall circumferential flaw lengths. Figure 3E.3-2 shows a schematic plot of the instability stresses as a function of flaw length. For the same stress level, the allowable flaw length for the bending is expected to be larger than the tension case.

When the applied stress is a combination of the tension and bending, a linear interaction rule is used to determine the instability stress or conversely the critical flaw length. The application of linear interaction rule is certainly conservative when the instability load is close to the limit load. The applicability of this proposed rule should be justified by providing a comparision of the predictions by the proposed approach (or an alternate approach) with those available for cases where the combination is treated together.

The interaction formulas are following: (See Figure 3E.3-2)

Critical Flaw Length

$$a_c = (\frac{\sigma_t}{\sigma_t+\sigma_b}) a_{c,t} + (\frac{\sigma_b}{\sigma_t+\sigma_b}) a_{c,b} \quad (3E.3\text{-}8)$$

where:

σ_t = applied membrane stress

σ_b = applied bending stress

$a_{c,t}$ = critical flaw length for a tension stress of $(\sigma_t+\sigma_b)$

$a_{c,b}$ = critical flaw length for a bending stress of $(\sigma_t+\sigma_b)$

Instability Bending Stress

$$S_b = (1 - \frac{\sigma_t}{\sigma_t'}) \sigma'_b \quad (3E.3\text{-}9a)$$

where:

S_b = instability bending stress for flaw length, a, in the presence of membrane stress, σ_t.

σ_t = applied membrane stress

σ_t = instability tension stress for flaw length, a.

σ_b' = instability bending stress for flaw length, a.

Once the instability bending stress, S_b, in the presence of membrane stress, σ_t, is determined, the instability load margin corresponding to the detectable leak-size crack (as required by LBB criterion in Section 3.6.3) can be calculated as follows:

Instability Load Margin (3E.3-9b)

$$\frac{\sigma_t + S_b}{\sigma_t + \sigma_b}$$

It is assumed in the preceding equation that the uncertainty in the calculated applied stress is essentially associated with the stress due to applied bending loads and that the membrane stress, which is generally due to the pressure loading, is known with greater certainty. This method of calculating the margin against loads is also consistent with the definition of load margin employed in Paragraph IWB-3640 of Section XI [7].

3E.3.2 Application of (J/T) Methodology to Carbon Steel Piping

From Figure 3E.2-3, it is evident that carbon steels exhibit transition temperature behavior marked by three distinct stages: lower shelf, transition and upper shelf. The carbon steels generally exhibit ductile failure mode at or above upper shelf temperatures. This would suggest that a net-section collapse approach may be feasible for the evaluation of postulated flaws in carbon steel piping. Such a suggestion was also made in a review report prepared by the Naval Research Lab [8]. Low temperature (i.e. less than 125^{O}F) pipe tests conducted by GE [9] and by Vassilaros [10] which involved circumferentially cracked pipes subjected to bending and/or pressure loading, also indicate that a limit load approach is feasible. However, test data at high temperatures specially involving large diameter pipes are currently not available. Therefore, a (J/T) based approach is used in the evaluation.

3E.3.2.1 Determination of Ramberg-Osgood Parameters for 550^{O}F Evaluation

Figure 3E.2-6a shows the true stress-true strain curves for the carbon steels at 550^{O}F. The same data is plotted here in Figure 3E.3-3 in the Ramberg-Osgood format. It is seen that, unlike the stainless steel case, each set for stress-strain data (i.e. data derived from one stress-strain curve) follow approximately a single slope line. Based on the visual observation, a line representing $\alpha = 2$, $n = 5$ in Figure 3E.3-3 was drawn as representing a reasonable upper bound to the data shown.

The third parameter in the Ramberg-Osgood format stress-stain curve is σ_o, the yield stress. Based on the several internal GE data on carbon steels such as SA 333 Gr.6, and SA 106 Gr.B, a reasonable value of 550^{O}F yield strength was judged as 34600 psi. To summarize, the following values are used in this appendix for the (J/T) methodology evaluation of carbon steels as 550^{O}F:

α = 2.0

n = 5.0

σ_0 = 34600 psi

E = 26×10^{6} psi

3E.3.2.2 Determination of Ramberg-Osgood Parameters for 420^{O}F Evaluation

Figure 3E.3-4 shows the Ramberg-Osgood (R-O) format plot of the 350^{O}F true stress-stain data on the carbon steel base metal. Also shown in Figure 3E.3-4 are the CE data a SA 106 Grade B at 400^{O}F. Since the difference between the ASME Code Specified minimum yield strength at 350^{O}F and 420^{O}F is small, the 350^{O}F stress-strain data were considered applicable in the determination of R-O parameters for evaluation at 420^{O}F.

A review of Figure 3E.3-4 indicates that the majority of the data associated with any one test can be approximated by one straight line.

It is seen that some of the data points associated with the yield point behavior fall along the y-axis. However, these data points at low stain level were not considered significant and, therefore, were not included in the R-O fit.

The 350^{O}F yield stress for the base material is given in Table 3E.2-3 as 37.9 ksi. Since the difference between the ASME Code specified minimum yield strengths of pipe and plate carbon steels at 420^{O}F and 350^{O}F is roughly 0.9 ksi, the σ_o value for use at 420^{O}F are chosen as (37.9^{0} - 0.9) or 37 ksi. In summary, the following values of R-O parameters are used for evaluation of 420^{O}F:

σ_o = 37,000 psi

α = 5.0

n = 4.0

3E.3.3 Modified Limit Load Methodology for Austenitic Stainless Steel Piping

Reference 16 describes a modified limit load methodology that may be used to calculate the critical flaw lengths and instability loads for austenitic stainless steel piping and associated welds. If appropriate, this or an equivalent methodology may be used in lieu of the (J/T) methodology described in 3E.3.1.

3E.3.4 Bimetallic Welds

For joining austenitic steel to ferritic steel, the Ni-Cr-Fe Alloys 82 or 182 are generally used for weld metals. The procedures recommended in Section 3E.3.3 for the austenitic welds are applicable to these weld metals. This is justified based on the common procedures adopted for flaw acceptance in the ASME Code Section XI, Article IWB-3600 and Appendix C, for both types of the welds. If other types of bimetallic weld metals are used, proper procedures should be used with generally acceptable justification.

3E.3.5 References

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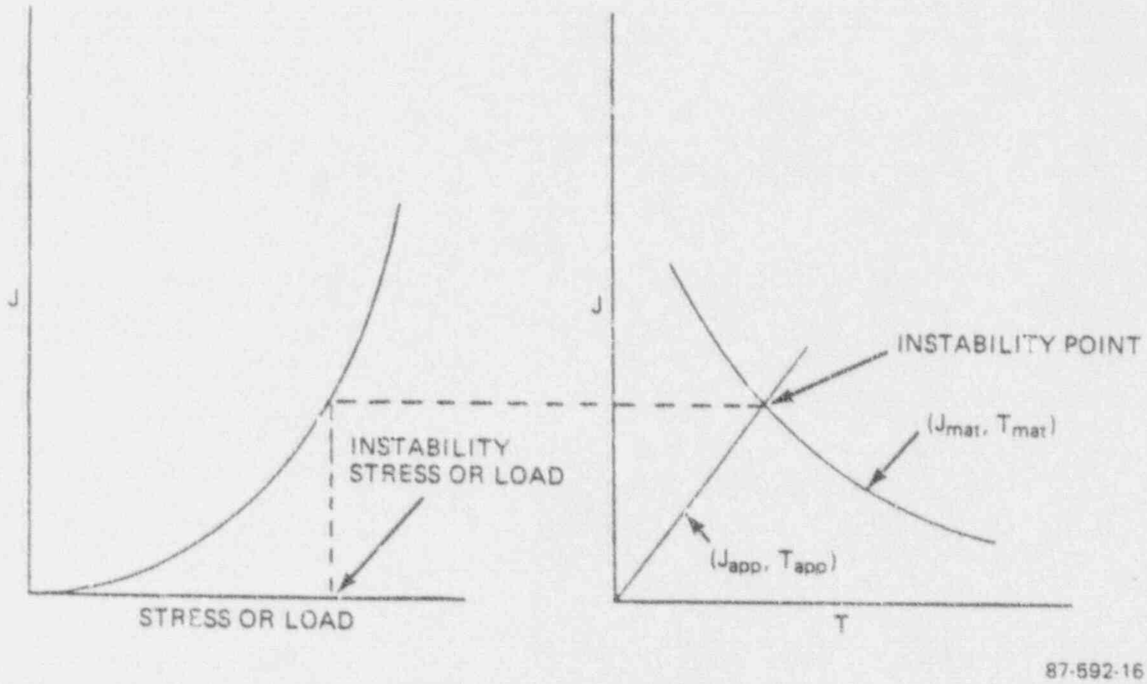

Figure 3E.3-1 SCHEMATIC ILLUSTRATION OF TEARING STABILITY EVALUATION

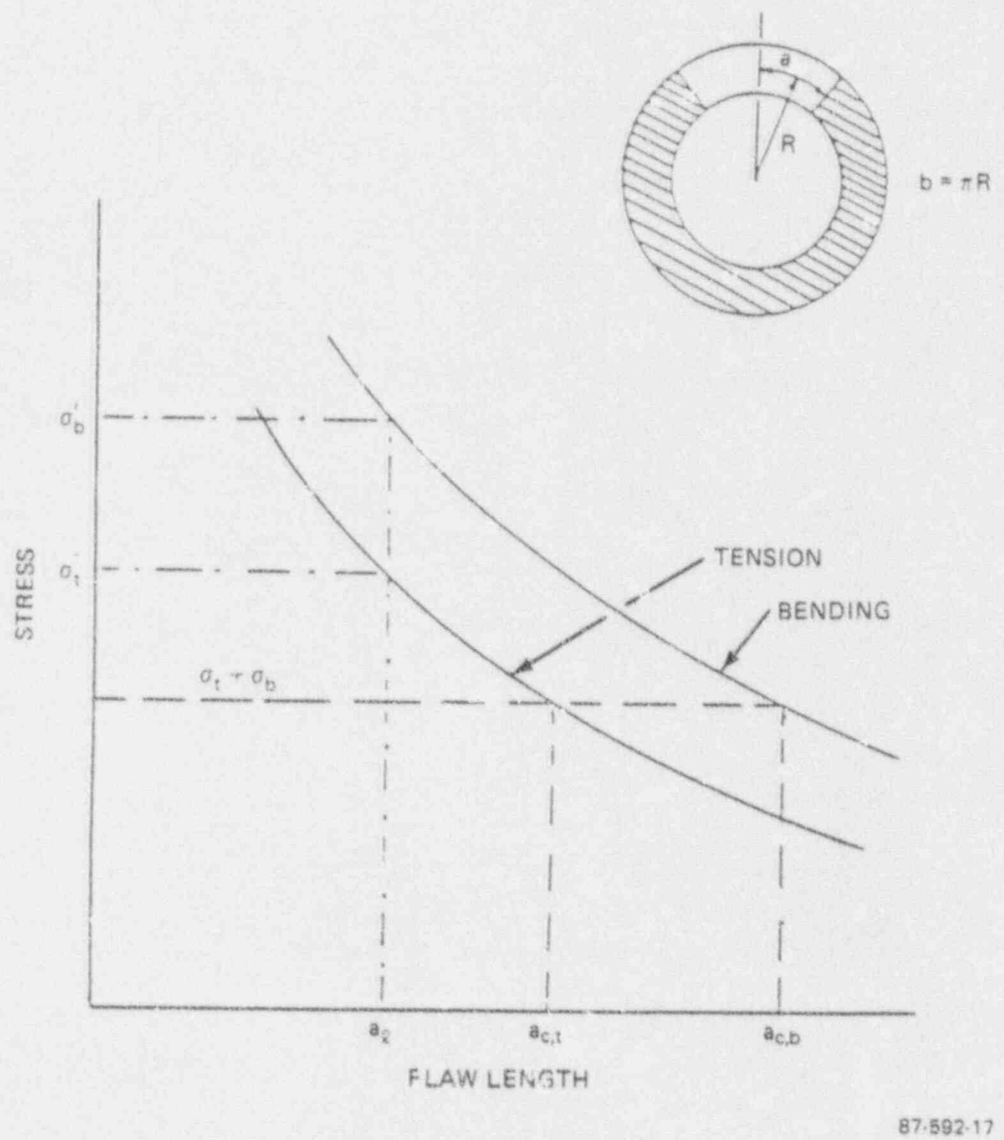

Figure 3E.3-2 A SCHEMATIC REPRESENTATION OF INSTABILITY TENSION AND BENDING STRESSES AS A FUNCTION OF FLAW STRENGTH

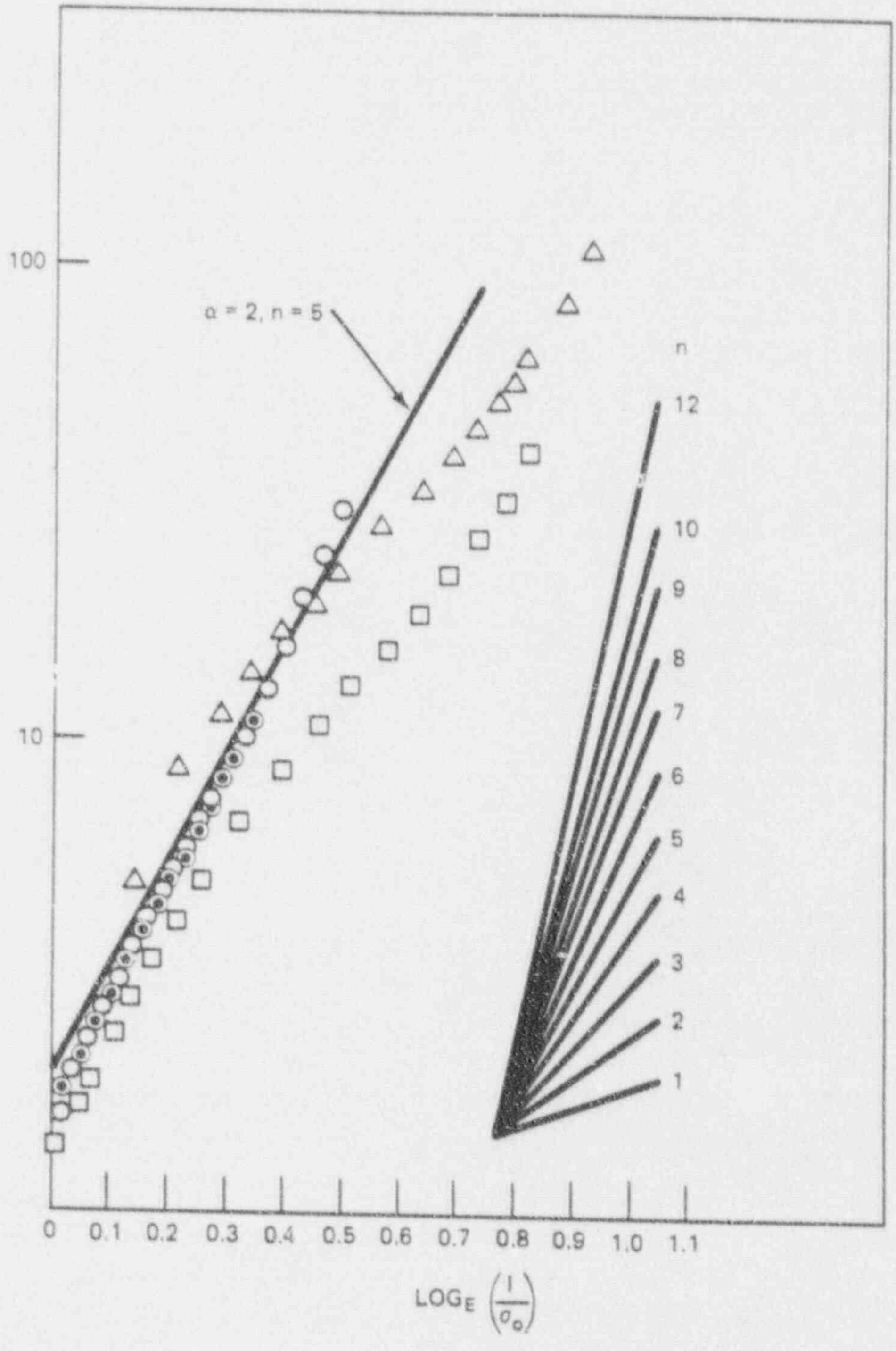

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Figure 3E.3-3 SA 333 GR. 6 STRESS–STRAIN DATA AT 550° F IN THE RAMBERG–OSGOOD FORMAT

