<u>LaSalle UNITS 1 AND 2</u>

<u>UFSAR, REVISION 23</u>

<u>AND</u>

<u>FIRE PROTECTION REPORT (FPR), REVISION 8</u>

THE SECURITY SENSITIVE INFORMATION HAS BEEN REDACTED FROM THE ORIGINAL DOCUMENT.

THIS DOCUMENT PROVIDES THE REDACTED VERSION.

THE REDACTED INFORMATION WITHIN THIS DOCUMENT IS INDICATED BY SOLID BLACKEDOUT REGIONS.

APPENDIX A

GLOSSARY

Appendix A of the FSAR contained a glossary defining key terminology and acronyms. This glossary is included in the UFSAR as Section 1.2.4.

APPENDIX B

CONFORMANCE TO REGULATORY GUIDES

B.0 INTRODUCTION

This Appendix to the UFSAR contains summary information on the Applicant's assessment of plant conformance to NRC Regulatory Guides 1.01 through 1-123, 1.145, 1.181, 1.183, 1.190, 1.194, 1.196, and 1.197. The following definitions are intended to assist the evaluator his interpretation of this Appendix:

> The phrase Assessed Capability in the Design identifies those Regulatory Guide provisions for which a design assessment was performed by the Applicant, through the A-E, NSSS supplier, and/or equipment vendors to determine the capability of the previously approved design to meet the Regulatory Guide requirements.
>
> The phrase Incorporated in the Design identifies those Regulatory Guide provisions which were included in the LaSalle County Station (LSCS) Units 1 and 2 design commitments during the construction permit review, or those provisions of more recently published Regulatory Guides which were subsequently included in the LSCS design.

Conformance to the provisions of the Regulatory Guides is indicated under either of two general categories, full compliance or compliance with the intent or objectives of UFSAR Regulatory Guide via the alternate approach cited in the UFSAR.

> Phrases indicating compliance with guidance mean that the provisions of the Regulatory Guides are met by direct conformance or by the assessed capability of the design.
>
> Phrases indicating compliance with the intent of Regulatory Guide provisions via an alternate approach, acknowledge the NRC Rules which state that "Regulatory Guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission." Where this method is employed, the description and justification of an alternate approach is provided. Engineering calculations and analytic details are not included in the UFSAR but are part of the design backup for the LSCS plant.

NRC letter of July 1, 1975 from P.C. Young to Byron Lee, Jr. acknowledged that the cut-off date for addressing Regulatory Guides for the LSCS Units 1 and 2 FSAR was 31 December 1975 (8 months prior to the initial tendering of the FSAR, which was 31 August 1976).

REGULATORY GUIDE 1.1 – REV. 0

NET POSITIVE SUCTION HEAD FOR EMERGENCY CORE COOLING AND CONTAINMENT HEAT REMOVAL SYSTEM PUMPS

Regulatory Guide 1.1 requires that containment overpressure not be relied upon to provide adequate net positive suction head (NPSH) for emergency core cooling and containment heat removal system pumps.

The emergency core cooling system pumps, which include the containment heat removal system pumps take suction from the suppression pool. Calculations for determining the available net positive suction head (NPSH) to these pumps are based on saturated water in the suppression pool, and will therefore be independent of containment pressure. The increase in containment pressures during a loss-of-coolant accident (LOCA) is not included in the determination of the available NPSH for these pumps. Available NPSB was determined by subtracting the suction line friction losses from the available static head. Pump performance tests verified that the available NPSH exceeds the required NPSH for rated flow conditions. Refer to Subsection 6.3.2.

Requirements for NPSH available at the centerline of the pump suction nozzles for each pump are given in Figures 6.3-1 (HPCS), 6.3-4 (LPCS), and 6.3-8 (LPCI). Pump characteristics curves are given in Figures 6.3-3 (HPCS), 6.3-6 (LPCS), and 6.3-9 (LPCI).

The requirements of this guide are incorporated in the LSCS design.

REGULATORY GUIDE 1.2 – REV. 0

THERMAL SHOCK TO REACTOR PRESSURE VESSELS

Regulatory Guide 1.2 states that potential reactor pressure vessel brittle fracture which may result from emergency core cooling system operation need not be reviewed in individual cases if no significant changes in presently approved core and pressure vessel designs are proposed. Should it be concluded that the margin of safety against reactor pressure vessel brittle failure due to emergency core cooling system operation is unacceptable, an engineering solution, such as annealing, could be applied to assure adequate recovery of the fracture toughness properties of the vessel material. This regulatory guide requires that available engineering solutions be outlined and requires that it be demonstrated that the design does not preclude their use.

An investigation of the structural integrity of boiling water reactor pressure vessels during a design—basis accident (DBA) has been conducted (refer to NEDO-10029, “An Analytical Study on Brittle Fracture of GE-BWR Vessel Subject to the Design Basis Accident”). It has been determined, based upon methods of fracture mechanics, that no failure of the vessel by brittle fracture as a result of a DBA will occur.

The investigation included:

a. a comprehensive thermal analysis considering the effect of blowdown and the low—pressure coolant injection (LPCI) system ref boding;

b. a stress analysis considering the effects of pressure, temperature, seismic load, jet load, dead weight, and residual stresses;

c. the radiation effect on material toughness (NDTT shift and critical stress intensity); and

d. methods for calculating crack tip stress intensity associated with a non-uniform stress field following the design-basis accident.

This analysis incorporated very conservative assumptions in all areas (particularly in the areas of heat transfer, stress analysis effects of radiation on material toughness, and crack tip stress intensity). Therefore, the results reported in NEDO-10029 provide an upper bound limit on brittle fracture failure mode studies. Because of the upper bound approach, it is concluded that catastrophic failure of the pressure vessel due to the DBA is shown to be impossible from a fracture mechanics point of view. In the case studies, even if an acute flaw does form on the vessel inner wall, it will not propagate as the result of the DBA.

The criteria of 10 CFR 50 Appendix G are interpreted as establishing the requirements for annealing. Paragraph IV C of Appendix G requires vessels to be designed for annealing of the beltline only where the predicted value of adjusted RT exceeds 200°F. This predicted value is not exceeded, therefore design for annealing is not required.

The reactor pressure vessels utilized for LSCS Units 1 and 2 employ no significant core or vessel design changes from previously approved BWR Pressure Vessels such as all units of Browns Ferry.

Reference Subsection 5.2.3.3.1.

The requirements of this guide are incorporated in the LSCS design.

REGULATORY GUIDE 1.3 – REV. 2

ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT FOR BOILING WATER REACTORS

Replaced by RG 1.183, Rev. 0.

REGULATORY GUIDE 1.4 – REV. 2

ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT FOR PRESSURIZED WATER REACTORS

This regulatory guide is not applicable to the LSCS.

REGULATORY GUIDE 1.5 – REV. 0

ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A STEAMLINE BREAK ACCIDENT FOR BOILING WATER REACTORS

Regulatory Guide 1.5 provides acceptable assumptions that may be used in evaluating the radiological consequences of a steamline break accident for boiling water reactors.

The applicant incorporated the guidelines given in this Regulatory Guide 1.5, and assumed a 5.5 second closure time for the steamline isolation valves.

Other key implementation assumptions used in the analyses for LSCS were as follows:

a. All Regulatory Position requirements included in the NRC solution.

b. Site Boundary as near as 509 m.

c. LPZ as near as 6400 in.

Some of the models and conditions that are prescribed are demonstrably inconsistent with actual physical phenomena. For this reason additional analyses are provided in Subsection 15.6.4 of the UFSAR which utilize realistic assumptions to demonstrate the Conservative bias in the regulatory guide requirements. In either case, regardless of model used for evaluation, the resultant dose is less than the specified regulatory limits.

Estimated doses from steamline breaks for ranges b and c above are dominated by estimated consequences from LOCA analysis.

Appendix C provides the solution for high energy line breaks outside the containment.

The provisions of this guide were incorporated in the analyses of potential radiological consequences from this type of accident.

INDEPENDENCE BETWEEN REDUNDANT STANDBY (ONSITE1 POWER SOURCES AND BETWEEN THEIR DISTRIBUTION SYSTEMS

Regulatory Guide 1.6 describes an acceptable degree of independence between redundant standby (onsite) power sources and between their distribution systems as required by GDC-17.

Conformance with Regulatory Guide 1.6 is described in the following subsections for each regulatory position of Paragraph D of that guide.

Independent Load Assignment

Each Class 1E load (ac or dc) is assigned to a Division 1, 2, or 3 load group. Assignment is determined by the nuclear safety functional redundance of the load.

Usually one (two at the most) division load group is required to perform nuclear safety functions. The loss of any single ESF division does not prevent the performance of the minimum safety functions required for a safe shutdown.

Independent A-C Sources

Each division a-c load group has a feed from ~ least one system auxiliary transformer (offsite) and one diesel generator (onsite), as shown in Figures 8.1-2 and 8.1-3. Divisions 1 and 2 are each provided with two off site power connections and Division 3 (HPCS) is provided with one off site power connection.

The diesel-generator breaker can be closed automatically only if the other source breakers to that load group are open, as shown in Figure 8.3-3.

Independence of D-C Sources

Each division d-c load group has a feed from one battery charger and one battery as shown in Figures 8.3-9, through 8.3-12.

The redundant d-c load groups cannot be connected to each other. The d-c batteries and battery chargers cannot be connected to each other. Each d-c load group of one unit can be connected to the corresponding nonredundant d-c load group of the second unit and satisfy the temporary design load.

Independence of Standby Sources

The diesel-generator breaker can be closed to its associated load group automatically only if the other source breakers to that load group are open, as shown in Figure 8.3-2.

When the diesel-generator breaker is closed, no other source breaker can be closed automatically, as shown in Figures 8.3-2 and 8.3-3. No other means exist for connecting redundant load groups with each other.

Each of the redundant load groups is fed from only one diesel generator, as shown in Figure 8.1-2. No means are provided for transferring loads between the diesel generators.

Sufficient interlocks are provided to prevent paralleling the diesel generators manually by operator error during loss of off site power. Figures 8.3-2 and 8.3-3, and Subsection 8.3.1.1.2 describe the interlocks and interlock defeat mechanisms that prevent paralleling sources, including diesel generators.

Prime Mover

Divisions 1, 2, and 3 diesel generators are provided with only one prime mover for each generator.

The HPCS standby power source and distribution system is redundant to the other two standby power sources and associated distribution systems in the plant.

No provisions exist for transferring loads from the HPCS power source to other power sources or transferring HPCS load groups either automatically or manually.

The provisions of this guide are incorporated into the LSCS design. Full compliance exists.

REGULATORY GUIDE 1.7 – REV. 0

CONTROL OF COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT

Regulatory Guide 1.7 requires the capability be provided for (1) measuring hydrogen concentration, (2) mixing the atmosphere in the containment following a loss-of-coolant accident, (3) purging of the containment atmosphere through appropriate fission product removal systems, and (4) controlling combustible gas concentrations without reliance on purging.

Information on the control of containment gases for the post-LOCA situation is included in Subsection 6.2.5.

The LSCS design incorporates the guidance set forth in this regulatory guide; we believe that we comply with this guide.

REGULATORY GUIDE 1.8 – REV. 1R

PERSONNEL SELECTION AND TRAINING

Regulatory Guide 1.8 identifies acceptable criteria for the selection and training of nuclear power plant personnel.

LaSalle County Station's training and personnel selection is described in Section 13.2.

We comply with the objectives set forth in the referenced revision of this regulatory guide, as indicated in the textual material mentioned above.

REGULATORY GUIDE 1.9 – REV. 3

SELECTION, DESIGN, QUALIFICATION AND TESTING OF DIESEL-GENERATOR UNITS USED AS CLASS 1E ONSITE ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS

Regulatory Guide (RG) 1.9 Revision 3 endorses IEEE Standard 387-1984, "IEEE Standard Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations". In addition to this standard RG 1.9 Rev. 3 provides supplemental regulatory positions. The applicant complies with these supplemental regulatory positions in Revision 3 with the following clarifications regarding:

1. Paragraph C.2.2. Exception is taken to the last sentence in this paragraph, "Jumpers and other nonstandard configurations or arrangements should not be used subsequent to initial equipment start up conditions". In order to successfully accomplish certain diesel tests it is necessary to use jumpers to simulate particular engine signals. The use of jumpers is a normal practice in diesel engine testing, and the safe use of the jumpers are ensured with detailed procedures.

2. Paragraph C.2.2.1 "Start Test". Each EDG undergoes a start up test on a monthly basis from "standby conditions" Once every six months this test is supplemented by verifying proper start up from "normal standby conditions". This test is covered in paragraph C.2.2.3, and is further discussed in Item 3 below.

3. Paragraphs C.2.2.1, C.2.2.4, C.2.2.5, and C.2.2.6. Exception has been taken against use of the term 'standby conditions' to denote 'normal standby conditions'.

   The term normal 'standby condition' is interpreted as any conditional state of the EDG in which the EDG is considered operable. More specifically, standby conditions for an EDG refer to a condition whereby the diesel engine lube oil is being continuously circulated and engine jacket water and lube oil temperatures are consistent with manufacturer's recommended operating range (low lube oil and jacket water temperature alarm settings to the high lube oil and jacket water temperature alarm settings).

   The term 'ambient standby condition' defines a conditional state of the EDG in which lube oil and jacket water temperatures are within the prescribed temperature bands of these subsystems when the EDG has been at rest for an extended period of time with the pre-lube oil and jacket water circulating systems operational. It should be noted that the semi-annual fast start test described in paragraph C.2.2.3 is performed from 'normal standby conditions'

4. Paragraph C.2.2.6, "Combined SIAS and LOOP Tests". Exception is taken to the statement that the EDG be tested for proper response to a loss-of-offsite power (LOOP) in conjunction with a safety injection actuation signal (SIAS) in whatever sequence they might occur. The EDGs are tested for response to a LOOP, to a SIAS and to a LOOP and SIAS when they occur concurrently. Performing a LOOP/SIAS EDG test in whatever sequence they might occur is beyond the Stations' original licensing basis, provides no additional value, and was not included in Regulatory Guide 1.9, Revision 2 or Regulatory Guide 1.108.

5. Paragraph C.2.2.7 and C.2.2.8, Single Load Rejection Test and "Full-Load Rejection Test". Exception is taken to the specified power factor requirements of this paragraph. Single and full load rejection tests have been successfully performed at rated power factor. During these tests the generator transient voltage approached the manufacturers recommended voltage limit. Such an event is considered to border on destructive testing and can cause premature generator aging. As a result, the station will continue to perform single load and full-load rejection tests; however, not at rated power factor.