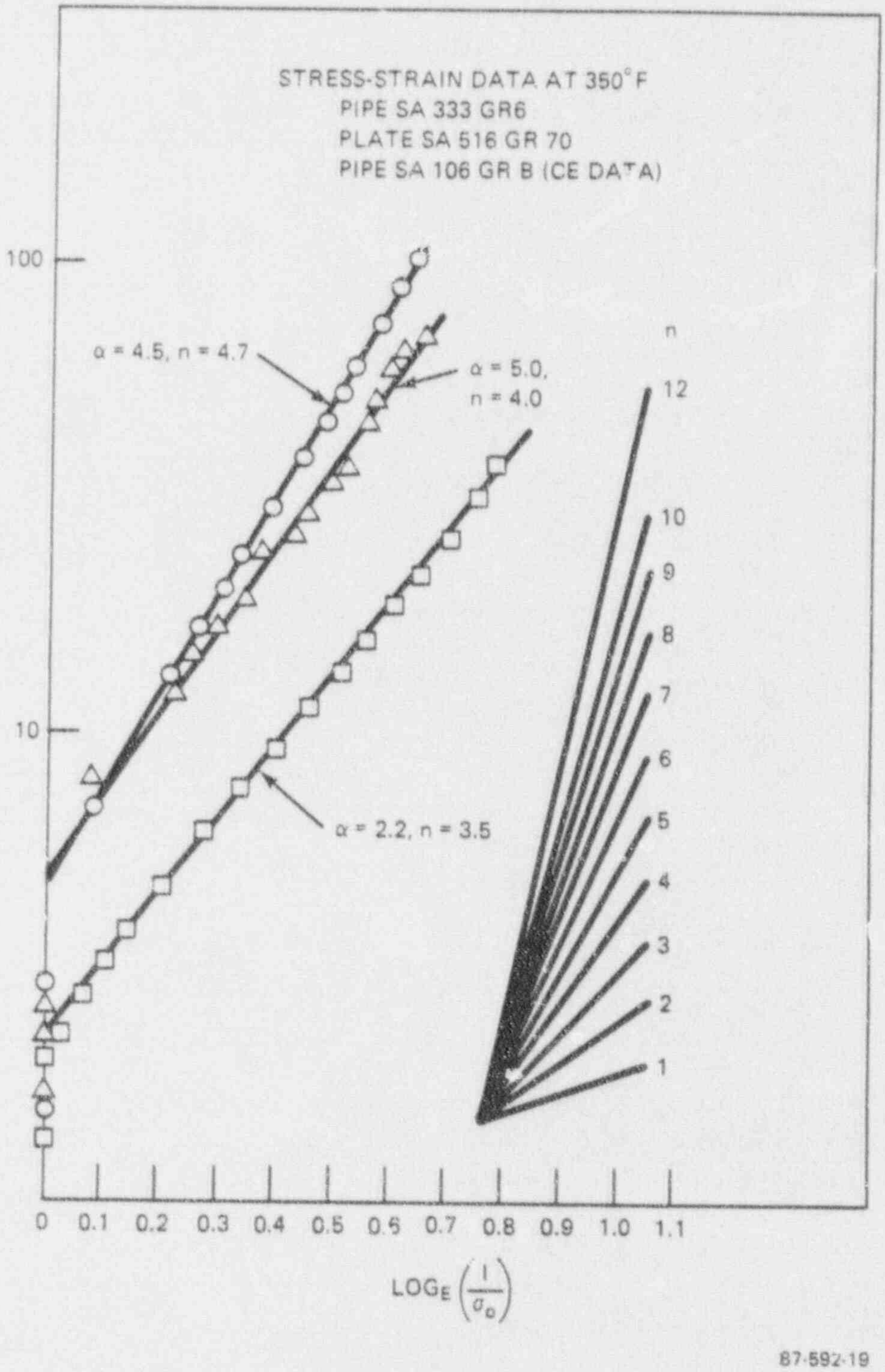

Figure 3E.3-4 CARBON STEEL STRESS–STRAIN DATA AT 350°F IN THE RAMBERG–OSGOOD FORMAT

SECTION 3E.4

CONTENTS

SECTION 3E.4

ILLUSTRATIONS

3E.4 LEAK RATE CALCULATION METHODS

Leak rates of high pressure fluids through cracks in pipes are a complex function of crack geometry, crack surface roughness, applied stresses, and inlet fluid thermodynamic state. Analytical predictions of leak rates essentially consist of two separate tasks: calculation of the crack opening area, and the estimation of the fluid flow rate per unit area. The first task requires the fracture mechanics evaluations based on the piping system stress state. The second task involves the fluid mechanics considerations in addition to the crack geometry and its surface roughness information. Each of these tasks are now discussed separately considering the type of fluid state in BWR piping.

3E.4.1 Leak Rate Estimation for Pipes Carrying Water

EPRI-developed computer code PICEP [1] may be used in the leak rate calculations. The basis for this code and comparison of its leak rate predictions with the experimental data is described in References 2 and 3. This code was has been used in the successful application of LBB to primary piping system of a PWR. The basis for flow rate and crack opening area calculations in PICEP is briefly described first. A comparison with experimental data is shown next.

Other methods (e.g., Reference 4) may be used for leak rate estimation at the descretion of the applicant.

3E.4.1.1 Description of Basis for Flow Rate Calculation

The thermodynamic model implemented in PICEP computer program assumes the leakage flow through pipe cracks to be isenthalpic and homogeneous, but it accounts for non-equilibrium "flashing" transfer process between the liquid and vapor phases.

Fluid friction due to surface roughness of the walls and curved flow paths has been incorporated in the model. Flows through both parallel and convergent cracks can be treated. Due to the complicated geometry within the flow path, the model uses some approximations and empirical factors which were confirmed by comparison against test data.

For given stagnation conditions and crack geometries, the leak rate and exit pressure are calculated using an iterative search for the exit pressure starting from the saturation pressure corresponding to the upstream temperature and allowing for friction, gravitational, acceleration and area change pressure drops. The initial flow calculation is performed when the critical pressure is lowered to the back pressure without finding a solution for the critical mass flux.

A conservative methodology was developed to handle the phase transformation into a two-phase mixture or superheated steam through a crack. To make the model continuous, a correction factor was applied to adjust the mass flow rate of a saturated mixture to be equal to that of a slightly subcooled liquid. Similarly, a correction factor was developed to ensure continuity as the steam became superheated. The superheated model was developed by applying thermodynamic principles to an isentropic expansion of the single phase steam.

The code can calculate flow rates through fatigue or IGSCC cracks and has been verified against data from both types. The crack surface roughness and the number of bends account for the difference in geometry of the two types of cracks. The guideline for predicting leak rates through IGSCCs when using this model was based on obtaining the number of turns that give the best agreement for Battelle Phase II test data of Collier et al. [5]. For fatigue cracks, it is assumed that the crack path has no bends.

3E.4.1.2 Basis for Crack Opening Area Calculation

The crack opening area in PICEP code is calculated using the estimation scheme formulas. The plastic contribution to the displacent is computed by summing the contributions of bending and tension alone, a procedure that underestimates the displacent from combined tension and bending. However, the plastic contribution is expected to be insignificant because the applied stresses at normal operation are generally such that they do not produce significant plasticity at the cracked location.

3E.4.1.3 Comparison Verification with Experimental Data

Figure 3E.4-1 from Reference 3 shows a comparison PICEP prediction with measured leak rate data. It is seen that PICEP predictions are virtually always conservative (i.e., the leak flow rate is underpredicted).

3E.4.2 Flow Rate Estimation for Saturated Steam

3E.4.2.1 Evaluation Method

The calculations for this case were based on the maximum two-phase flow model developed by Moody [Reference 6]. However, in an LBB-report, a justification should be provided by comparing the predictions of this method with the available experimental data, or a generally accepted method, if available, should be used.

The Moody predicts the flow rate of steam-water mixtures in vessel blowdown from pipes (see Figure 3E.4-2). A key parameter that characterized the flow passage in the Moody analysis is fL/D_h, where, f is the coefficient of friction, L, the length of the flow passage and D_h, the hydraulic diameter. The hydraulic diameter for the case of flow through a crack is 2δ where δ is the crack opening displacement and the length of the flow passage is t, the thickness of the pipe. Thus, the parameter fL/D_h in the Moody analysis was interpreted as $ft/2\delta$ for the purpose of this evaluation.

Figure 3E.4-3 shows the predicted mass flow rates by Moody for fL/D_h of 0 and 1. Similar plots are given in Reference 6 for additional fL/D_h values of 2 through 100. Since the steam in the ABWR main steam lines would be essentially saturated, the mass flow rate corresponding to the upper saturation envelope line is the appropriate one to use. Table 3E.4-1 shows the mass flow rates for a range of fL/D_h values for a stagnation pressure of 1000 psi which is roughly equal to the pressure in an ABWR piping system carrying steam.

A major uncertainty in calculating the leakage rate is the value of f. This is discussed next.

3E.4.2.2 Selection of Appropriate Friction Factor

Typical relationships between Reynolds' Number and relative roughness ϵ/D_h, the ratio of effective surface protrusion height to hydraulic diameter, were relied upon in this case. Figure 3E.4-4, from Reference 7, graphically shows such a relationship for pipes. The ϵ/D_h ratio for pipes generally ranges from 0 to 0.50. However, for a fatigue crack consisting of rough fracture surfaces represented by a few mils, the roughness height ϵ at some location may be almost as much as δ. In such cases, ϵ/D_h would seem to approach 1/2. There are no data or any analytical model for such cases, but a crude estimate based on the extrapolation of the results in Figure 3E.4-4 would indicate that f may be of the order of 0.1 to 0.2. For this evaluation an average value of 0.15 was used with the modification as discussed next.

For blowdown of saturated vapor, with no liquid present, Moody states that the friction factor should be modified according to

$$f_g = f_{GSP}\left[\frac{\nu f}{\nu g}\right]^{1/3} \qquad \text{(3E.4-1)}$$

where

f_g = modified friction factor

f_{GSP} = factor for single phase

$\frac{\nu f}{\nu g}$ = liquid/vapor specific volume ratio evaluated at an average static pressure in the flow path

This correction is necessary because the absence of a liquid film on the walls of the flow channel at high quality makes the two-phase flow model invalid as it stands. The average static pressure in the flow path is going to be something in excess of 500 psia if the initial pressure is 1000 psia; this depends on the amount of flow choking and can be determined from Reference 6. However, a fair estimate of $(\nu f/\nu g)^{1/3}$ is 0.3, so the friction factor for saturated steam blowdown may be taken as 0.3 of that for mixed flow.

Based on this discussion, a coefficient of friction of 0.15 x 0.3 = 0.045 was used in the flow rate estimation. Currently experimental data are unavailable to validate this assumed value of coefficient of friction.

3E.4.2.3 Crack Opening Area Formulation

The crack opening areas were calculated using LEFM procedures with the customary plastic zone correction. The loadings included in the crack opening area calculations were: pressure, weight and thermal expansion.

The mathematical expressions given by Paris and Tada [8] are used in this case. The crack opening areas for pressure (A_p) and bending stresses (A_b) were separately calculated and then added together to obtain the total area, A_c.

For simplicity, the calculated membrane stresses from weight and thermal expansion loads were combined with the axial membrane stress, σ_p, due to the pressure.

The formulas are summarized below:

$$A_p = \frac{\sigma_p}{E} (2\pi Rt) G_p (\lambda) \qquad (3E.4\text{-}2)$$

where,

σ_p = axial membrane stress due to pressure, weight and thermal expansion loads.

E = Young's modulus

R = pipe radius

t = pipe thickness

λ = shell parameter = $a/\sqrt{Rt}$

a = half crack length

(3E.4-3)

$$G_p(\lambda) = \lambda^2 + 0.16\lambda^4 \quad (0 \le \lambda \le 1)$$

$$= 0.02 + 0.81\lambda^2 + 0.30\lambda^3 + 0.03\lambda^4 \quad (1 \le \lambda \le 5)$$

$$A_b = \frac{\sigma_b}{E} \cdot \pi \cdot R^2 \cdot \left(\frac{3 + \cos\theta}{4}\right) I_t^{(\theta)} \qquad (3E.4\text{-}4)$$

where,

σ_b = bending stress due to weight and thermal expansion loads

θ is half crack angle

(3E.4-5)

$$I_t(\theta) = 2\theta^2 \left[1 + \left(\frac{\theta}{\pi}\right)^{3/2} \left\{8.6 - 13.3\left(\frac{\theta}{\pi}\right) + 24\left(\frac{\theta}{\pi}\right)^2\right\} + \left(\frac{\theta}{\pi}\right)^3 \left\{22.5 - 75\left(\frac{\theta}{\pi}\right) + 205.7\left(\frac{\theta}{\pi}\right)^2 - 247.5\left(\frac{\theta}{\pi}\right)^3 + 242\left(\frac{\theta}{\pi}\right)^4\right\}\right]$$

$(0 < \theta < 100^\circ)$

The plastic zone correction was incorporated by replacing a and θ in these formulas by a_e and θ_e which are given by

$$\theta_{eff} = \theta + \frac{K_{total}^2}{2\pi R \sigma_Y^2} \qquad (3E.4\text{-}6)$$

$$a_e = \theta_e \cdot R$$

The yield stress, σ_y, was conservatively assumed as the average of the code specified yield and ultimate strength. The stress intensity factor, K_{total}, includes contribution due to both the membrane and bending stress and is determined as follows:

$$K_{total} = K_m + K_b \qquad (3E.4\text{-}7)$$

where,

$$K_m = \sigma_p \sqrt{a} \cdot F_p(\lambda)$$

$$F_p(\lambda) = (1 + 0.3225\lambda^2)^{\frac{1}{2}}$$

$$= 0.9 + 0.25\lambda \qquad (0 \le \lambda \le 1) \quad (1 \le \lambda \le 5)$$

$$K_b = \sigma_b \cdot \sqrt{\pi a} \cdot F_b(\theta)$$

$$F_b(\theta) = 1 + 6.8\left(\frac{\theta}{\pi}\right)^{3/2} - 13.6\left(\frac{\theta}{\pi}\right)^{5/2} + 20\left(\frac{\theta}{\pi}\right)^{7/2}$$

$$(0 \le \theta \le 100^0)$$

The steam mass flow rate, M, shown in Table 3E.4-1 is a function of parameter, $f\ell/2\delta$. Once the mass flow rate is determined corresponding to the calculated value of this parameter, the leak rate in gpm can then be calculated.

3E.4.3 References

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TABLE 3E.4-1

MASS FLOW RATE FOR SEVERAL fl/D_h VALUES

fl/D_h	MASS FLOW RATE, 1bm/sec-ft.^{2} M
0	3800
1	2200
2	1600
3	1150
4	920
5	800
10	580
20	400
50	260
100	185

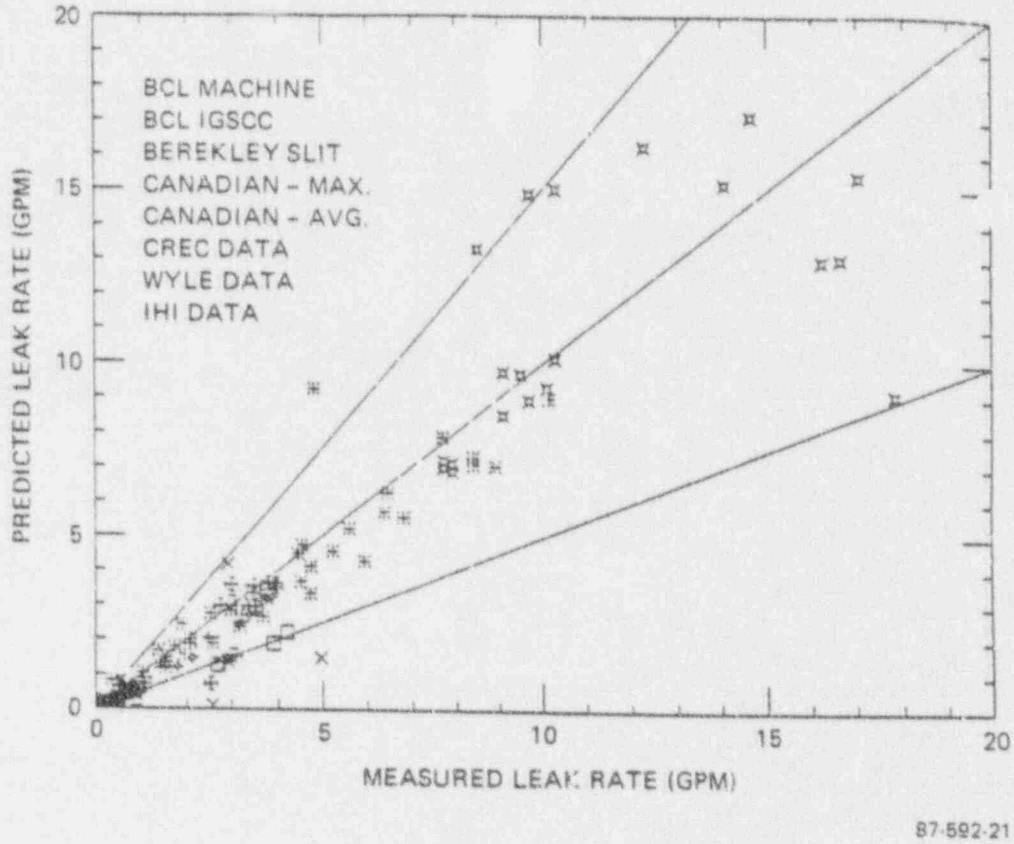

Figure 3E.4-1 COMPARISON OF PICEP PREDICTIONS WITH MEASURED LEAK RATES

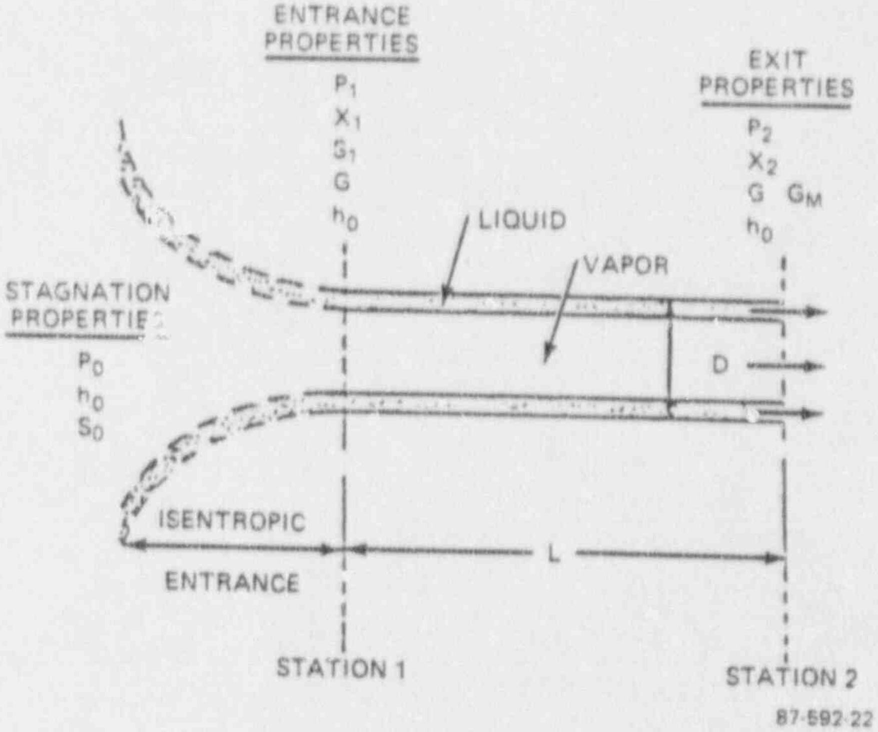

Figure 3E.4–2 PIPE FLOW MODEL

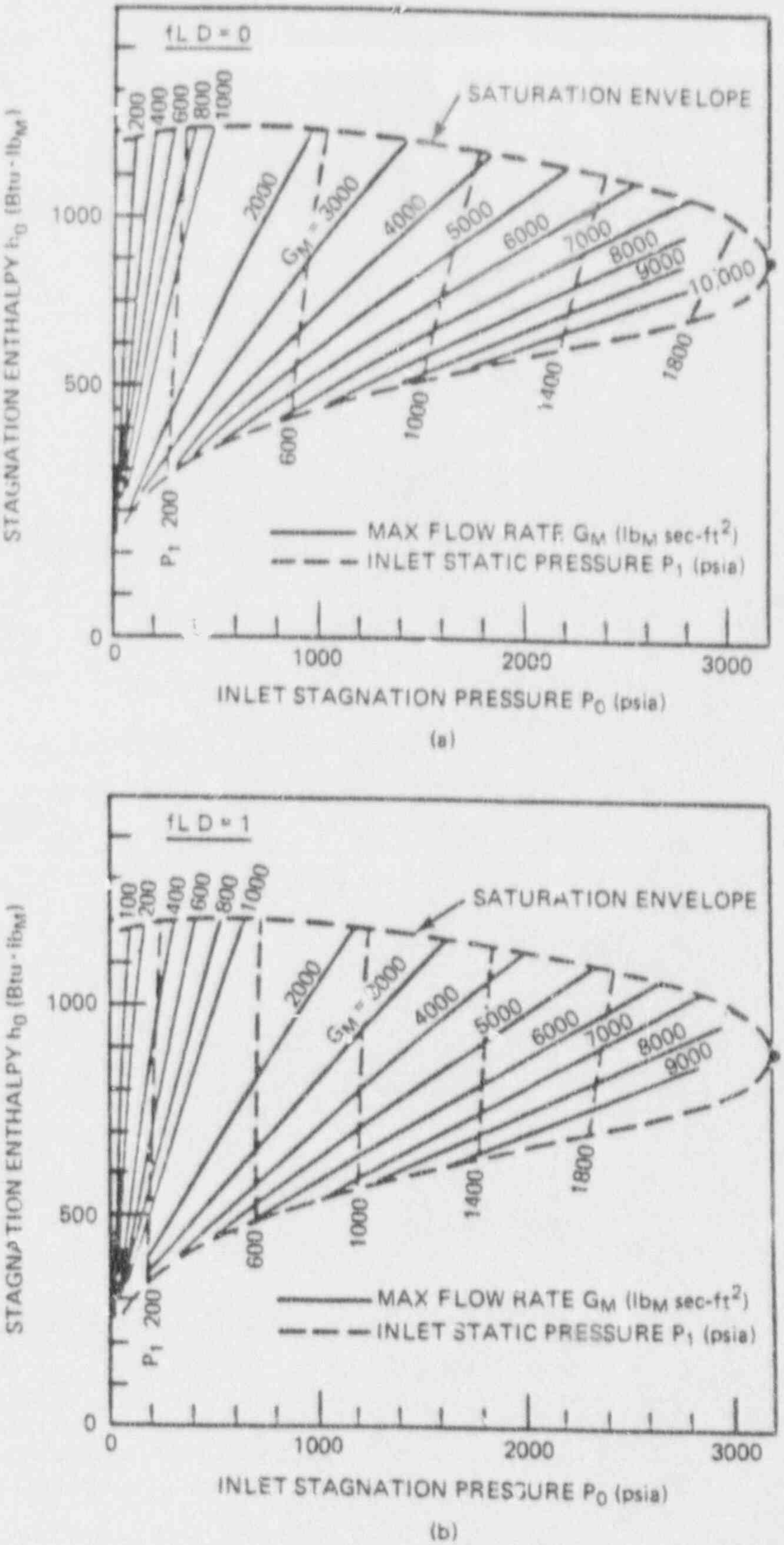

Figure 3E.4-3 MASS FLOW RATES FOR STEAM/WATER MIXTURES

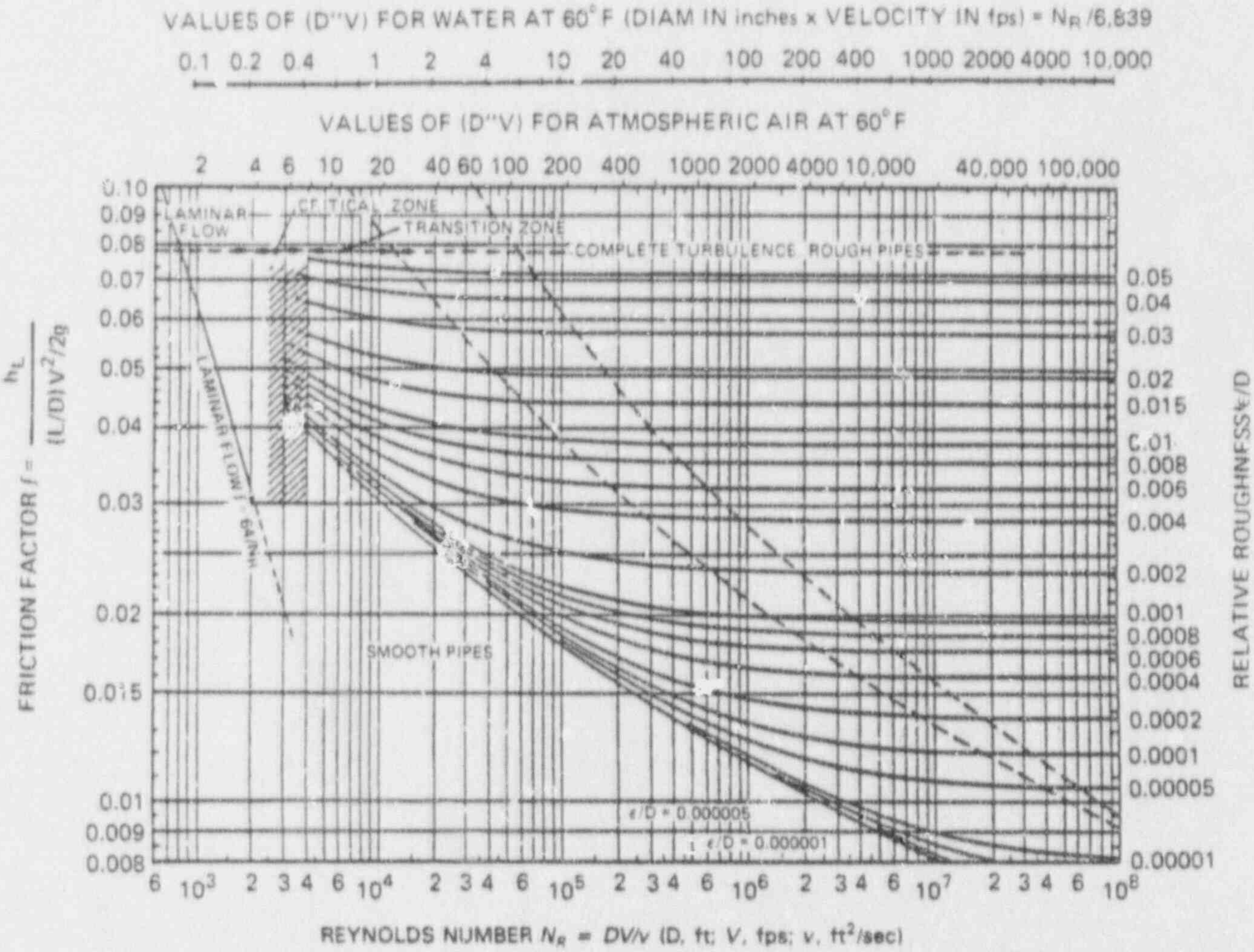

Figure 3E.4-4 FRICTION FACTORS FOR PIPES

3E.5 LEAK DETECTION CAPABILITIES

A complete description of various leak detection systems is provided in Subsection 5.2.5. The leakage detection system gives separate considerations to: leakage within the drywell and leakage external to the drywell. The limits for reactor coolant leakage are described in Subsection 5.2.5.4.

The total leakage in the drywell consists of the identified leakage and the unidentified leakage. The identified leakage is that from pumps, valve stem packings, reactor vessel head seal and other seals, which all discharge to the equipment drain sump. The technical specification limit on the identified leak rate is expected to be 25 gpm.

The unidentified leak rate in the drywell is the portion of the total leakage received in the drywell sumps that is not identified as previously described. The licensing (technical specification) limit on unidentified leak rate is 1 gpm. To cover uncertainties in leak detection capability, although it meets Regulatory Guide 1.45 requirements, a margin factor of 10 is required per Reference 16 of Subsection 3E.3.4 to determine a reference leak rate. A reduced margin factor may be used if accounts can be made of effects of sources of uncertainties such as plugging of the leakage crack with particulate material over time, leakage prediction, measurement techniques, personnel and frequency of monitoring. For the piping in drywell, a reference leak rate of 10 gpm may be used, unless a smaller rate can be justified.

The sensitivity and reliability of leakage detection systems used outside the drywell must be demonstrated to be equivalent to Regulatory Guide 1.45 systems. Methods that have been shown to be acceptable include local leak detection, for example, visual observation or instrumentation. Outside the drywell, the leakage rate detection and the margin factor depend upon the design of the leakage detection systems.

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TABLES

SECTION 3E.6

TABLES (Continued)

ILLUSTRATIONS

3E.6 GUIDELINES FOR PREPARATION OF AN LBB REPORT

Some of the key elements of an LBB evaluation report for a high energy piping system are: system description, evaluation of susceptibility to water hammer and thermal fatigue, material specification, piping geometry, stresses and the LBB margin in evaluation results. Two examples are presented in the following subsections to provide guidelines and illustrations for preparing an lbb evaluation report.

3E.6.1 Main Steam Piping

3E.6.1.1 System Description

The four 28-inch (700 mm) main steam (MS) lines carry steam from the reactor to the turbine and auxiliary systems. The reactor coolant pressure boundary portion of each line being evaluated in this section connects to a flow restrictor which is a part of the reactor pressure vessel nozzle and is designed to limit the rate of escaping steam from the postulated break in the downstream steam line. The restrictor is also used for flow measurements during plant operation. The safety relief valves (SRVs) discharge into the pressure suppression pool through SRV discharge piping. The SRV safety function includes protection against overpressure of the reactor primary system. The main steam line A has a branch connection to supply steam to the reactor core isolation cooling (RCIC) system turbine.

This section addresses the MS piping system in the reactor building which is designed and constructed to the requirements of the ASME Code, Section III, Class 1 piping (within outermost isolation valve) and Class 2 piping. It is classified as Seismic Category I. It is inspected according to ASME Code Section XI.

3E.6.1.2 Susceptibility to Water Hammer

Significant pressure pulsation of water hammer effect in the pipe may occur as a result of opening of SRVs or closing of the turbine stop valve. A brief description of these phenomena follows. These two transients are considered in the main steam piping system design and fatigue analysis. These events are more severe than the opening or closing of a main steam isolation valve or water carry over through main steam and SRV piping. Moreover, the probability of water carry over during core flooding in case of an accident is low.

Safety Relief Valve Lift Transient Description

SRV produces momentary unbalanced forces acting on the discharge piping system for the period from the opening of the SRV until a steady discharge flow from the reactor pressure vessel to the suppression pool is established. This period includes clearing of the water slug at the end of the discharge piping submerged in the suppression pool. Pressure waves traveling through the discharge piping following the relatively rapid opening of the SRVs causes the discharge piping to vibrate. This in turn produces time dependent forces that act on the main steam piping segments.

There are a number of events/transients/postulated accidents that result in SRV lift:

a. Automatic opening signal when main steam system pressure exceeds the set point for a given valve (there are different set points for different valves in a given plant).

b. Automatic opening signal for all valves assigned to the automatic depressurization system function on receipt of proper actuation signal.

c. Manual opening signal to valve selected by plant operator.

The SRVs close when the main steam system pressure reaches the relief mode reseat pressure or when the plant operator manually releases the opening signals.

It is assumed (for conservatism) that all SRVs are activated at the same time, which produces simultaneous forces on the main steam piping system.