6. Table 1, Refuel Outage frequency designation for Section 2.2.4 “Loss-of-Offsite-Power (LOOP) Test,” Section 2.2.6 “Combined SIAS and LOOP Tests,” Section 2.2.11 “Synchronizing Test,” and Section 2.2.12 “Protective-Trip Bypass Test.” The surveillance frequency for these tests shall be controlled under the Surveillance

Frequency Control Program (SFCP) in the same manner as the Technical Specification surveillance frequency requirement.

Regulatory Guide 1.108 has been superseded by Regulatory Guide 1.9 Revision 3. A majority of the guidance provided in RG 1.108 has been transferred to RG 1.9 Rev. 3. The following discussion contains the applicable responses to these sections of RG 1.9 Rev. 3.

The preoperational test program (FSAR Table 14.2-36) was designed to show functional compliance to the operational requirements set forth in this guide.

A separate demonstration test program was exercised on a one-time basis to show compliance with the availability requirements of this guide. The demonstration test of the LaSalle HPCS diesel generator (all LaSalle diesel generators are design equivalents) was in response to the NRC's request letter of December 17, 1976, O. D. Parr to GE's Dr. Sherwood. A proposed test program was accepted with minor comments and recommendations by the NRC staff by letter of March 31, 1978 from O. D. Parr to Dr. Sherwood. GE completed the tests with the staff recommendations and reported the results in NEDO-10905 (Amendment 3) which was submitted to O. D. Parr from J. F. Quirk (GE) on December 20, 1979. These test results and conclusions demonstrate the acceptability of this HPCS diesel generator configuration for most BWR-5 plants and all BWR-6 plants. Specific reference was made to the LaSalle HPCS diesel generator because the demonstration test was performed at LaSalle with LaSalle equipment.

The following objectives were demonstrated successfully:

1. Generator acceleration and bulk load acceptance within acceptable time and performance parameters.

2. Confirmation of analytic prediction models used for design and factory test of this diesel engine and load combination.

3. Established a 0.99 reliability for DG set starting through load acceptance (for the LaSalle diesel generator per se, and for identical units elsewhere).

4. Margins of 20% excess rated engine torque; attainment of max rated bhp in 19 seconds with an 8 second margin on the requirement; and a 10% margin on minimum load carrying capability without exceeding manufacturer's design limits. The long term continuous load rating was demonstrated to be 30%. This is the power margin for operation during accident conditions when the reactor pressure is high. Full test results are included in NEDO-10905-3 which is part of the LaSalle docket. Sixty-nine consecutive start-ups through load acceptance form both cold and hot engine conditions demonstrated the 0.99 performance reliability number for the LaSalle units.

From these two test programs the following test objectives are covered: initial start; loss of a-c power; rated load; load shedding; load transfer; and, stability of long term operation.

The demonstration test program of Paragraphs C.2.a and C.2.b of RG 1.108 (Paragraphs C.2.2 and C.2.3 of RG 1.9 Rev. 3) are not repeated every 18 months. However, it should be noted that the demonstration test program of paragraph C.2.a(9) of RG 1.108 (paragraph C.2.3.1 of RG 1.9 Rev. 3) regarding 69/n start test, is only a requirement during the plant preoperational test program and not required every 18 months was complied with only during preoperational testing. This clarification was made by a NRC errata sheet dated September 1977 to RG 1.108. By letter of April 7, 1980 from O. D. Parr to GE's Dr. Sherwood, the NRC concluded that these test results qualified the DG units for LaSalle. As a result of this qualification testing and the comprehensive pre-operational tests it is understood that Paragraphs C.2.a and C.2.b of RG 1.108 (Paragraphs C.2.2 and C.2.3 of RG 1.9 Rev. 3) are not a part of the normal operational surveillance requirement. The Edison alternate to these recent requirements promulgated under RG 1.108 (RG 1.9 Rev. 3) has been previously submitted with LaSalle Technical Specification.

Regulatory Guide 1.9 Rev. 2 has been superseded by Regulatory Guide 1.9 Revision 3. A majority of the guidance provided in RG 1.9 Rev. 2 has been transferred to RG 1.9 Rev.3. The following discussion contains the applicable responses to these sections of RG 1.9 Rev. 3.

UFSAR Chapter 8.0 shows that the 2000-hour rating of the diesel generator is greater than the maximum coincidental steady-state loads requiring power at any time. Intermittent loads such as motor-operated valves are not considered for long-term loads.

UFSAR Table 8.3-1 contains the maximum expected coincidental loads for each of the diesel-generator sets, for LOCA conditions, and for safe shutdown conditions The 2000-hour rating of each standby diesel generator is 2860 KW, and the 30-minute rating is 3040 KW (UFSAR Table 8.3-3).

UFSAR Chapter 8.0 illustrates that the 2000-hour rating of the HPCS diesel-generator, the 90% of 30-minute rating, and the maximum coincidental load, are in conformance with this guide.

During preoperational testing, the predicted standby diesel-generator loads for each ESF Division were verified, as well as the capability of the diesel generators to carry these loads.

Preoperational test results for the LaSalle Unit "0" and "1A" diesel generators were reported to the NRC on March 25, 1982. Initial test results indicated noncompliance with the underfrequency and undervoltage guidelines of Position C.4. RG 1.9 Rev. 2 (Position C.1.4 of RG 1.9, Rev. 3). Result indicated an undervoltage of 63.7% (2650 volts) and an underfrequency of 94.3% (56.5 Hz) and a frequency recovery of the load sequence interval of 82.5% versus the C.4-RG 1.9 Rev. 2 (C.1.4-RG 1.9 Rev. 3) guidelines of 75%, 95% and 60%, respectively.

Justification of the LaSalle diesel generator design is as follows:

1. The preoperational test used actual emergency core cooling system loads and demonstrated that the diesel generators have the capability to start and accelerate for all loads to rated speed within the required time period without failing.

2. A number of margin tests demonstrated that the diesel generators can accept a step load increase that is greater than the actual step load and can endure voltage drops of 60, 58.7, and 51.5 percent without experiencing instability resulting in generator voltage collapse or inability of the voltage to recover.

3. During load sequencing, no loads tripped due to undervoltage.

4. The voltage recovered to within 10% of the nominal voltage in less than 50% of the load sequence interval (approximately 2 seconds).

5. The frequency recovered to within 2% of nominal before the next load was applied automatically.

In addition, motors connected to the Class 1E buses that will be subject to the undervoltage transients, are designed to start and accelerate their loads with terminal voltages at 80%. The time interval that the bus voltage will be below 80% due to the starting transient is less than 1 second and will not appreciably reduce the overall reliability of the motors.

Based upon these justifications, the LaSalle County Station diesel generator performance demonstrates practical compliance with Position C.4 of RG 1.9 , Rev. 2 (Position C.1.4 of RG Rev. 3) A requirement was levied to monitor that the undervoltage trip signal is less than or equal to 3 seconds when subjected to a 40% voltage drop.

Position C.2.a.(8) of RG 1.108 (Position C.2.2 of RG 1.9, Rev. 3) allows the Division 1 and Division 2 diesel generators to be running in standby at 50 droop. Under this rare condition the Division 1 and 2 diesel generators do not meet the voltage requirements of RG 1.90 during the ECCS pump starts. A test of the Division 1 diesel generator (worst case) was done for two conditions. One test was a simultaneous ECCS initiation with bus undervoltage, while the diesel was in a test mode at 50 droop. The other test was an ECCS initiation signal with the diesel in a test mode at 50 droop, followed later by bus undervoltage. The tests verified the capability of the diesels to pick up the pump loads with no indication of any possible failures due to the voltage transient and recovery time.