Turbine Stop Valve Closure Transient Description

Prior to turbine stop valve closure, saturated steam flows through each main steam line at nuclear boiler rated pressure and mass flow rate. Upon signal, the turbine stop valves close rapidly and the steam flow stops at the upstream side of these valves. A pressure wave is created and travels at sonic velocity toward the reactor vessel through each main stream line. The flow of steam into each main steam line from the reactor vessel continues until the fluid compression wave reaches the reactor vessel nozzle. Repeated reflection of the pressure wave at the reactor vessel and stop valve ends of the main steam lines produces time varying pressures and velocities at each point along the main steam lines. The combination of fluid momentum changes, shear forces, and pressure differences cause forcing functions which vary with position and time to act on the main steam piping system. The fluid transient loads due to turbine stop valve closure is considered as design load for upset condition.

Basic Fluid Transient Concept

Despite the fact that the SRV discharge and the turbine stop valve closure are flow-starting and flow-stopping processes, respectively, the concepts of mass, momentum, and energy conservation and the differential equations which represent these concepts are similar for both problems. The particular solution for either of the problems is obtained by incorporating the appropriate initial conditions and boundary conditions into the basic equations. Thus, relief valve discharge and turbine stop valve closure are seen to be specific solutions of the more general problem of compressible, non-steady fluid flow in a pipe.

The basic fluid dynamic equations which are applicable to both relief valve discharge and turbine stop valve closure are used with the particular fluid boundary conditions of these occurrences. Step-wise solution of these equations generates a time-history of fluid properties at numerous locations along the pipe. Simultaneously, reaction loads on the pipe are determined at each location corresponding to the position of an elbow.

The computer programs RVFOR and TSFOR described in Appendix 3D are used to calculate the fluid transient forces on the piping system due to safety relief valve discharge and turbine stop valve closure. Both of the programs use method of characteristics to calculate the fluid transients.

The results from the RVFOR program have been verified with various inplant test measurements such as from the Monticello tests and Caoroso tests and the test sponsored by BWR owner for NUREG-0737 at Wyle test facilities, Huntsville, Alabama. Various data from the strain gages on the pipes and the load cells on the supports were compared with the analytical data and found to be in good correlation.

Evaluation of the ensuing effects are considered as a normal design process for the main steam piping system. The peak pressure pulses are within the design capability of a typical piping design and the piping stresses and support loads remain within the ASME Code allowables.

It is concluded that, during these water hammer type events, the peak pressures and segment loads would not cause overstressed conditions for the main steam piping system.

3E.6.1.3 Thermal Fatigue

No thermal stratification and thermal fatigue are expected in the main steam piping since there is no large source of cold water in these lines. A small amount of water may collect in the near horizontal leg of the main steam line due to steam condensation. However, a slope of 1/8 inch per foot of main steam piping is provided in each main steam line. Water drain lines are provided at the end of slope to drain out the condensate. Thus, in this case no significant thermal cycling effects on the main steam piping are expected.

3E.6.1.4 Piping, Fittings and Safe End Materials

The material specified for the 28-inch main steam pipe is SA672 Grade C70. The corresponding specification for the piping fittings and forgings are given as SA420, WPL6 and SA350, LF2, respectively. The material for the safe end forging welded between the main steam piping and the steam nozzle is SA508 Class 3.

3E.6.1.5 LBB Margin Evaluation

The Code stress analysis of the piping is reviewed to obtain representative stress magnitudes. Table 3E.6-1 shows, for example purposes, the stress magnitudes due to pressure, weight, thermal expansion and SSE loads.

The leak rate calculations are performed assuming saturated steam conditions at 1050 psi. The leak rate model for saturated steam developed in Section 3E.4.2 is be used in this evaluation. Pressure, weight and thermal expansion stresses are included in calculating the crack opening area. A plot of leak rate as a function of crack size is developed as is shown in Figure 3E.6-1. The leakage flaw length corresponding to the reference leak rate (see Section 3E.5) is determined from this figure.

The calculations for the critical flaw size and instability load corresponding to leakage-size crack are performed using the J-T methodology. Specifically, the 550°F J-R curve shown in Figure 3E.2-8 and the Ramberg-Osgood parameters given in Subsection 3E.3.2.1 are used. A plot of instability tension and bending stresses as a function of crack length is developed. Table 3E.6-2 shows the example presentation of calculated critical crack size and the margin along with the instability load margin for the leakage size cracks. It is noted that the critical crack size margin is greater than 2 and the instability load margin also exceeds $\sqrt{2}$.

3E.6.1.6 Conclusion

For all example main steam lines, based upon the reference leakage rates and assumed stress magnitudes, leakage flaw lengths are calculated and compared against the critical flaw length. The margin is shown to be greater than 2 for the leakage rates. Also, the leak-size crack stability evaluation is shown to have a margin of at least !2.

It is also shown that the conditions required for applicability of LBB (see Subsection 3.6.3.2), such as high resistance to failure from effects of IGSCC, water hammer and thermal fatigue, are satisfied. Therefore, all four of the main steam lines qualify for LBB behavior.

3E.6.2 Feedwater Piping Example

3E.6.2.1 System Description

The function of the feedwater (FW) system is to conduct water to the reactor vessel over the full range of the reactor power operation. The feedwater piping consists of two 22-inch (550 mm) diameter lines from the high-pressure feedwater heaters, connecting to the reactor vessel through three 12-inch (300 mm) risers on each line. Each line has one check valve inside the containment drywell and one positive closing check valve outside containment. During shutdown cooling mode, reactor water pumped through the RHR heat exchanger in one loop is returned to the vessel by way of one feedwater line.

This section addresses the feedwater piping in the reactor building, extending from the vessel out to the outboard isolation valve (ASME Class 1) and further through the shutoff valve to and including the seismic interface restraint (ASME Class 2). This section of the feedwater piping is classified as Seismic Category I.

3E.6.2.2 Susceptibility to Water Hammer

There is no record of feedwater piping failure due to water hammer. Although there are several check valves in the feedwater system, operating procedure and the control systems have been designed to limit the magnitude of water hammer load to the extent that a formal design is not required.

3E.6.2.3 Thermal Fatigue

Thermal fatigue is not a concern in ABWR feedwater piping. The ASME Code evaluation includes operating temperature transients, cold and hot water mixing and thermal stratification.

3E.6.2.4 Piping, Fittings and Safe End Material

The material for piping is either SA333, Gr. 6 or SA-672, Gr. C70.

3E.6.2.5 Piping Sizes, Geometries and Stresses

Table 3E.6-3 shows the normal operating temperatures, pressures and thickness for representative pipe sizes in the example feedwater system. The nominal thickness for both pipe sizes correspond to schedule 80. Table 3E.6-4 shows, for example purposes, the stress magnitudes for each pipe size due to pressure, weight, thermal expansion and SSE loads. Only the pressure weight and thermal expansion stresses are used in the leak rate evaluation, where a sum of all stresses is used in the instability load and critical flaw evaluation.

3E.6.2.6 LBB Margin Evaluation

The incoming water of the feedwater system is in a subcooled state. Accordingly, the leakage flaw length calculations are based on the procedure outlined in Section 3E.4.1. The saturation pressure, P_{sat}, for each pipe size is calculated from the normal operation temperatures given in Table 3E.6-3. The leak rates are calculated as a function of crack length. The leakage flaw lengths corresponding to the reference leak rate (see Section 3E.5) are then determined.

The calculations for the critical flaw size and the instability load corresponding to leakage size cracks is performed using the J-T methodology. Specifically, the J-T curve shown in Figure 3E.2-9 and the Ramberg-Osgood parameters given in Subsection 3E.3.2.2 are used. Table 3E.6-5 shows the example presentation of calculated critical crack sizes, and the margins along with the instability load margins for the leakage size cracks. Results are shown for both the 22-inch and 12-inch lines. It is noted that the critical crack size margin is greater than 2 and the instability load margin also exceeds $\sqrt{2}$.

3E.6.2.7 Conclusion

For the example feedwater piping, based upon the reference leakage rate and assumed stress magnitudes, leakage flaw lengths are calculated for 22-inch and 12-inch lines. Comparison with critical crack lengths shows margin to be greater than 2. Also, the leak-size crack stability evaluation shows a margin of at least $\sqrt{2}$.

It is also demonstrated that the feedwater line meets other LBB criteria of Subsection 3.6.3.2 including immunity to failure from effects of IGSCC, water hammer and thermal fatigue. Therefore, the feedwater lines qualify for LBB behavior.

Table 3E.6-1

STRESSES IN THE MAIN STEAM LINES
(Assumed for example)

Nominal Pipe Size (in)	Pipe O.D. (in)	Nominal Thickness (in)	Long. Pressure Stress (ksi)	Weight + Thermal Expansion Stress (ksi)	SSE Stress (ksi)
28	28.0	1.32	5.17	3.0	5.0

Table 3E.6-2

CRITICAL CRACK LENGTH AND INSTABILITY LOAD MARGIN EVALUATIONS FOR MAIN STEAM LINES (Example)

Pipe Size (in)	Reference Leak Rate (gpm)	Reference Leakage Crack Length (in)	Critical Crack Length (in)	Instability[1] Bending Stress, S_b (ksi)	Margins on Critical Crack	Margins on Load[2] at Leakage Crack
28	10[3]	13.45	30.7	24.2	2.3	2.2

Notes:

1. Based on Equation 3E.3-9a
2. Based on Equation 3E-9b.
3. See Section 3E.5.

Table 3E.6-3

DATA FOR FEEDWATER SYSTEM PIPING (EXAMPLE)

Nominal Pipe Size (in)	Pipe O.D. (in)	Nominal Thickness (in)	Nominal Temperature (°F)	Operating Pressure (psig)
12	12.75	0.687	420	1100
22	22.0	1.031	420	1100

Table 3E.6-4

STRESSES IN FEEDWATER LINES (ASSUMED FOR EXAMPLE)

Nominal Pipe Size (in)	Logitudinal Pressure Stress (ksi)	Weight + Thermal Expansion Stress (ksi)	Safe Shut-down Earthquake (SSE) Stress (ksi)
12	5.1	4.0	5.0
22	5.4	4.0	5.0

Table 3E.6-5

CRITICAL CRACK LENGTH AND INSTABILITY LOAD MARGIN EVALUATIONS FOR FEEDWATER LINES (EXAMPLE)

Pipe Size (in)	Reference Leak Rate (gpm)	Reference Leakage Crack Length (in)	Critical Crack Length (in)	Instability[1] Bending Stress, S_b (ksi)	Margins on Critical Crack	Margins on Load[2] at Leakage Crack
12	10^3	5.7	13.1	24.0	2.3	2.1
22	10^3	6.7	20.4	25.6	3.1	2.2

Notes:

1. Based on Equation 3E.3-9a
2. Based on Equation 3E-9b.
3. See Section 3E.5.

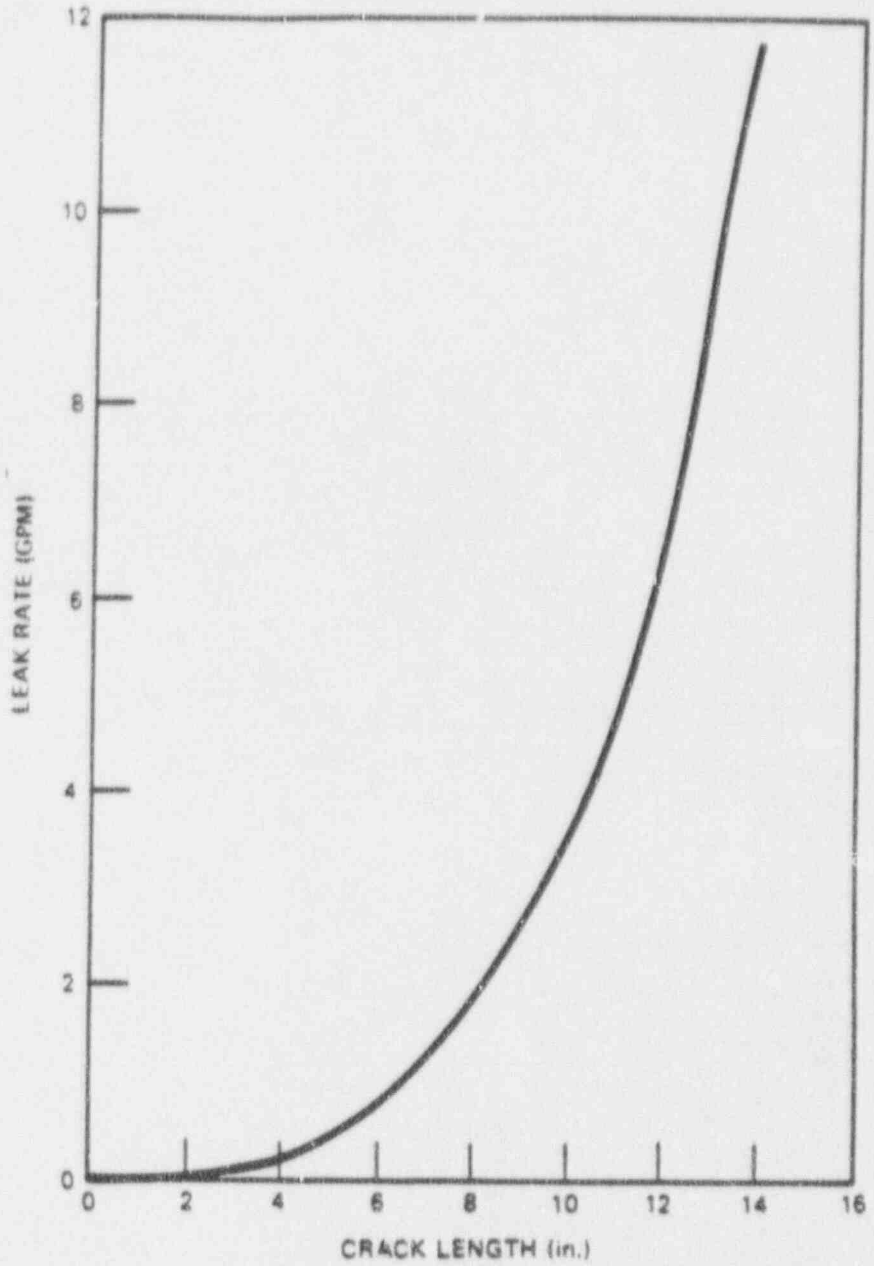

Figure 3E.6-1 LEAK RATE AS A FUNCTION OF CRACK LENGTH IN MAIN STEAM PIPE (EXAMPLE)

APPENDIX 3F
DELETED

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

E51 Reactor Core Isolation Cooling System (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F701	1	Pump suction line pressure instrumentation instrument root valve	2	B	P	E1		5.4-8a
F702	1	Pump discharge line pressure instrumentation instrument root valve	2	B	P	E1		5.4-8a
F703	1	Pump discharge line pressure instrumentation instrument root valve	2	B	P	E1		5.4-8a
F704	1	Pump discharge line pressure instrumentation instrument root valve	2	B	P	E1		5.4-8a
F705	1	Pump discharge line pressure instrumentation instrument root valve	2	B	P	E1		5.4-8a
F706	1	Pump discharge line flow instrument root valve	2	B	P	E1		5.4-8a
F707	1	Pump discharge line flow instrument root valve	2	B	P	E1		5.4-8a
F708	1	Pump discharge line flow instrument root valve	2	B	P	E1		5.4-8a
F709	1	Pump discharge line flow instrument root valve	2	B	P	E1		5.4-8a
F710	1	Pump discharge line pressure instrument root valve	2	B	P	E1		5.4-8a
F711	1	Pump discharge line pressure instrument root valve	2	B	P	E1		5.4-8a
F712	1	Turbine accessories cooling water line instrument root valve	2	B	P	E1		5.4-8c
F713	1	Turbine accessories cooling water line instrument root valve	2	B	P	E1		5.4-8c
F714	1	Turbine accessories cooling water line instrument root valve	2	B	P	E1		5.4-8c
F716	1	Steam supply line pressure instrument root valve	2	B	P	E1		5.4-8b
F717	1	Steam supply line pressure instrument root valve	2	B	P	E1		5.4-8b
F718	1	Steam supply line drain pot instrument root valve	2	B	P	E1		5.4-8b
F719	1	Steam supply line drain pot instrument root valve	2	B	P	E1		5.4-8b
F720	1	Steam supply line drain pot instrument root valve	2	B	P	E1		5.4-8b
F721	1	Steam supply line drain pot instrument root valve	2	B	P	E1		5.4-8b
F722	1	Turbine exhaust pressure instrument root valve	2	B	P	E1		5.4-8c