The suitability of each standby diesel generator is confirmed by prototype qualification test data and by preoperational tests.

The HPCS diesel-generator unit is considered as a unique application, justifiable departure from the strict conformance to RG 1.9 regarding voltage and frequency limits during the initial loading transient. The HPCS system consists of one large pump and motor combination which represents more than 90% of the total load; consequently, limiting the momentary voltage drop to 25% and the momentary frequency droop to 5% would not significantly enhance the reliability of HPCS operation. To meet the specific Regulatory Guide requirements, a diesel-generator unit approximately two to three times as large as that required to carry the continuous rated load, would be necessary. The specified diesel engine-electric generator-pump assembly was designed specifically for this integral operation. The frequency and voltage overshoot requirements of RG 1.9 are met. A factory testing program on a prototype unit has verified the following functions:

1. system fast-start capabilities,
2. load-carrying capability,
3. load shedding capability,
4. ability of the system to accept and carry the required loads, and
5. the mechanical integrity of the diesel-engine generator unit and all of the major system auxiliaries.

GE licensing Topical Report, “HPCS Power Supply, NEDO-10905", describes the theoretical analytical aspects of the unique application including prototype and reliability test considerations.

The design of the HPCS diesel-generator conforms with the applicable sections of IEEE Criteria for Class I.E “Electrical Systems for Nuclear Power Generation Stations", IEEE Standard 308-1971.

The generator has the capability of providing power for starting the required loads with operationally acceptable voltage and frequency recovery characteristics. A partial or complete load rejection will not cause the diesel-engine to trip on overspeed.

A special prototype test was conducted at the LSCS facility to field (site) verify the hardware real load aspects of the HPCS power supply concept. This test was conducted in February 1979. This prototype test verified the acceptability of the HPCS power supply concept.

The HPCS diesel generators utilized in LSCS Units 1 and 2 are in compliance with the intent of this guide through the alternate approach cited above. The other standby-power diesel-generators and their loading schemes are in compliance with the guidance set forth in this guide.

REGULATORY GUIDE 1.10 – REV. 1

MECHANICAL (CADWELD) SPLICES IN REINFORCING BARS OF CATEGORY I CONCRETE STRUCTURES

Regulatory Guide 1.10 provides acceptable procedures for splicing, crew qualification, visual inspection, tensile testing, tensile test frequency and procedure for handling substandard tensile test results for mechanical (cadweld) splices in reinforcing bars of Category I concrete structures.

Information on cadweld splices is provided in Appendix E.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.11 – REV, 0

INSTRUMENT LINES PENETRATING PRIMARY REACTOR CONTAINMENT

Regulatory Guide 1.11 describes an acceptable basis on which to demonstrate the acceptability of instrument lines which penetrate or form a part of the reactor primary containment.

Instrument sensing lines that penetrate or connect to the primary reactor containment are designed to strict penetration requirements as discussed in Subsection 6.2.4 and shown in Table 6.2-21.

The provisions of this regulatory guide are incorporated into the design of the instrument sensing lines that penetrate or connect the primary containment at LSCS. We believe that we comply with the guidance set forth in this guide.

REGULATORY GUIDE 1.12 – REV. 2

INSTRUMENTATION FOR EARTHQUAKES

Regulatory Guide 1.12 describes acceptable seismic instrumentation to determine promptly the seismic response of nuclear power plant features important to safety to permit comparison of such response with that used as the design basis.

The seismic monitoring instrumentation consists of recorders, triaxial accelerometers and a central analyzing panel. The location and function of these seismic devices were selected to provide adequately for the determination of seismic event loads into the structures via computerized analysis programs. Subsection 3.7.4 provides descriptive details on the seismic instrumentation.

Technical Requirements Manual (TRM) Bases B 3.3.o, Seismic Monitoring Instrumentation, provides a detailed discussion of the instrumentation.

The provisions of this guide were incorporated into the design, location, and function of the seismic instrumentation utilized at the LSCS.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.13 – REV. 1

SPENT FUEL STORAGE FACILITY DESIGN BASIS

Regulatory Guide 1.13 requires that the fuel in pool storage be safeguarded from damage by missiles or the dropping of heavy loads from overhead cranes, that the fuel pool makeup system and filter system be designed to assure adequate water to cover the fuel, that low pool water level and high local radiation level be alarmed locally and in a continuously manned location, and that the pool be housed in a controlled leakage building having adequate ventilation and filtration to limit the potential release of radioactive materials.

The spent fuel pool, located inside the secondary containment, is designed to withstand the safe shutdown earthquake (SSE). (See Subsection 3.8.4 and Section 9.1.) The spent fuel storage racks are also designed to withstand the SSE.

A controlled leakage building is provided which encloses the fuel pool. The building is not designed to withstand extremely high winds, but leakage is suitably controlled during refueling operations. The building is equipped with a ventilation and filtration system which is designed to limit the potential consequences of the release of that activity specified in Regulatory Guide 1.25 to those guidelines set forth in 10 CFR 100.

Detection of high radioactivity in the reactor building ventilation system initiates the standby gas treatment system. Analysis of a potential refueling accident is presented in Chapter 15.0.

The reactor building below the refueling floor is designed to be tornado proof as discussed in Sections 3.3 and 3.5.

The spent fuel shipping cask storage area is separated from the fuel pool by a spent fuel transfer canal. Under no circumstances is it necessary to transport the shipping cask over the fuel pool. A reactor building crane path control system and explicit crane operating procedures are employed to ensure that the shipping cask is not transported over the fuel pool. Administrative procedures also prohibit transport of heavy objects over the fuel pool. See Subsection 9.1.4 for further details.

The refueling platform is designed to prevent it from toppling into the pools during an SSE.

The design of the spent fuel pooling cooling and cleanup system is such that the failure of any component in the system would not drain the fuel pool or uncover the fuel. There are no drains in the fuel pool. (Refer to Section 9.1.)

Normal fuel pool makeup is provided from the cycle condensate storage system through a connection to the fuel pool skimmer surge tanks. In the event that a source of emergency makeup is required, a Seismic Category I portion of the CSCS equipment cooling water system (Subsection 9.2.1) is provided. This system is capable of supplying enough water to prevent the uncovering of the fuel. There is no permanent connection between the emergency makeup, fuel pool cooling, and filter/demineralizing systems.

A fuel pool water level alarm is provided in the control room. High radiation in the fuel pool area will start the standby gas treatment system and alarms in the control room.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.14 – REV. 1

REACTOR COOLANT PUMP FLYWHEEL INTEGRITY

Regulatory Guide 1.14 describes a method to minimize the potential for failure of flywheels of reactor coolant pump motors.

Flywheels are not used in the LSCS reactor coolant pumps or pump motors.

REGULATORY GUIDE 1.15 – REV. 1

TESTING OF REINFORCING BARS FOR CATEGORY I CONCRETE STRUCTURES

Regulatory Guide 1.15 describes the tests and inspections of reinforcing bars for Category I concrete structures to assure that bars are tested to quality standards commensurate with the importance of the safety function to be performed.

The testing of reinforcing bars for Seismic Category I structures is described in Appendix E.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.16, – REV. 4

REPORTING OF OPERATING INFORMATION

Regulatory Guide 1.16, Revision 4 as revised by Generic Letter 97-02, "Revised Contents of the Monthly Operating Report" presents an acceptable format for meeting the reporting requirements of the LaSalle Technical Specifications.

REGULATORY GUIDE 1.17 – REV. 1

PROTECTION OF NUCLEAR POSIER PLANTS AGAINST INDUSTRIAL SABOTAGE

Regulatory Guide 1.17 describes physical security criteria that are generally acceptable for the protection of nuclear power plants against acts of industrial sabotage which could lead to a threat to the health and safety of the public.

The LaSalle County Station revised master security plan sets forth the principles, policies, and general requirements for industrial security at LSCS. Because of its sensitive contents it was separately submitted with the FSAR docket materials. It is intended to meet the requirements of the proposed 10 CFR 73.55. Implementing security procedures for onsite execution of the LSCS master security plan are a part of the station operating procedures. A reference is provided in Section 13.7.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.18 – REV. 1

STRUCTURAL ACCEPTANCE TEST FOR CONCRETE PRIMARY REACTOR CONTAINMENTS

Regulatory Guide 1.18 describes an acceptable method of demonstrating the capability of a concrete primary containment to withstand postulated pressure loads.

The LSCS structural acceptance test is described in Subsection 3.8.1.7. The concrete primary containment structure was designed as a Seismic Category I structure as described in Sections 3.7 and 3.8.

We believe that we comply with the structural acceptance test requirements as described in this regulatory guide.

REGULATORY GUIDE 1.19 – REV. 1

NONDESTRUCTIVE EXAMI NATI ON OF PRIMARY CONTA INMENT LINER WELDS

Regulatory Guide 1.19 describes acceptable procedures for demonstrating the leaktight integrity of the liner and penetrations of primary reactor containments of concrete construction by the use of nondestructive methods to examine the welds in the liner and penetrations.

Where the leak-chase-system channels are installed over liner welds, the channel- to liner-plate welds are tested for leaktightness by pressurizing the channels to the containment design pressure.

Further information concerning the examination of the primary containment liner welds are given in Appendix E.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.20 – REV. 2

VIBRATION MEASUREMENTS ON REACTOR INTERNALS

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 Records," of Appendix A to 10 CFR 50 and Section 50.34, "Contents of Applications; Technical Information," of 10 CFR 50.

An inspection test program has been defined for the confirmatory testing of this plant during preoperational flow tests based on the results of Browns Ferry and other vibration measurement programs. Core support structure and other reactor internals will be inspected at this plant for evidence of vibration following preoperational testing at specified recirculation flow conditions.

The Unit 1 pump adapter at its support plate, is a unique design, therefore it will be instrumented with vibration sinsois, as was done in the prototype plant, to qualify it to the same criteria as the remainder of the Unit 1 design.

Further information with respect to reactor internals vibration inspection programs is given in Subsections 3.9.2.3, 3.9.2.4, and 3.9.2.6.

We believe that we comply with the guidance set forth in this regulatory guide through incorporation of the alternate approach cited above.

REGULATORY GUIDE 1.21 – REV. 1

MEASURING, EVALUATING, AND REPORTING RADIOACTIVITY IN SOLID WASTES AND RELEASES OF RADIOACTIVE MATERIALS IN LIQUID AND GASEOUS EFFLUENTS FROM LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

Regulatory Guide 1.21 describes programs for measuring, reporting, and evaluating releases of radioactive materials in liquid and gaseous effluents and guidelines for classifying and reporting the categories and curie content of solid wastes.

The process and effluent radiological monitoring and sampling system is designed to provide the monitoring and sampling capability required to make the measurements, evaluations, and reports recommended b this guide. Refer to Sections 7.3, 7.6, 7.7, and 11.5.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.22 – REV. 0

PERIODIC TESTING OF PROTECTION SYSTEM ACTUATION FUNCTIONS

Regulatory Guide 1.22 delineates acceptable design approaches that facilitate the periodic testing, during reactor operation, of actuation devices/equipment incorporated into the reactor protection system design.

The reactor protection system is designed so that it may be tested during plant operation from the sensor device to final actuator device. The test must be performed in overlapping portions so that an actual reactor scram will not occur as a result of the testing. Refer to sections listed below for detailed information for the frequency of testing and minimum instrumentation availability.

Compliance for each system is described in the text as follows:

a. Reactor Protection System, Subsection 7.2.1.1;

b. Emergency Core Cooling System, Subsection 7.3.1.1;

c. PC RVICS, Subsection 7.3.2.2;

d. RCIC, Subsection 7.4.1.1;

e. Reactor Shutdown Cooling, Subsection 7 4.1.3;

f. Leak Detection System, Subsection 7.6.2.1;

g. HPCS Standby Power, Subsection 7.3.2.1;

h. RHR Containment Spray Cooling, Subsection 7.3.9.1;

i. Standby Liquid Control, Subsection 7.4.2.1;

j. Suppression Pool Cooling, Subsection 6.2.2.3; and

k. Process Radiation Monitor, Subsection 7.6.1.2.

We believe that we comply with the guidance set forth in this regulatory guide via NSSS analysis, design and equipment which is incorporated in the alternate approach cited above.

REGULATORY GUIDE 1.23 – REV. 0

ONSITE METEOROLOGICAL PROGRAMS

Regulatory Guide 1.23 describes a suitable onsite meteorological program to provide meteorological data needed to estimate potential radiation doses to the public as a result of the routine or accidental release of radioactive materials to the atmosphere and to assess other environmental effects.

The onsite meteorological program is discussed in Subsection 2.3.3. The original FSAR submittal utilized Dresden meteorology to indicate the type of diffusion and atmospheric dispersion attributable to the LSCS area. These data have now been superseded by revised submittals in early 1978 and currently for Alternative Source Terms, which categorizes LSCS onsite meteorology into the typical joint frequency distributions by receptor sector around LSCS. Non-accident Chi- over-Que's for the classical receptors were calculated for model and a realistic assessment of receptor dose via the NRC's models in Regulatory Guide 1.111.

Of particular importance is the wind velocity and stability parameters applicable for the elevated single release point (station vent stack). The wind instrumentation was moved from the 33-foot level to the 200-foot level to provide a backup capability for the wind instruments at 375 feet. Differential temperature data for the atmospheric stability parameter now references 33-375 feet; wind data references 375 feet (with backup at 200 feet). These data characterize the meteorology conditions for both normal and accidental releases. Adequate instrumentation is provided to assure the availability of reliable meteorological data for the analysis of routine and postulated accidental releases of gaseous radioactivity. Refinements in the LSCS meteorological base will be made as the cumulative data is acquired.

We believe that we comply with the objectives set forth in this regulatory guide.

REGULATORY GUIDE 1.24 – REV. 0

ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED WATER REACTOR RADIOACTIVE GAS S'IORAGE TANK FAILURE

Since LSCS uses a boiling water reactor, this guide is deemed not applicable.

REGULATORY GUIDE 1.25 – REV. 0

ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT IN THE FUEL HANDLING AND STORAGE FACILITY FOR BOILING AND PRESSURIZED WATER REACTORS

Replaced by RG 1.183, Rev. 0

REGULATORY GUIDE 1.26 – REV. 3

QUALITY GROUP CLASSIFICATIONS AND STANDARDS FOR WATER-, STEAM-, AND RADIOACTIVE-WASTE-CONTAINING COMPONENTS OF NUCLEAR POWER PLANTS

Regulatory Guide 1.26 describes a quality classification system for determining acceptable quality standards for safety-related components containing water, steam, or radioactive material other than those components addressed in Section 50.55a of 10 CFR 50.