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

E51 Reactor Core Isolation Cooling System Valves (Continued)

No.	Quan	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F723	1	Turbine exhaust pressure instrument root valve	2	B	P	E1		5.4-8c
F724	1	Turbine exhaust pressure between rupture disk instrument root valve	2	B	P	E1		5.4-8c
F725	1	Turbine exhaust pressure between rupture disk instrument root valve	2	B	P	E1		5.4-8c
D014	1	Turbine exhaust pressure rupture disk	2	D	A	Rplc.	5 yrs	5.4-8c
D015	1	Turbine exhaust pressure rupture disk	2	D	A	Rplc.	5 yrs	5.4-8c

G31 Reactor Water Cleanup System Valves

No.	Quan	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F001	1	Line inside containment from RHR system maintenance valve	1	B	P	E1		5.4-12a
F002	1	CUW System suction line inboard isolation valve	1	A	I A	L P,S	2 yrs 3 mo	5.4-12a
F003	1	CUW System suction line outboard isolation valve	1	A	I,A	L P,S	2 yrs 3 mo	5.4-12a
F017	1	CUW System RPV head spray line outboard isolation valve	1	A	I,A	L P,S	2 yrs 3 mo	5.4-12a
F018	1	CUW System RPV head spray line inboard check valve	1	C	I,A	L P,S	2 yrs 3 mo	5.4-12a
F019	1	CUW Sys bottom head drain line maintenance valve	1	B	P	E1		5.4-12a
F050	1	Test line off the suct line outboard isolation valve G31-F003	2	B	P	E1		5.4-12a
[illegible]	[illegible]	Test line off RPV head spray line outboard isolation valve	2	B	P	E1		5.4-12a
F060	[illegible]	RPV bottom head drain line sample line test line valve	2	B	P	E1		5.4-12a
F070	1	RPV bottom head drain line sample line maintenance valve	2	B	P	E1		5.4-12a
F071	1	RPV bottom head drain line sample line vlv	2	A	I,A	L P,S	2 yrs 3 mo	5.4-12a
F072	1	RPV bottom head drain line sample line vlv	2	A	I,A	L P,S	2 yrs 3 mo	5.4-12a
F500	1	CUW Sys bottom head drain line drain vlv	2	B	P	E1		5.4-12a

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

G31 Reactor Water Cleanup System Valves (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F501	1	CUW Sys bottom head drain line drain vlv	2	B	P	E1		5.4-12a
F700	2	CUW System suction line FE upstream instrument root valve	2	B	I,P	E1		5.4-12a
F701	2	CUW System suction line FE downstream instrument root valve	2	B	I,P	E1		5.4-12a
F702	2	CUW System suction line FE upstream instrument root valve	2	B	I,A	LS	2 yrs	5.4-12a
F703	2	CUW System suction line FE downstream instrument root valve	2	B	I,A	LS	2 yrs	5.4-12a

G41 Fuel Pool Cooling and Cleanup Valves

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F015	2	FPC system heat exchanger outlet line maintenance valve	3	B	P	E1		9.1-1b
F016	1	FPC system discharge line to spent fuel pool check valve	3	C	A	P,S	3 mo	9.1-1b
F017	1	FPC system discharge line to spent fuel pool maintenance valve	3	B	P	E1		9.1-1b
F018	1	FPC system discharge line to spent fuel pool check valve	3	C	A	P,S	3 mo	9.1-1b
F019	2	FPC system discharge line to spent fuel pool valve	3	B	P	E1		9.1-1a
F020	2	FPC system discharge line to spent fuel pool check valve	3	C	A	P,S	3 mo	9.1-1a
F022	1	FPC system discharge line to reactor well maintenance valve	3	B	P	E1		9.1-1b
F023	1	FPC system discharge line to reactor well maintenance valve	3	B	P	E1		9.1-1b
F091	1	FPC system supply line from SPCU check vlv	3	C	P	P,S	3 mo	9.1-1b
F093	1	FPC system RHR return line valve to FPC	3	B	P	E1		9.1-1b
F094	1	FPC system RHR return line check valve to FPC	3	C	P	P,S	3 mo	9.1-1b
F095	1	FPC system discharge line to spent fuel pool sample line	3	B	P	E1		9.1-1b
F506	1	FPC system line valve from RHR-to-FPC line to LCW	3	B	P	E1		9.1-1b

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

G51 Suppression Pool Cleanup System Valves

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para. (e)	Test Freq. (f)	SSAR Fig.
F001	1	SPCU suction Line inboard isolation valve	2	A	I,A	L	2 yrs	9.5-1
						P,S	3 mo	
F002	1	SPCU suction line outboard isolation valve	2	A	I,A	L	2 yrs	9.5-1
						P,S	3mo	
F006	1	SPCU return line isolation valve	2	A	I,A	L	2yrs	9.5-1
						P,S	3 mo	
F007	1	SPCU return line isolation valve	2	A	I,A	L	2 yrs	9.5-1
						P,S	3 mo	

K17 Radwaste System Valves

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para. (e)	Test Freq. (f)	SSAR Fig.
F003	1	Drywell LCW sump pump disch. line isolation valve	2	B	I,A	P	2 yrs	11.2-2cc
F004	1	Drywell LCW sump pump disch. line isolation valve	2	B	I,A	P	2 yrs	11.2-2cc
F103	1	Drywell HCW sump pump disch line isolation valve	2	B	I,A	P	2 yrs	11.2-2cc
F104	1	Drywell HCW sump pump disch line isolation valve	2	B	I,A	P	2 yrs	11.2-2cc

P11 Makeup Water (Purified) System Valves

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para. (e)	Test Freq. (f)	SSAR Fig.
F141	1	Outboard isolation valve	2	A	I,P	L	2 yrs	9.2-5b
F142	1	Inboard isolation valve	2	A	I,P	L	2 yrs	9.2-5b

P21 Reactor Building Cooling Water System Valves

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para. (e)	Test Freq. (f)	SSAR Fig.
F001	6	Pump discharge line check valve	3	C	A	E2		9.2-1a,d,g
F002	6	Pump discharge line maintenance valve	3	B	P	E1		9.2-1a,d,g
F003	6	Heat exchanger inlet line valve	3	B	P	E1		9.2-1a,d,g
F004	6	Heat exchanger outlet line MOV	3	B	P	E1		9.2-1a,d,g
F005	3	Cold water line to hot/cold water blender	3	B	P	E1		9.2-1a.d,g
F006	3	Hot/cold water blender valve - cold water	3	B	P	E1		9.2-1a,d,g
F007	3	Hot/cold water blender outlet line valve	3	B	A	E2		9.2-1a,d,g
F008	3	Hot/cold water blender cold water byps line	3	B	P	E1		9.2-1a,d,g
F009	3	Hot water line to hot/cold water blender	3	B	P	E1		9.2-1a,d,g
F010	3	Hot/cold water blender valve - hot water	3	B	P	E1		9.2-1a,d,g
F011	3	Hot/cold water blender hot water bypass line	3	B	A	E2		9.2-1a,d,g

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

P21 Reactor Building Cooling Water System Valves (Continued)

No.	Quan	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F012	3	Cooling water supply line to RHR System maintenance valve	3	B	P	E1		9.2-1b,e,h
F013	3	Cooling wtr return line from RHR Sys MOV	3	B	A	P S	2yrs 3 mo	9.2-1b,e,h
F014	3	Cooling water return line from RHR Hx maintenace valve	3	B	P	E1		9.2-1b,e,h
F015	6	Pump suction line maintenance valve	3	B	P	E1		9.2-1a,d,g
F016	3	Surge tank outlet line to RCW pump suction	3	B	P	E1		9.2-1b,e,h
F017	3	Surge tank make-up water line from SPCU	3	B	P	E1		9.2-1b,e,h
F018	3	Surge tank make-up water line from SPCU	3	B	P	P	2 yrs	9.2-1b,e,h
F019	3	Surge tank make-up from MUWP	3	B	P	P	2 yrs	9.2-1b.e.h
F020	3	Surge tank make-up water line from MUWP	3	B	P	E1		9.2-1b,e,h
F021	3	Chemical addition tank inlet line valve	3	B	P	E1		9.2-1a,d,g
F022	3	Chemical addition tank outlet line valve	3	B	P	E1		9.2-1a,d,g
F024	6	Cooling water supply line to HECW refrigator maintenance valve	3	B	P	E1		9.2-1b,e,h
F025	6	Cooling wtr supply line to HECW refrig PCV	3	B	A	E2		9.2-1b,e,h
F026	6	Cooling water supply line to HECW refrigator maintenance valve	3	B	P	E1		9.2-1b,e,h
F027	6	Cooling water line to HECW refrigator bypass line	3	B	P	E1		9.2-1b,e,h
F028	6	Cooling water return line from HECW refrig	3	B	P	E1		9.2-1b,e,h
F029	2	Cooling water supply line to FPC HX	3	B	P	E1		9.2-1b,e,
F030	2	Cooling water return line from FPC HX	3	B	P	E1		9.2-1b,e
F031	2	Cooling water supply line to FPC pump room air conditioning	3	B	P	E1		9.2-1b,e
F032	2	Cooling wtr return line from FPC pump room air conditioner	3	B	P	E1		9.2-1b,e
F033	2	Cooling wtr line to PCV Atmos Monit Sys clr	3	B	P	E1		9.2-1b,e
F034	2	Return line from PCV Atmos Monit Sys clr	3	B	P	E1		9.2-1b,e
F035	2	Cooling wtr supply line to SGTS rm air cond.	3	B	P	E1		9.2-1b,e
F036	2	Cooling water return line fr SGTS room air conditioner	3	B	P	E1		9.2-1b,e
F037	2	Cooling water supply line to FCS room air conditioner	3	B	P	E1		9.2-1b,e
F038	2	Cooling water return line fr FCS room air conditioner	3	B	P	E1		9.2-1b,e
F039	3	Cooling water supply line to RHR equipment room air conditioner	3	B	P	E1		9.2-1b,e,h
F040	3	Cooling water return line from RHR equipment room air conditioner	3	B	P	E1		9.2-1b,e,h
F041	3	Cooling water supply line to RHR pump mtr	3	B	P	E1		9.2-1b,e,h

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

P21 Reactor Building Cooling Water System Valves (Continued)

No.	Qty	Description	Safety Class (g)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F042	3	Cooling water return line fr RHR pump mtr	3	B	P	E1		9.2-1b,e,h
F043	3	Clng wtr sply line to RHR pump mech seals	3	B	P	E1		9.2-1b,e,h
F044	3	Clng wtr return line fr RHR pump mech seals	3	B	P	E1		9.2-1b,e,h
F045	1	Cooling water supply line to RCIC equipment room air conditioner	3	B	P	E1		9.2-1b
F046	1	Cooling water supply line from RCIC equipment room air conditioner	3	B	P	E1		9.2-1b
F047	2	Cooling water supply line to HPCF equipment room air conditioner	3	B	P	E1		9.2-1e,h
F048	2	Cooling water supply line from HPCF equipment room air conditioner	3	B	P	E1		9.2-1e,h
F049	2	Cooling water supply line to HPCF pump motor bearing	3	B	P	E1		9.2-1e,h
F050	2	Cooling water return linr from HPCF pump motor bearing	3	B	P	E1		9.2-1e,h
F051	2	Cooling water supply line to HPCF pump mechanical seals	3	B	P	E1		9.2-1e,h
F052	2	Cooling water return from HPCF pump mechanical seals	3	B	P	E1		9.2-1e,h
F053	2	Surge tank outlet line to HECW System	3	B	P	E1		9.2-1b,e
F055	6	Cooling water return line from Emer Diesel Generator	3	B B	A	P S	2 yrs 3 mo	9.2-1b,e,h
F056	3	Cooling water return line from Emer Diesel Generator	3	B	P	E1		9.2-1b,e,h
F057	2	Cooling water line to PCV Atmos Monitor System air conditioner	3	B	P	E1		9.2-1b,e
F058	2	Return line from PCV Atmos Monitor System air conditioner	3	B	P	E1		9.2-1b,e
F061	3	Cooling water line Emer Diesel Generators	3	B	P	E1		9.2-1b,e,h
F071	6	Cooling water supply line-to non-essential coolers	3	B	P	E1		9.2-1b,e,h
F072	6	Cooling water supply line-to non-essential coolers	3	B	A	P S	2 yrs 3 mo	9.2-1b,e,h
F075	2	Cooling water supply line to PCV iso valve	2	A	I,A	L ,P S	2 yrs 3 mo	9.2-1c,f
F076	2	Cooling water supply line to PCV iso valve	2	C	I,A	L,P S	2 yrs 3 mo	9.2-1c,f
F080	2	Cooling water return line fr PCV iso valve	2	A	I,A	L,P S	2 yrs 3 mo	9.2-1c,f
F081	2	Cooling water retrun line fr PCV iso valve	2	A	I,A	L,P S	2 yrs 3 mo	9.2-1c,f

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

P21 Reactor Building Cooling Water System Valves (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F083	3	Cooling water return line from non-essential coolers	3	C	A	S	Refuel	9.2-1b,c,h
F084	3	Cooling water return line fr contmt byps line	3	B	P	E1		9.2-1b,e,h
F175	3	Cooling water supply to RHR System HX pressure relief valve	3	C	P			9.2-1b,e,h
F220	6	Bypass line around RCW Sys otlt line MOV	3	B	P	E1		9.2-1a,d,g
F251	2	Cooling water supply line to PCV test line	2	B	P	E1		9.2-1c,f
F252	2	Cooling water return line fr PCV test line	2	B	P	E1		9.2-1e,f
F501	6	Heat exchanger shell side vent line	3	B	P	E1		9.2-1a,d,g
F502	6	Heat exchanger shell side drain line	3	B	P	E1		9.2-1a,d,g
F503	3	Surge tank drain line to SD.	3	B	P	E1		9.2-1b,e,h
F601	3	Cooling water supply line to RHR System drain line to SD	3	B	P	E1		9.2-1b,e,h
F602	3	Cooling water supply line to RHR System drain line to HCW	3	B	P	E1		9.2-1b,e,h
F603	3	Cooling water return line from RHR HX drain line to SD	3	B	P	E1		9.2-1b,e,h
F604	3	Cooling water return line from RHR HX drain line to HCW	3	B	P	E1		9.2-1b,e,h
F701	6	Pump discharge line press instr line	3	B	P	E1		9.2-1a,d,g
F702	6	HX discharge line sample line valve	3	B	P	E1		9.2-1a,d,g
F703	3	Cooling water supply line press instr line	3	B	P	E1		9.2-1a,d,g
F704	3	Cooling water supply line sample line valve	3	B	P	E1		9.2-1a,d,g
F705	3	Cooling water supply line elbow tap instr line	3	B	P	E1		9.2-1a,d,g
F706	3	Cooling water supply line elbow tap instr line	3	B	P	E1		9.2-1a,d,g
F707	3	Cooling wtr sply line to RHR Sys FT instr line	3	B	P	E1		9.2-1b,e,h
F708	3	Cooling wtr sply line to RHR Sys FT instr line	3	B	P	E1		9.2-1b,e,h
F709	3	Cooling wtr rtn line fr RHR HX sample line	3	B	P	E1		9.2-1b,e,h
F710	6	Pump suction line PX instr line	3	B	P	E1		9.2-1a,d,g
F711	6	Pump suction line press instr line	3	B	P	E1		9.2-1a,d,g
F712	3	Surge tank level instr root valve	3	B	P	E1		9.2-1b,e,h
F713	3	Surge tank level instr line root valve	3	B	P	E1		9.2-1b,e,h
F714	3	Surge tank level instr line root valve	3	B	P	E1		9.2-1b,e,h
F717	3	Cooling water line to DG instr line	3	B	P	E1		9.2-1b,e,h
F718	3	Return water line from DG instr line	3	B	P	E1		9.2-1b,e,h
F719	3	Cooling wtr line to DG instr line	3	B	P	E1		9.2-1b,e,h
F720	3	Return wtr line from DG instr line	3	B	P	E1		9.2-1b,e,h