The definition of quality group classifications for LSCS was made initially and recorded in the PSAR in accordance with ASME Boiler and Pressure Vessel Code, Sections III and VIII. Quality group classifications have been maintained during design and construction and are actively maintained during plant operations and modifications commensurate with the safety functions performed by the safety—related components. See Sections 3.9 and 5.2. This regulatory guide is applicable to Quality Groups B through D for pressure parts including piping, pumps, valves and vessels. Section 3.2 shows the Quality Group classifications of these parts.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.27 – REV. 2

ULTIMATE HEAT SINK FOR NUCLEAR POWER PLANTS

Regulatory Guide 1.27 requires the ultimate heat sink (UHS) to be capable of providing sufficient cooling for at least 30 days based on the worst period of recorded weather conditions and to be capable of withstanding 1) the most severe natural phenomena taken individually, 2) the site-related events that historically have occurred or may occur, 3) reasonably probable combinations of less severe natural phenomena and/or site—related events, and 4) a single failure of man-made structural features.

The ultimate heat sink consists of an impoundment which holds 460 acre-feet of water with a surface area of 83 acres. In the unlikely event that the main dike is breached, the analysis in Subsection 9.2.6 supports the availability of a 30-day supply of water to ensure that design temperatures are not exceeded. It is based on the record period of worst weather conditions in accord with the guidelines. Additional information is provided in Sections 2.4 and 2.5, respecting natural events and their effects on the UHS.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.28 – REVISION 3

QUALITY ASSURANCE PROGRAM REQUIREMENTS (DESIGN AND CONSTRUCTION)

Regulatory Guide 1.28 states that the general requirements and guidelines for establishing and executing a quality assurance program during the design and construction phases of nuclear power plants, which are included in ANSI N45.2 are acceptable and provide an adequate basis for compliance with Appendix B to 10 CFR 50.

Chapter 17.0 cites a reference document NO-AA-10 which is the updated version of Exelon Generation Company's (EGC) Quality Assurance Program. This document was submitted as a topical report and was accepted on December 24, 2002 by the NRC as a docketed record adequate for cross-referencing in licensing matters.

The General Electric BWR Quality Assurance Program has been developed over the years such that at any point in time it has been in compliance with mandatory regulatory requirements such as 10 CFR 50, Appendix B, and the ASME Code. Implementation of the applicable ANSI-N45.2 series standards and the associated NRC Regulatory Guides (or NRC-approved GE alternate positions) has been an evolutionary process and although partial implementation has always been effected before the date of issue of the Regulatory Guide or "AEC Guidance on Quality Assurance" which recognized applicable ANSI standards, full implementation was not necessarily in place until the GE commitment date.

We comply with the objectives set forth in the referenced revision of this regulatory guide through the implementation of the identified GE-BWR QA Program; S&L's QA Program; and the referenced Commonwealth Edison QA Program under which all LSCS activity pertaining to nuclear safety has been conducted.

In a letter to Station Managers from C. Reed dated December 4, 1986, CECo committed to Regulatory Guide 1.28 Rev. 3 which endorses ANSI/ASME NQA-1-1983 and the ANSI/ASME NQA-1a-1983 Addenda.

NRC SER dated 12/24/02, approved Revision 70 of the Exelon Nuclear Quality Assurance Topical Report which utilizes ANSI/ASME NQA-1-1994 as the basis for Exelon Nuclear quality assurance program.

REGULATORY GUIDE 1.29 – REV. 2

SEISMIC DESIGN CLASSIFICATIONS

Regulatory Guide 1.29 describes an acceptable method of identifying and classifying those features of light-water-cooled nuclear power plants that should be designed to withstand the effects of the SSE.

Information concerning the seismic design classification of structures, systems, and components of the LaSalle County Station, intended to withstand the effects of the SSE, is presented in Sections 3.2, 3.7, 3.8, 3.9, and 3.10.

For the NSSS equipment this regulatory guide was used as a basis for defining the systems and components which must meet Seismic Class I requirements. For the purpose of defining equipment that should withstand the SSE, NSSS equipment conforms to the guide. Also the regulatory guide includes positions applicable to parts of the NSSS equipment as follows:

<u>C.1(b)</u>

This guide is applicable to those reactor vessel internals which utilize engineered safety features, such as core spray piping, core spray spargers and hardware, etc.

The steam dryer, steam separator and feedwater spargers are not required to be Seismic Category I qualified components. However, these components are designated <u>Reliability Related</u> at LaSalle and as such come under the control of the Quality Assurance Program for future modifications and repairs during refueling outages.

<u>C.l(h)</u>

The component cooling water portions of the reactor recir culation pumps are not required to be Seismic Category I since the pumps do not perform a safety function.

<u>C.1(i)</u>

The seismic classification of the off-gas charcoal decay system is based upon Paragraph C.l.i, Regulatory Guide 1.29, which requires that such systems be Seismic Category I unless simultaneous failure would not result in conservatively calculated potential offsite exposure comparable to the guideline exposures of 10 CFR 100. GE Licensing Topical Report NEDO-l0734 is based upon calculations of release rates for off-gas charcoal delay system assuming seismically induced failure of the equipment. Offsite doses from component failure given in Table B-3 of that report show that the doses are well below the 10 CFR 100 limits.

Subsection 15.7.3 conservatively analyzes a postulated simultaneous failure of all the radwaste tanks in the radwaste building. The analysis assumes that 1% of the iodine is released to the atmosphere and at the time of failure, all tanks are filled to capacity (this condition is not expected). The analysis evaluates the possible control room site boundary and LPZ exposures to the whole body and thyroid. The results of the analysis indicate, in light of the requirements of Regulatory Guide 1.29, that the radwaste system is properly classified and changes are not required.

For the BOP equipment and design the following clarification and exceptions are noted.
<u>C.l(d)</u>

The normal cooling system for the spent fuel storage pool is not a Seismic Class I system because the service-water side of the heat exchanger is not a Class I system. This system will be routinely used for cooling the spent fuel storage pool, as the term normal implies. The emergency cooling system for the spent fuel storage pool is a Seismic Class I system when it is spooled into the loop for this service. Adequate post-SSE time is available to spool in this emergency system.

Alternately, a calculation of the radioiodine release from a boiling fuel pool has been made: thyroid doses are 4.5×10^{-4} rem at the EAB and 6.0×10^{-4} rem at the LPZ for 0-4 days inhalation. This shows that acceptable fuel storage exists even during boiling of the fuel storage pool without hazard to offsite personnel.

C.l (e)

LaSalle has a special D+ category of piping (superpipe) between the outermost containment isolation valves and the turbine stop valves. It is structurally equivalent to Seismic Class B piping. It has an augmented inspection requirement. The turbine stop valves are not Seismic Class I valves.

The D+ category for LaSalle piping was defined and accepted via exemption at the PSAR review stage. This plant was designed, equipment was ordered and built, and installation of the non-safety-related turbine stop valve fixed approximately 3 years before this NRC position was formulated. Future maintenance of the superpipe will be under QA cognizance, but maintenance on the turbine stop valve will not be under QA coverage.