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

P24 HVAC Normal Cooling Water System Valves

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F053	1	Outboard isolation valve	2	A	I,A	L,P	2 yrs	9.2-2b
						S	3 mo	
F054	1	Inboard isolation check valve	2	A	I,A	L	2 yrs	9.2-2b
F141	1	Return inboard isolation valve	2	A	I,A	L,P	2 yrs	9.2-2b
						S	3 mo	
F142	1	Return outboard isolation valve	2	A	I,A	L,P	2 yrs	9.2-2b
						S	3 mo	

P25 HVAC Emergency Cooling Water System Valves

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F001	6	Pump discharge line check valve	3	C	A	P	2 yrs	9.2-3a,b,c
						S,E2		
F002	6	Pump discharge line maintenace valve	3	B	P	E1		9.2-1a,b,c
F003	6	Refrig. outlet line maintenance valve	3	B	P	E1		9.2-1a,b,c
F004	2	Line to MCR cooling coil TCV maint vlv	3	B	P	E1		9.2-3a,b,c
F005	2	Disch line to MCR Clng coil Temp Cont Vlv	3	B	A	E2		9.2-1a,b,c
F006	2	Line to MCR cooling coil TCV maint vlv	3	B	P	E1		9.2-3a,b,c
F007	6	Disch line to MCR cooling maint valve	3	B	P	E1		9.2-3a,b,c
F008	6	Cooling coil return line to HECW maint vlv	3	B	P	E1		9.2-3a,b,c
F009	6	Pump suction line maintenance valve	3	B	P	E1		9.2-3a,b,c
F010	2	Disch line to MCR clng TCV byp line	3	B	P	E1		9.2-3a,b,c
F011	3	Pump suct line/disch line PCV maint vlv	3	B	P	E1		9.2-3a,b,c
F012	3	Pump suction line/disch line PCV	3	B	A	E2		9.2-3a,b,c
F013	3	Pump suction line/disch line PCV maint vlv	3	B	P	E1		9.2-3a,b,c
F014	3	Pump suct line/disch line PCV bypass line	3	B	P	E1		9.2-3a,b
F015	3	Line to C/B Essential Equip Rm maint vlv	3	B	P	E1		9.2-3a,b
F016	3	Line to C/B Essent Equip Rm temp Cont Vlv	3	B	A	E2		9.2-3a,b
F017	3	Line to C/B Essent Equip Rm maint valve	3	B	P	E1		9.2-3a,b
F018	6	Line to C/B Essent Equip Rm Maint valve	3	B	P	E1		9.2-3a,b
F019	6	C/B Essent Equip Rm return line maint vlv	3	B	P	E1		9.2-3a,b
F020	3	Line to C/B Essnt Equip Rm TCV byp ln vlv	3	B	P	E1		9.2-3a,b
F021	3	Line to DG cooling coil TCV maint vlv	3	B	P	E1		9.2-3a,b
F022	3	Disch line to DG cooling Temp Cont vlv	3	B	A	E2		9.2-3a,b
F023	3	Line to DG cooling coil TCV maint vlv	3	B	P	E1		9.2-3a,b
F024	6	Disch line to DG cooling coil maint vlv	3	B	P	E1		9.2-3a,b
F025	6	Disch line to DG cooling coil maint vlv	3	B	P	E1		9.2-3a,b
F026	3	Line to DG cooling coil TCV bypass line vlv	3	B	P	E1		9.2-3a,b
F030	3	Pump disch line to chemical addition tank	3	B	P	E1		9.2-3a,b
F031	3	Chemical addition tank return line valve	3	B	P	E1		9.2-3a,b

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

P25 HVAC Emergency Cooling Water System Valves (Continued)

No.	Quan	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F050	2	Make-up Water Purified (MUWP) line to pump suction	3	C	A	E2		9.2-3a,b
F070	6	Pump disch line drain line valve	3	B	P	E1		9.2-3a,b
F400	6	Pump drain line valve	3	B	P	E1		9.2-3a,b
F401	6	Pump bearing cooling wtr line needle vlv	3	B	P	E1		9.2-3a,b
F402	3	Refrig outlet line sample line valve	3	B	P	E1		9.2-3a,b
F406	3	Surge tank drain line valve	3	B	P	E1		9.2-3a,b
F700	6	Pump disch line pressure instr line	3	B	P	E1		9.2-3a,b
F701	6	FE P25-FE003 dwnstrm instr line	3	B	P	E1		9.2-3a,b
F702	6	FE P25-FE003 upstrm instr line	3	B	P	E1		9.2-3a,b
F703	6	Pump suction line PI instr line valve	3	B	P	E1		9.2-3a,b
F704	6	Pump suct/disch line dpt instr line vlv	3	B	P	E1		9.2-3a,b

P41 Reactor Service Water System Valves

No.	Quan	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F001	6	Pump discharge line check flow	3	C	A	E2		
F002	6	Pump discharge line mainttenance valve	3	B	P	E1		
F003	6	Inlet line to RCW System heat exchanger	3	B	A	E2		
F004	6	Inlet line to service water strainer	3	B	A	E2		
F005	6	Outlet line from RCW heat exchanger	3	B	A	E2		
F006	6	Service water strainer blowout line MOV	3	B	A	E2		
F007	6	Supply line from Domestic Water (DW) Sys	3	B	A	E2		
F010	6	RCW HX tube side (service wtr side) relief valve	3	C	P	E1		
F011	6	Bypass line around RCW HX outlet line MOV P41-F005	3	B	P	E1		
F012	3	Ferrous Ion Injection line to RSW pump discharge line	3	C	A	E2		
F014	3	Discharge line to discharge canal MOV	3	B	P	E1		
F401	6	RCW HX tube side drain line to SWSD at HX inlet	3	B	P	E1		
F402	6	RCW HX tube side drain line to SWSD at HX outlet	3	B	P	E1		
F403	6	RCW HX [illegible]side drain line to SWSD	3	B	P	E1		
F404	6	RCW HX tube side vent line to SWSD	3	B	P	E1		
F701	6	Pump discharge line pressure instr line	3	B	P	E1		
F702	3	Service water supply line pressure instr line	3	B	P	E1		
F703	6	Diff P across service water strainer upstream instrument line	3	B	P	E1		

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

P41 Reactor Service Water System Valves (Continued)

No.	Quan	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F704	6	Diff P across service water strainer downstream instrument line	3	B	P	E1		
F705	6	Diff P across RCW HX upstream instr line	3	B	P	E1		
F706	6	Diff P across RCW HX downstream instr line	3	B	P	E1		

P51 Service Air System Valves

No.	Quan	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F131	1	Outboard isolation manual valve	2	A	I,P	L	2 yrs	9.3-7
F132	1	Inboard isolation manual valve	2	A	I,P	L	2 yrs	9.3-7

P52 Instrument Air System Valves

No.	Quan	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F276	1	Outboard isoaltion valve	2	A	I,A	L	2 yrs	9.3-6
F277	1	Inboard isoaltion check valve	2	A,C	I,A	L	2 yrs	9.3-6

P54 High Pressure Nitrogen Gas Supply System Valves (Continued)

No.	Quan	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F002	4	Nitrogen bottles N2 supply line valve	3	B	P	E1		6.7-1
F003	2	Nitrogen bottles N2 supply line MOV	3	B	A	L,P	2 yrs	6.7-1
						S	3 mo	
F004	2	N2 bottle supply line PCV maint valve	3	B	P	E1		6.7-1
F005	2	N2 bottle supply line PCV	3	B	A	S	3 mo	6.7-1
F006	2	N2 bottle supply line PCV maint valve	3	B	P	E1		6.7-1
F007	2	Safety grade N2 supply line iso valve	2	A	I,A	P	2 yrs	6.7-1
						S	3 mo	
F008	2	Safety grade N2 supply line iso chk vlv	2	A,C	I,A	S	Refuel	6.7-1
F009	8	Safety grade N2 supply line to SRV	3	B	P	E1		6.7-1
F010	2	Bypass line around the N2 bottle supply line PCV	3	B	P	E1		6.7-1
F011	2	N2 bottle supply line relief valve	3	C	P	E1		6.7-1
F012	2	MOV at safety/non-safety boundary	3	A	A	P	2 yrs	6.7-1
						S	3 mo	
F200	1	Non-safety N2 supply line iso valve	2	A	I,A	P	2 yrs	6.7-1
						S	3 mo	
F209	1	Non-safety N2 supply line iso chk vlv	2	A,C	I,A	S	Refuel	6.7-1

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

T22 Standby Gas Treatment System Valves

No.	Quan	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F001	2	Fuel handling floor inlet butterfly valve	3	B	A	P	2 yrs	6.5-1
						S	3 mo	
F002	2	Dryer inlet butterfly valve	3	B	A	P	2 yrs	6.5-1
						S	3 mo	
F003	2	Dryer exhaust gravity damper	3	B	A	P	2 yrs	6.5-1
						S	3 mo	
F004	2	Filter train exhaust butterfly valve	3	B	A	P	2 yrs	6.5-1
						S	3 mo	
F006	1	Filter train R112 injection line valve	3	B	P	E1		6.5-1
F007	1	Filter train DOP injection line valve to pre HEPA filter	3	B	P	E1		6.5-1
F008	1	Filter train DOP sampling line valve downstream of pre HEPA	3	B	P	E1		6.5-1
F009	1	Filter train DOP sampling line valve downstream of pre HEPA	3	B	P	E1		6.5-1
F010	1	Filter train DOP injection line valve downstream of charcoal absorbent	3	B	P	E1		6.5-1
F011	1	Fliter train DOP sampling line valve downstream of charcoal absorbent	3	B	P	E1		6.5-1
F012	1	Filter train DOP sampling line valve downstream of after HEPA	3	B	P	E1		6.5-1
F014	1	STGS sample line valve	3	B	P	E1		6.5-1
F015	1	PRM discharge to stack valve	3	B	P	E1		6.5-1
F500	2	Dryer unit vent line valve	3	B	P	E1		6.5-1
F501	2	Dryer unit drain line valve	3	B	P	E1		6.5-1
F504	2	Dryer unit vent line valve	3	B	P	E1		6.5-1
F505	2	Exhaust fan vent line valve	3	B	P	E1		6.5-1
F506	1	Filter train vent line valve	3	B	P	E1		6.5-1
F507	1	Filter train vent line valve	3	B	P	E1		6.5-1
F508	1	Filter train vent line valve	3	B	P	E1		6.5-1
F509	1	Filter train vent line valve	3	B	P	E1		6.5-1
F510	1	Filter train vent line valve	3	B	P	E1		6.5-1
F511	1	Exhaust stack drain line valve	3	B	P	E1		6.5-1
F700	2	Dryer unit demister dp instrument line valve	3	B	P	E1		6.5-1
F701	2	Dryer unit demister dp instrument line valve	3	B	P	E1		6.5-1
F705	1	Filter train prefilter dp instrument line valve	3	B	P	E1		6.5-1
F706	1	Filter train prefilterdp instrument line valve	3	B	P	E1		6.5-1
F707	1	Filter train preHEPA dp instrument line valve	3	B	P	E1		6.5-1
F708	1	Filter train preHEPA dp instrument line valve	3	B	P	E1		6.5-1
F709	1	Fliter train charcoal absorber dp inst. line vlv	3	B	P	E1		6.5-1

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

T22 Standby Gas Treatment System Valves (Continued)

No.	Quan	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F710	1	Filter train charcoal absorber dp inst line vlv	3	B	P	E1		6.5-1
F711	1	Filter train after HEPA dp inst line valve	3	B	P	E1		6.5-1
F712	1	Filter train after HEPA dp inst line valve	3	B	P	E1		6.5-1
F713	2	Filter train exhaust flow instrument line valve	3	B	P	E1		6.5-1
F714	2	Filter train exhaust flow instrument line valve	3	B	P	E1		6.5-1

T31 Atmospheric Control System Valves

No.	Quan	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F001	1	N2 supply line from Reactor Building HVAC	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39a
F002	1	N2 supply line to drywell inboard containment isoaltion valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39a
F003	1	N2 supply line to wetwell inboard containment isoaltion valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39a
F004	1	Containment atmosphere exhaust line from drywell isoaltion valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39a
F005	1	Drywell atmosphere exhaust line valve T31-F004 bypass line	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39a
F006	1	Containment atmosphere exhaust line form wetwell isolation valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39a
F007	1	Wetwell overpressure line valve	2	A	P	L,P	2 yrs	6.2-39a
F008	1	Containment atmosphere exhaust line to SGTS	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39a
F009	1	Containment atmosphere exhaust line to R/B HVAC	2	A	I,A	L,P S	2 yrs 3 mo	6.2-[illegible]
F010	1	Drywell overpressure line valve	2	A	P	L,P	2 yrs	6.2-39a
F025	1	N2 supply line from K-5 outboard containment isolation valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39a
F039	1	N2 supply line from K-5 outboard containment isolation valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39a
F040	1	N2 supply line from K-5 to drywell inboard isoaltion valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39a
F041	1	N2 supply line from K-5 to wetwell inboard isoaltion valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39a
F044	8	Drywell/wetwell vacuum breaker valve	2	C	A	S	refuel	6.2-39b
F050	1	N2 supply line to drywell test line valve	2	B	P	E1		6.2-39a
F051	1	Containment atmosphere exhaust line test line valve	2	B	P	E1		6.2-39a
F054	1	Drywell personnel air lock hatch test line valve	2	B	P	E1		6.2-39b
F055	1	N2 supply line from test line valve	2	B	P	E1		6.2-39a

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

T31 Atmospheric Control System Valves (Continued)

No.	Quan	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F056	1	Wetwell personnel air lock hatch test line valve	2	B	P	E1		6.2-39b
F700	1	N2 supply line to drywell FE upstream instrument line	2	B	P	E1		6.2-39a
F701	1	N2 supply line to drywell FE downstream instrument line	2	B	P	E1		6.2-39a
F702	1	N2 supply line to wetwell FE upstream instrument line	2	B	P	E1		6.2-39a
F703	1	N2 supply line to wetwell FE downstream instrument line	2	B	P	E1		6.2-39a
F720	2	DW/WW vacuum breaker valve N2 supply line isolation valve	2	B	I,A	L,S	2 yrs	6.2-39b
F730	1	Drywell pressure instrument line isolation valve	2	B	I,P	L,S	2 yrs	6.2-39b
F731	1	Drywell pressure instrument line solenoid valve	2	B	P	E1		6.2-39b
F732	2	Drywell pressure instrument line iso valve	2	B	I,P	L,S	2 yrs	6.2-39b
F733	2	Drywell pressure instrument line solenoid valve	2	B	P	E1		6.2-39b
F734	4	Drywell pressure instrument line for NBS isolation valve	2	B	I,P	L,S	2 yrs	6.2-39b
F735	4	Drywell pressure instrument line for NBS solenoid valve	2	B	P	E1		6.2-39b
F736	2	Wetwell pressure instrument line iso valve	2	B	I,P	L,S	2 yrs	6.2-39b
F737	2	Wetwell pressure instrument line solenoid valve	2	B	P	E1		6.2-39b
F738	4	Suppression pool water level reference leg instrument line isolation valve	2	B	I,P	L,S	2 yrs	6.2-39b
F739	4	Suppression pool water level reference leg instrument line solenoid valve	2	B	P	E1		6.2-39b
F740	4	Suppression pool water level reference leg instrument line isolation valve	2	B	I,P	L,S	2 yrs	6.2-39b
F741	4	Suppression pool water level reference leg instrument line solenoid valve	2	B	P	E1		6.2-39b
F742	2	Suppression pool water level reference leg instrument line isolation valve	2	B	I,P	L,S	2 yrs	6.2-39b
F743	2	Suppression pool water level reference leg instrument line solenoid valve	2	B	P	E1		6.2-39b
F744	2	Suppression pool water level instrument line isoaltion valve	2	B	I,P	L,S	2 yrs	6.2-39b
F74[illegible]	2	Suppression pool water level instrument line solenoid valve	2	B	P	E1		6.2-39b