C.2

Those portions of structures, systems, or components whose continued function is not required but whose failure could reduce the functioning of any plant feature required for safe shutdown to an unacceptable safety level are designed and constructed so that the SSE would not cause such failure. Plant features required for safe shutdown are described in Section H.4 "Safe Shutdown Analysis" of Appendix H.

We believe that we comply with the guidance set forth in the regulatory guide. The equipment and design utilized at LSCS as cited above, is in compliance with the intent of this guide via incorporation of the alternate approach specifically identified.

REGULATORY GUIDE 1.30 – REV. 0

QUALITY ASSURANCE REQUIREMENTS FOR THE INSTALLATION, INSPECTION, AND TESTING OF INSTRUMENTATION AND ELECTRIC EQUIPMENT

Regulatory Guide 1.30 describes an acceptable method of complying with regulations regarding the quality assurance requirements for the installation, inspection, and testing of nuclear power plant instrumentation and electric equipment.

The testing, inspection, and proper installation of electrical and instrumentation equipment are described for various subsystems in Chapter 7.0.

For operation and maintenance, detailed station procedures cover inspection and testing of safety-related instrumentation and electrical equipment. They are reviewed and approved by the station supervisory and management staff prior to use. The station activities affecting quality are accomplished in consonance with CE-lA by use of appropriate equipment, approval procedures, and station management controls.

For the NSSS, see compliance assessment made under Regulatory Guide 1.28 in this Appendix B.

We comply with the objectives set forth in the referenced revision of this regulatory guide because quality assurance for the installation, inspection, and testing of instrumentation and electric equipment are in accordance with ANSI N45.2.4 (1972) and IEEE 336-1971, as indicated in the textual material referenced above.

REGULATORY GUIDE 1.31 – REV. 1

CONTROL OF STAINLESS STEEL WELDING

Regulatory Guide 1.31 describes an acceptable method of implementing requirements with regard to the control of welding when fabricating and joining austenitic stainless steel components and systems.

All austenitic stainless steel weld filler materials were supplied with a minimum of 5% delta ferrite. This amount of ferrite is considered adequate to prevent microfissuring in austenitic stainless steel welds.

An extensive test program performed by General Electric Company, with the concurrence of the Regulatory Staff, has demonstrated that controlling weld filler metal ferrite at 5% minimum produces production welds which meet the requirements of this regulatory guide.

A total of approximately 400 production welds in five BWR plants were measured and all welds met the requirements of the Interim Regulatory Position.

We believe that we comply with the intent of the guidance set forth in this regulatory guide via an assessed capability in the design and the related test program which demonstrated adequate welds.

REGULATORY GUIDE 1.32 – REV. 0

USE OF IEEE STANDARD 308-1971, "CRITERIA FOR CLASS 1E ELECTRIC SYSTEMS FOR NUCLEAR POWER GENERATING STATIONS"

Regulatory Guide 1.32 states that IEEE 308-1971 provides criteria that may be used in establishing some of the bases for the design of electric power systems, except that conflicts with General Design Criterion 17 should be resolved by:

a. provision of two immediate access circuits from the transmission network; and

b. the capacity of the battery charger supply should be based on: the largest combined demands of the various steady-state loads and enough charging capacity to restore the battery from the minimum design charge state to a fully charged state, irrespective of the states of the plant.

Two physically independent 345kV transmission circuits (four transmission lines) occupying separate rights-of-way are available to each of the two LSCS reactor units so that each unit has an immediate access to a second off site power source. Off site power is available to each unit from the systems auxiliary transformer of the opposite unit via non-redundant bus ties between emergency 4kV buses. Switchyard power is available to both units as long as one of the four 345kV transmission lines is available (Section 8.2).

Each Division 1 and 2 battery charger as well as each HPCS (Division 3) battery charger has sufficient capacity to restore its battery to full charge under the maximum steady-state load (Section 8.3).

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.32 REV. 2

CRITERIA FOR SAFETY-RELATED ELECTRIC POWER SYSTEMS FOR NUCLEAR POWER PLANTS

Regulatory Guide 1.32 endorses IEEE Standard 450-1975, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations." The regulatory guide specifies what type of battery tests and the testing interval based on IEEE Standard 450-1975. The applicant complies with these regulatory positions in Revision 2 with the following clarification.

1. Paragraph C.1.c. Exception is taken to the requirements of IEEE Standard 450-1975 for battery testing. IEEE Standard 450-1995 will be utilized to perform battery testing. A battery service test will be performed to verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle. A battery performance test discharge test or a modified performance discharge test will be performed to verify the battery capacity is >80% of the manufacture's rating. The modified performance test may be performed in lieu of the service test provided the modified performance test completely envelops the service test. The battery testing intervals are in accordance with the Technical Specification requirements.

REGULATORY GUIDE 1.33 – REV. 2

QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATION)

Regulatory Guide 1.33 describes an acceptable method of complying with regulations regarding overall quality assurance program requirements for the operation of nuclear power plant structures, systems, and components.

The quality assurance program for nuclear generating stations is described in Exelon Nuclear Quality Assurance Program Topical Report NO-AA-10, referenced in Chapter 17.0. Under the alternate methods allowed by Regulatory Guide 1.33, the LSCS operating procedures were originally compiled per ANSI N18.7-1972 and ANSI N45.2-1971; subsequently, they were updated, where applicable, to ANSI N18.7-1976 standards during station review and approval. Fulfillment of the recommendations of this regulatory guide indicates conformance to the requirements of the ANSI guidelines applicable to the BWR standardized operating procedures used at LaSalle. For standardization purposes among Edison BWR plants, additional procedures were added and revisions were made to incorporate experience at similar operating stations such that uniform operating practices are utilized throughout Edison's BWR stations.

The Exelon Nuclear QA Program utilized for LSCS complies with the objectives set forth in the referenced revision of this regulatory guide, as indicated in the textual material mentioned above.

The NRC has approved the use of ANS 3.2, ANSI N18.7-1988 sections pertaining to biennial review of procedures, as defined in the Exelon Nuclear Quality Assurance Topical Report.

REGULATORY GUIDE 1.34 – REV. 0

CONTROL OF ELECTROSLAG WELD PROPERTIES

Regulatory Guide 1.34 describes an acceptable method of implementing requirements regarding control of weld properties when fabricating electroslag welds for nuclear components made of ferritic or austenitic materials.

Procurement specifications for ASME Code Section III Class 1 and 2 components require adherence to the Section IX, Welding Qualifications. Product quality certifications are required of vendors for applicable hardware. Records of qualifying tests and inspections are maintained by the vendor and are audited by the procuring contractor. Electroslag welding is not used on NSSS components; therefore this regulatory guide is not applicable thereto.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.35 – REVISION 3

INSERVICE INSPECTION OF UNGROUTED TENDONS IN PRESTRESSED CONCRETE CONTAINMENT STRUCTURES

Regulatory Guide 1.35 describes a surveillance program for "typical" containments which have vertical, hoop, and dome tendons. The LSCS containments do not have dome tendons. Therefore, the provisions of Regulatory Guide 1.35 are not directly applicable.

The LSCS tendon surveillance program is described in UFSAR Section 3.8.1.7.

REGULATORY GUIDE 1.36 - REVISION 0

NONMETHLLIC THERMAL ISULATION FOR AUSTENITIC STAINLESS STEEL

Regulatory Guide 1.36 requires that levels of leachable contaminants in nonmetallic insulation materials that come in contact with austenitic stainless steels of the AISI 3XX series used in fluid systems important to safety be carefully controlled so that stress-corrosion cracking is not promoted.