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

T31 Atmospheric Control System Valves (Continued)

No.	Quan	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F800	2	Drywell water level instrument line reference leg isolation valve	2	B	I,P	L,S	2 yrs	6.2-39b
F801	2	Drywell water level instrument line reference leg solenoid valve	2	B	P	E1		6.2-39b
F802	2	Drywell water level instrument line iso valve	2	B	I,P	L,S	2 yrs	6.2-39b
F803	2	Drywell water level instrumewnt line solenoid valve	2	B	P	E1		6.2-39b
F804	2	DW/WW differential pressure instrument line isolation valve	2	B	I,P	L,S	2 yrs	6.2-39b
F805	2	DW/WW diffrential pressure instrument solenoid valve	2	B	P	E1		6.2-39b
D001	1	Wetwell overpressure rupture disk	2	D	P	Rplc.	5 yrs	6.2-39a
D002	1	Drywell overpressure rupture disk	2	D	P	Rplc.	5 yrs	6.2-39a

T49 Flammability Control System Valves

No.	Quan	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F001	2	Inlet line from drywell inboard isolation valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-40
F002	2	Inlet line from drywell outboard isolation valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-40
F003	2	Flow control valve for the FCS inlet line fron drywell	3	B	A	P S	2 yrs 3 mo	6.2-40
F004	2	Blower bypass line flow control valve	3	B	A	P S	2 yrs 3 mo	6.2-40
F005	2	Blower discharge line to wetwell check valve	3	B	A	P S	2 yrs 3 mo	6.2-40
F006	2	Discharge line to wetwell outboard isolation valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-40
F007	2	Discharge line to wetwell inboard isolation valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-40
F008	2	Cooling water supply line from the RHR System MOV	3	A	A	P S	2 yrs 3 mo	6.2-40
F009	2	Cooling water supply line maintenance valve	3	B	P	P S	2 yrs 3 mo	6.2-40
F010	2	Cooling water supply line admission MOV	3	A	A	P S	2 yrs 3 mo	6.2-40
F012	2	Inlet line from drywell drain line valve	3	B	A	P S	2 yrs 3 mo	6.2-40
F013	2	Drain line from blower suction line	3	B	A	P S	2 yrs 3 mo	6.2-40
F014	2	Blower drain line valve	3	B	P	P S	2 yrs 3 mo	6.2-40

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

T49 Flammability Control System Valves (Continued)

No.	Quan	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F015	1	Blower discharge line to wetwell pressure relief valve	2	A	P	P S	2 yrs 3 mo	6.2-40
F016	1	Blower discharge line to wetwell pressure relief line check valve	2	A	A	P	2 yrs	6.2-40
F501	2	Inlet line from drywell test line valve	2	A	P	P	2 yrs	6.2-40
F502	2	Discharge line to wetwell test line valve	2	A	P	P	2 yrs	6.2-40
F504	2	Blower suction line test line valve	3	B	P	P	2 yrs	6.2-40
F505	2	Blower discharge line test line valve	3	B	P	P	2 yrs	6.2-40
F506	2	Drain line to Low Conductivity Waste (LCW) valve	3	B	P	P	2 yrs	6.2-40
F507	2	Cooling water supply line test line valve	3	B	P	P	2 yrs	6.2-40
F701	2	FE T49-FE002 upstream instrument line root valve	3	B	P	P	2 yrs	6.2-40
F702	2	FE T49-FE002 downstream instrument line root valve	3	B	P	P	2 yrs	6.2-40
F703	2	Blower suction line pressure instrument line root valve	3	B	P	P	2 yrs	6.2-40
F704	2	FE T49-FE004 upstream instrument line root valve	3	B	P	P	2 yrs	6.2-40
F705	2	FE T49-FE004 downstream instrument line root valve	3	B	P	P	2 yrs	6.2-40

U41 Heating, Ventilating and Air Conditioning System Valves

No.	Quan	Description	Safety Class	Code Cat.	Valve Func.	Test Para	Test Freq.
F001	2	Reactor area supply isolation valve	3	B	I,A	L,P,S	2 yrs
F002	2	Reactor area exhaust isolation valve	3	B	I,A	L,P,S	2 yrs
F003	2	FCS room supply isolation valve	3	B	I,A	P S	2 yrs 3 mo
F004	2	FCS room exhaust isolation valve	3	B	I,A	P S	2 yrs 3 mo
F005	2	FCS room connecting valve	3	B	P	S	2 yrs
Fxxx	2	CAMS emergency supply isolation damper	3	B	I,A	P S	2 yrs 3 mo
Fxxx	2	CAMS emergency exhaust isolation damper	3	B	I,A	P S	2 yrs 3 mo
Fxxx	4	Control room supply isolation valve	3	B	I,A	P S	2 yrs 3 mo
Fxxx	4	Control room exhaust isolation valve	3	B	I,A	P S	2 yrs 3 mo
Fxxx	4	Control room bypass line isolation valve	3	B	I,A	P S	2 yrs 3 mo
Fxxx	4	Emergency HVAC supply valves	2	B	A	P S	2 yrs 3 mo

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

NOTES:

(a) 1, 2, or 3 - Safety Classification, Subsection 3.2.3.

(b) Pump test parameters or exclusion per ASME Code, Section XI, Subsection IWP, ASME/ANSI OM Part 6:

- N - Speed
- DP - Differential Pressure
- P - Discharge Pressure
- Q - Flow Rate
- Vd - Peak-to-peak vibration displacement
- Vξ - Peak vibration velocity

- E10 - In regular use (Paragraph 5.3)
- E11 - Lacking required fluid inventory (Paragraph 5.5)

(c) A, B, C or D - Valve category per ASME Code Section XI, Subsection IWV, ASME/ANSI OM Part 10 (Paragraph 1.4).

(d) Valve function:

- I - Primary containment isolation, Subsection 6.2.4
- A or P - Active or passive per ASME Code in (c) above (Paragraph 1.3).

(e) Valve test parameters or exclusions per ASME Code in (c) above:

- L - Leakage rate (Paragraph 4.2.2)
- P - Local position verification (Paragraph 4.1)
- S - Stroke exercise Category A or B (Paragraph 4.2.1.1, 4.2.1.2)
  Category C (Paragraph 4.3.2.1, 4.3.2.2)

- E1- Used for operating convenience, i.e., passive vent, drain, instrument, test, maintenance valves, or system control pressure relief valves (Paragraph 1.2).
- E2- In regular use Category A Leakage (Paragraph 4.2.2.1)
  Category A or B, Stroke (Paragraph 4.2.1.5)
  Category C, Stroke (Paragraph 4.3.2.3)

(f) CS - Cold shutdown
RO - Refueling outage

Table 3.9-9

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

STANDBY LIQUID CONTROL SYSTEM	
C41-F006 A,B	Injection Valves
C41-F008	Inboard Check Valve
RESIDUAL HEAT REMOVAL SYSTEM	
E11-F005 A,B,C	Injection Valve Loops A,B&C
E11-F006 A,B,C	Testable Check Valve A,B&C
E11-F010 A,B,C	Shutdown Cooling Inboard Suction Isolation Valve Loops A,B&C
E11-F011 A,B,C	Shutdown Cooling Outboard Suction Isolation Valve Loops A,B&C
HIGH PRESSURE CORE FLOODER SYSTEM	
E22-F003 B,C	Injection Valve Loops B&C
E22-F004 B,C	Testable Check Valve Loops B&C
REACTOR CORE ISOLATION COOLING SYSTEM	
E51-F004	Injection Valve
E51-F005	Testable Check Valve

210.49

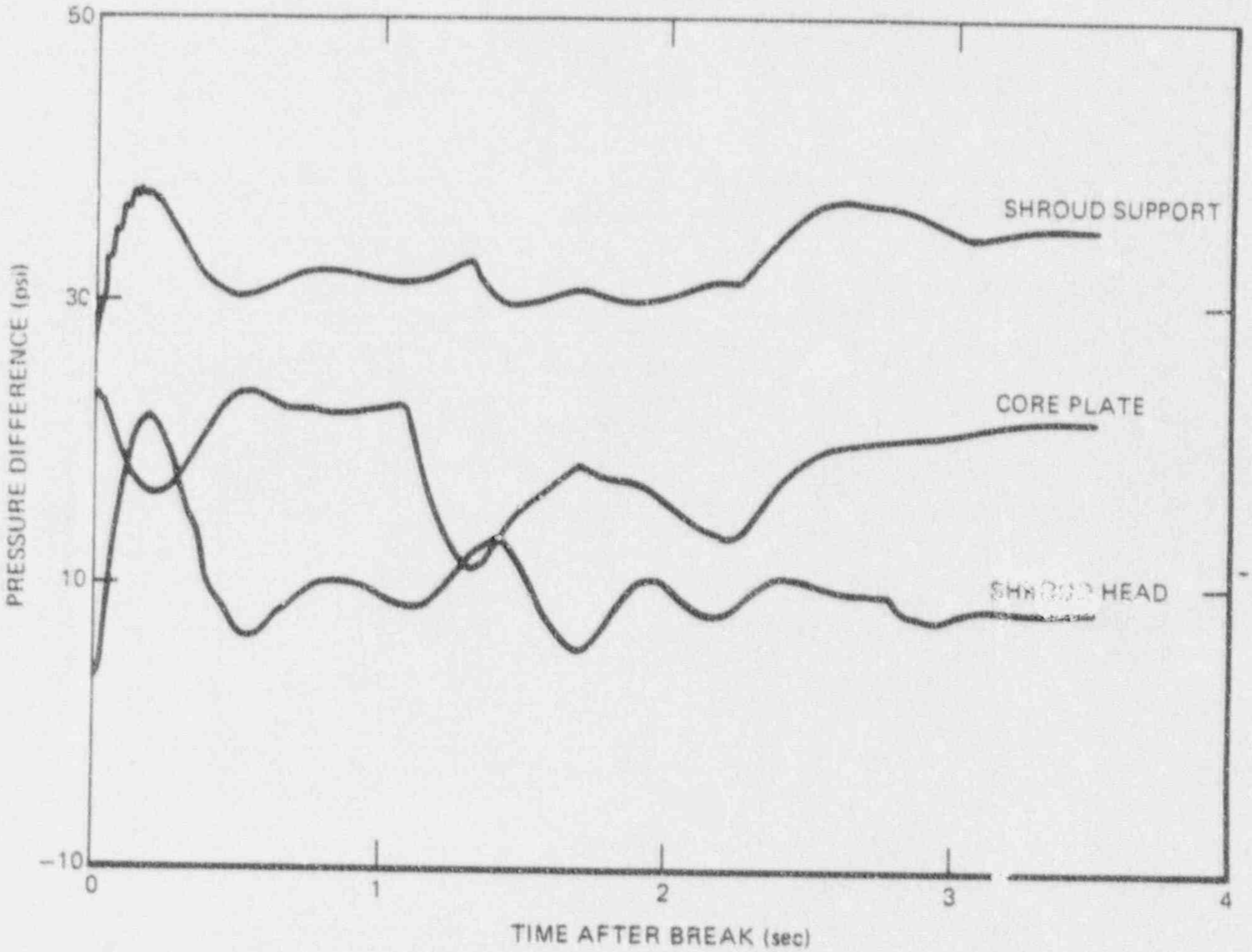

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Figure 3.9–1 TRANSIENT PRESSURE DIFFERENTIALS FOLLOWING A STEAM LINE BREAK

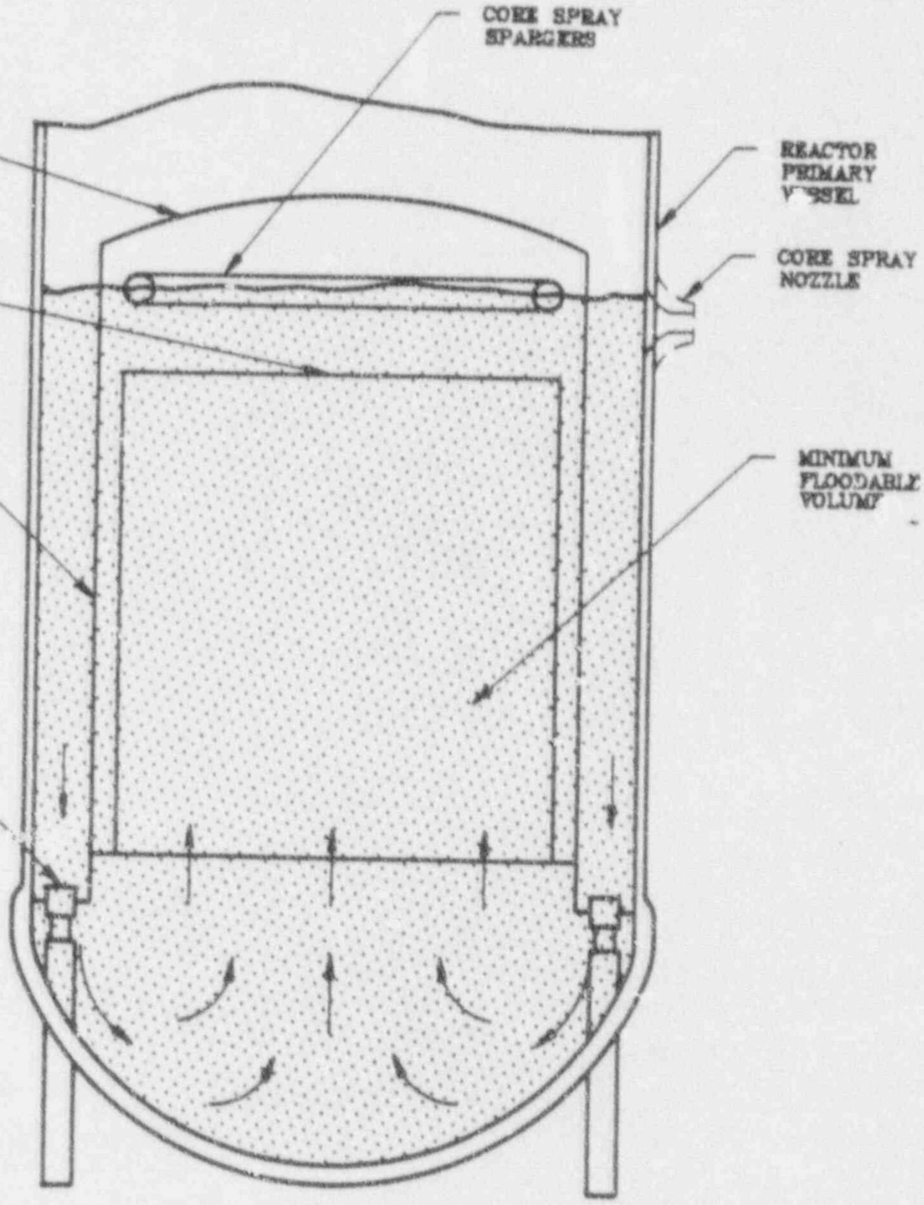

Figure 3.9-2 Reactor Internal Flow Paths and Minimum Floodable Volume

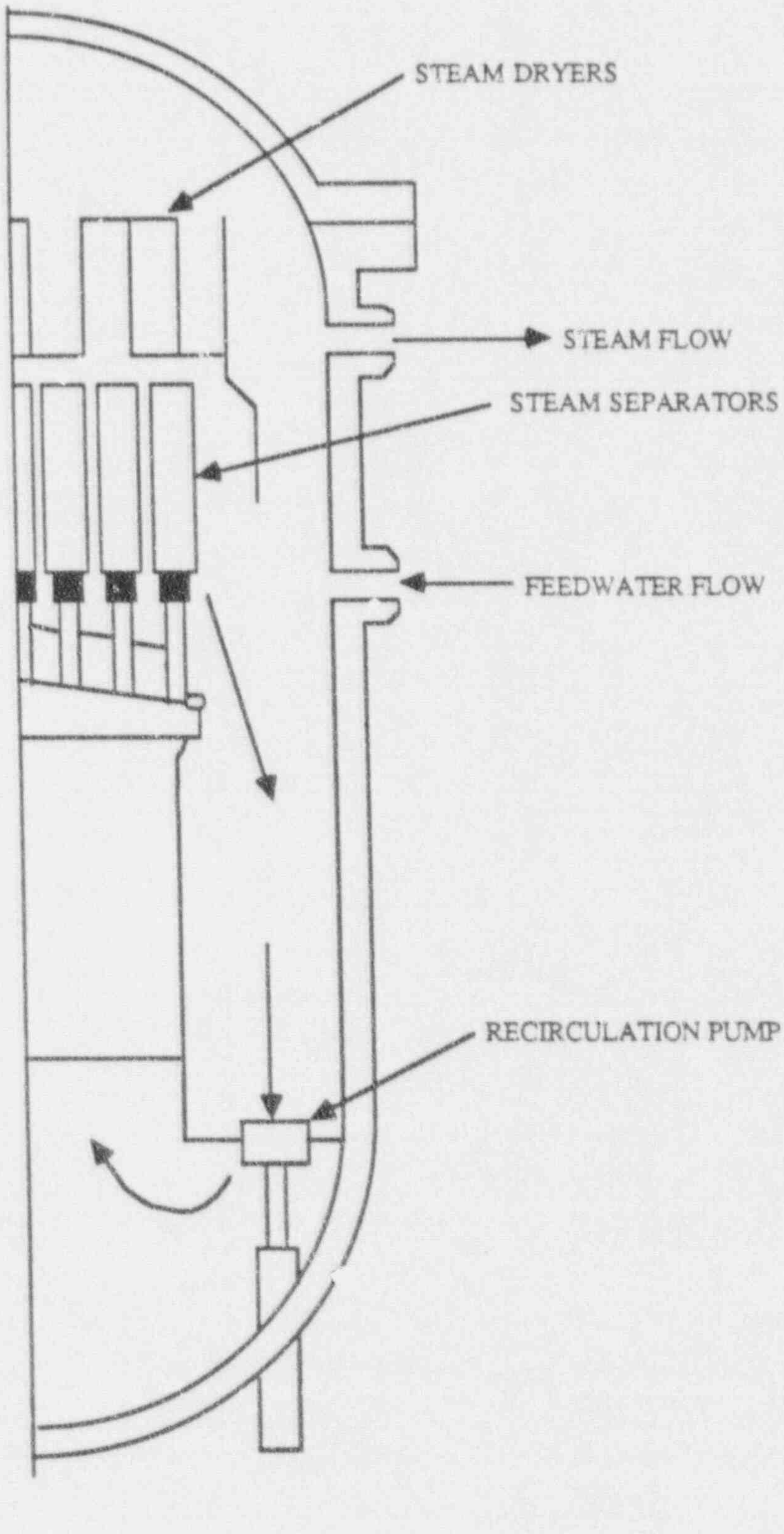

Figure 3.9-3 ABWR Recirculation Flow Path

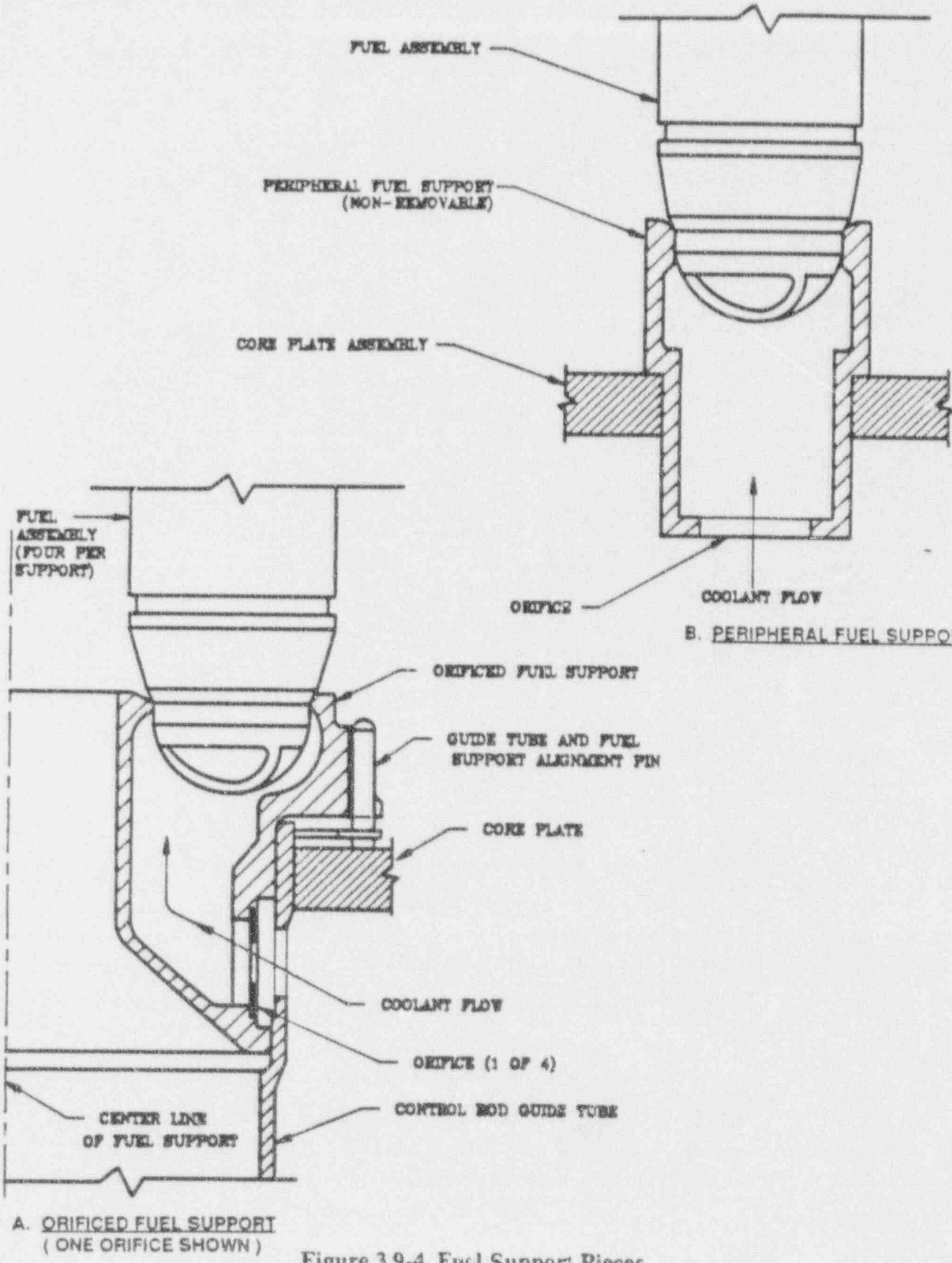

Figure 3.9-4 Fuel Support Pieces

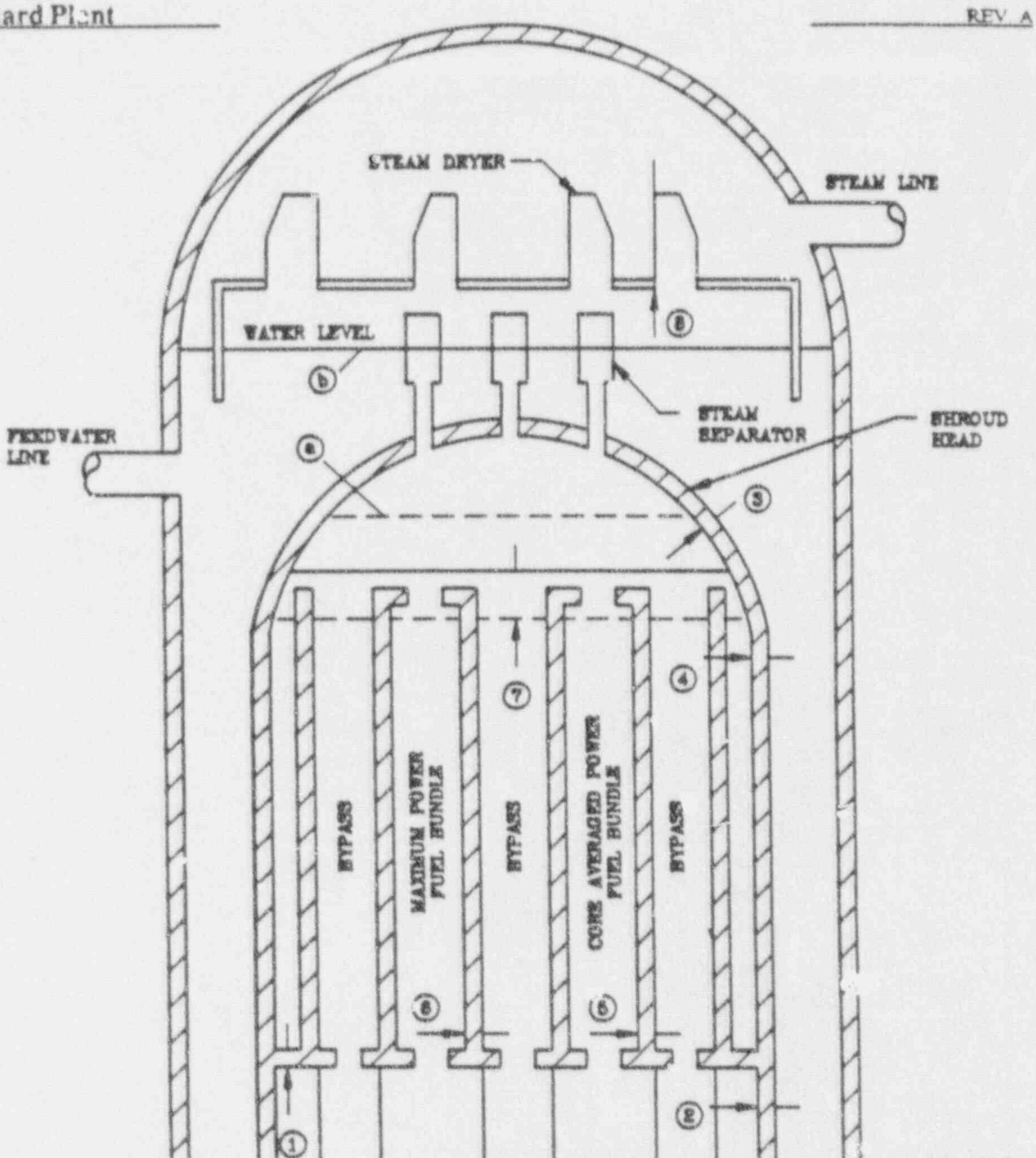

Figure 3.9-5 Pressure Nodes for Depressurization Analysis

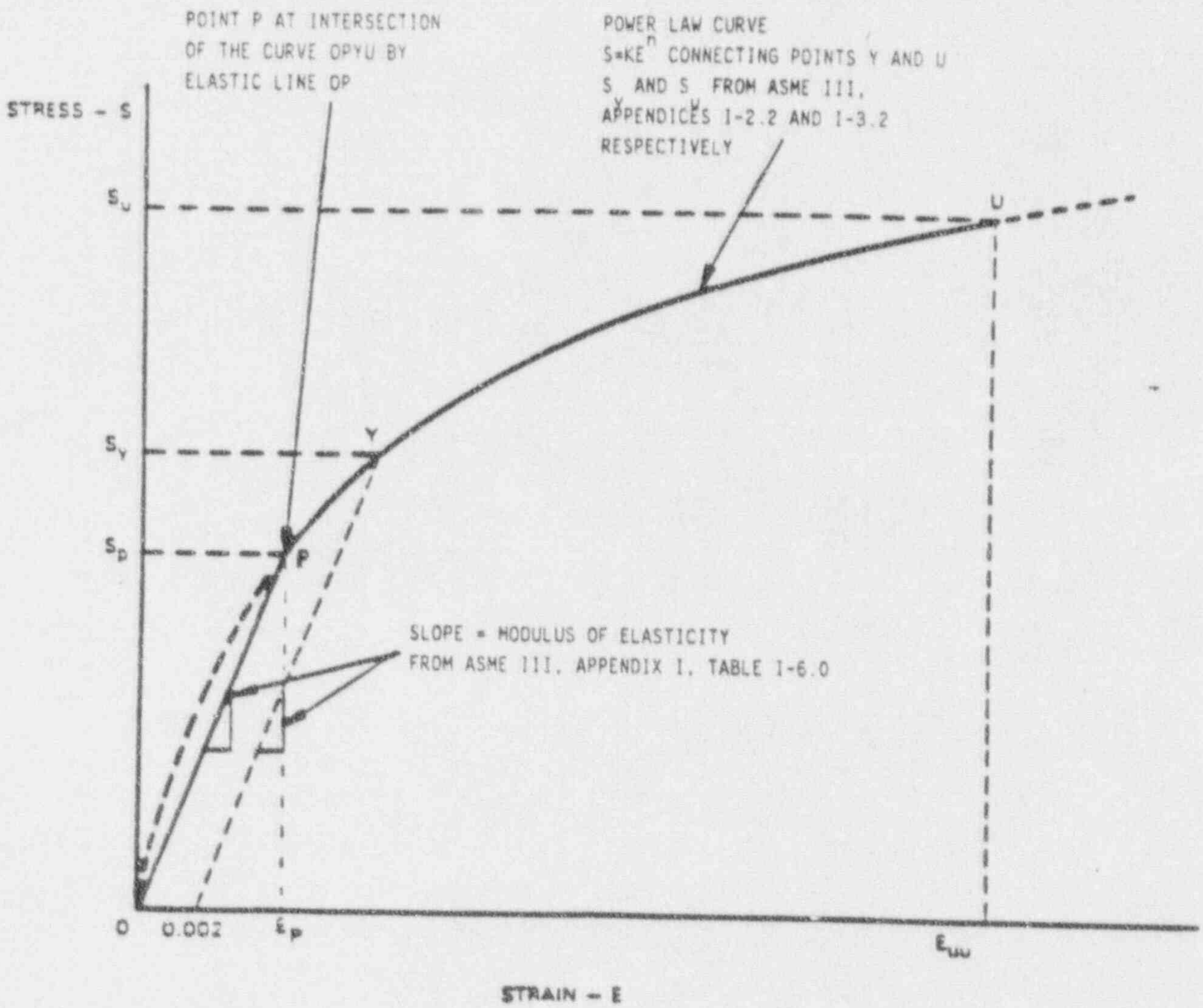

Figure 3.9-6 STRESS-STRAIN CURVE FOR BLOWOUT RESTRAINTS

APPENDIX 3D
COMPUTER PROGRAMS USFD IN THE DESIGN OF COMPONENTS, EQUIPMENT AND STRUCTURES

APPENDIX 3D

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APPENDIX 3D

TABLE OF CONTENTS (Continued)

3D.1 INTRODUCTION

As discussed in Subsection 3.9.1.2, this appendix describes the major computer programs used in the analysis of the safety-related components, equipment and structures. The quality of the programs and the computed results are controlled. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature including analytical results or numerical results to the benchmark problems.

The updates to Appendix 3D will be provided to indicate any additional programs used by GE and especially by vendors of components and equipment, or the later version of the described programs, and the method of their verification.

SECTION 3D.2

CONTENTS

3D.2 FINE MOTION CONTROL ROD DRIVE

3D.2.1 Fine Motion Control Rod Drive--FMCRD01

The program FMCRD01 is used to obtain scram performance data for various inputs to the fine motion control rod drive (FMCRD) stress analysis for both code and non-code parts. The use of this program is addressed in Subsection 3.9.1.3.2. Experimental data on pressure drops, friction factors, effects of misalignment, etc., are used in the setting up and perfecting of this code. Internal drive pressures and temperatures used in the stress analysis are also determined during actual testing of the prototype FMCRD.

3D.2.2 Structural Analysis Programs

Structural analysis programs, such as NASTRO4V and ANSYS, that are mentioned in Subsections 3D.3 and 3D.5 are used in the analysis of the FMCRD.

3D.3 REACTOR PRESSURE VESSEL AND INTERNALS

The following computer programs are used in the analysis of the reactor pressure vessel, core support structures, and other safety class reactor internals: NASTR04V, SAP4G07, HEATER, USAGE01, ANSYS, CLAPS, ASSIST, SEISM03 AND SASSI01. These programs are described in Subsection 4.1.4.

SECTION 3D.4

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