Reflective metallic insulation is used around all stainless steel Class I process fluid transporting equipment. Control of halogens is precisely specified in all materials which contact the austenitic stainless steels during fabrication, storage, erection, testing and service life. Use of nonmetallic thermal insulation is restricted to non-stainless steel parts of the plant.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.37 – REV. 0

QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER-COOLED NUCLEAR POWER PLANTS

Regulatory Guide 1.37 states that requirements and recommendations for onsite cleaning of materials and components, cleanness control, and preoperational cleaning and lay-up of water-cooled nuclear power plant fluid systems included in ANSI N45.2.1-1973 provide an adequate basis for complying with pertinent quality assurance requirements of Appendix B to 10 CFR 50, subject to the following:

a. applicability and acceptability of references addressed elsewhere,

b. should be used during operation phase of a nuclear power plant if applicable,

c. water quality for final flushes to be at least equivalent to quality of operating system water,

d. chemical compounds that could contribute to intergranular cracking or stress- corrosion cracking should not be used on austenitic stainless steels or nickel-based alloys,

e. tools which contain materials that could contribute to intergranular cracking or stress-corrosion cracking or which may have become contaminated with such materials should not be used on the surfaces of corrosion—resistant alloys, and

f. ASTM A 393-63 is no longer a valid test.

Onsite cleaning of materials and components, cleanness control, and preoperational cleaning and lay-up of fluid systems for LSCS are based on ANSI N145.2.1 and acknowledge the specific guidelines enumerated. Cleaning and flushing instructions are incorporated in the criterion tests in addition to procedures for cleaning the plant's fluid systems.

Therefore we comply with the objectives set forth in the referenced revision of this regulatory guide.

REGULATORY GUIDE 1.38 – REV. 2

QUALITY ASSURANCE REQUIREMENTS FOR PACKAGING, SHIPPING, RECEIVING, STORAGE, AND HANDLING OF ITEMS FOR WATER-COOLED NUCLEAR POWER PLANTS

Regulatory Guide 1.38 describes a method acceptable to the NRC for complying with regulations regarding the quality assurance requirements for the packaging, shipping, receiving, storage, and handling of items for water-cooled nuclear power plants.

Packaging, shipping, receiving, storage, and handling of BWR power plant equipment are covered by quality specifications based on ANSI N45.2.2-1972. These practices are audited for compliance in accordance with the Commonwealth Edison Quality Assurance Program previously referenced. Compliance with these guidelines extends beyond the construction phase into the operational phase and are followed where appropriate.

Therefore we comply with the objectives set forth in the referenced revision of this regulatory guide.

REGULATORY GUIDE 1.39 – REV. 1

HOUSEKEEPING REQUIREMENTS FOR WATER-COOLED NUCLEAR POWER PLANTS

Regulatory Guide 1.39 states that the requirements and guidelines for the control of housekeeping activities, conditions and environments at water-cooled nuclear power plant sites which are included in ANSI N45.2.3-1973 are acceptable and provide an adequate basis for complying with pertinent quality assurance requirements of Appendix B to 10 CFR 50, subject to the following:

a. Applicability and acceptability of ANSI references as addressed elsewhere in commission regulations.

b. Requirements should be considered applicable to the operation phase of a nuclear power plant.

Control of facility cleanness, care of material and equipment, fire prevention and protection, disposal of debris, protection of material and control of access are all standard practices of the station construction department during construction. After transfer of the plant to the operating staff, these functions are the responsibility of the production department. Normal management attention and periodic audits under the CEC0 QA program will provide the desired result at LSCS. Independent audits by NRC Region III personnel also contribute to the effectiveness of good housekeeping practices.

We comply with the objectives set forth in the referenced revision of this regulatory guide.

REGULATORY GUIDE 1.40 – REV. 0

QUALIFICATION TESTS OF CONTINUOUS-DUTY MOTORS INSTALLED INSIDE THE CONTAINMENT OF WATER-COOLED NUCLEAR POWER PLANTS

Regulatory Guide 1.40 states that the procedures for conducting qualification tests of continuous-duty motors installed inside the containment of water-cooled nuclear power plants which are specified by IEEE 334-1971 are acceptable, subject to the following:

a. Auxiliary equipment that will be part of the installed motor assembly should also be qualified hereunder.

b. Tests should simulate as closely as practicable all design basis events which affect operation of the motor's auxiliary equipment.

c. Applicability and acceptability of references will be addressed elsewhere.

The recirculation pump is part of the RCPB and is designed to ASME III Class 1 requirements, however, the motor is not Class 1E. Performance of the safety function (i.e., coastdown) of the pump and motor combination is assured by specifying the following additional requirements:

a. A LOCA shall not degrade puirp and motor coastdown performance in the unbroken loop to the extent that the core is deprived of adequate cooling.

b. The pump and motor bearing shall have sufficient dynamic load capability at rated operating conditions to withstand the safe shutdown earthquake.

There are no NSSS Class 1E safety-related continuous duty motors within the containment of this plant.

REGULATORY GUIDE 1.41 – REV. 0

PREOPERATIONAL TESTING OF REDUNDANT ONSITE ELECTRIC POWER SYSTEMS TO VERIFY PROPER LOAD GROUP ASSIGNMENTS

Regulatory Guide 1.41 describes an acceptable testing program to verify the existence of independence among redundant onsite power sources and their load groups.

Each standby power system is designed to be tested independently of any other redundant load group in conformance with the provisions of this regulatory guide.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.42 – REV. 0

INTERIUM LICENSING POLICY ON AS LOW AS PRACTICABLE FOR GASEOUS RADIOIODINE RELEASES FROM LIGHT WATER-COOLED NUCLEAR POWER REACTORS

This guide is no longer applicable.

REGULATORY GUIDE 1.43 – REV. 0

CONTROL OF STAINLESS STEEL WELD CLADDING OF LOW-ALLOY STEEL COMPONENTS

Regulatory Guide 1.43 describes acceptable methods for the selection and control of welding processes used for cladding ferritic steel components with austenitic stainless steel to restrict practices that could result in underclad cracking.

Procurement specification for ASME Code Section III structures and components using stainless steel cladding on low alloy steel require adherence to ASME Section IX, Welding Qualifications. Product certifications are required of vendors for applicable hardware. Records of fabrication and qualification tests and inspections are maintained by the vendor. These and the process controls and performance test results are audited by the procuring contractor (such as GE for the NSSS).

The stainless steel clad on the blend radii of the principal RPV nozzles is being removed by a special cutting machine. Stainless steel safe-ends were replaced with carbon steel safe-ends.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.44 – REV. 0

CONTROL OF THE USE OF SENSITIZED STAINLESS STEEL

Regulatory Guide 1.44 describes acceptable methods for the control of the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking.

The purpose of this guide is to address 10 CFR 50 Appendix A, GDC's 1 and 4, and Appendix B requirements to control "the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking." The guide proposes that this should be done by limiting sensitization due to welding as measured by ASTM A262 Practice A or E, or another method that can be demonstrated to show nonsensitization in austenitic stainless steels.

Tests by General Electric indicate that the test specified by A262 A or E (Detecting Susceptibility to Intergranular Attack in Stainless Steel) detects sensitization in a gross way, and that the tests do not provide a precise method of predicting susceptibility to stress corrosion cracking in the BWR environment.

All austenitic stainless steel for LSCS Units 1 and 2 was purchased in the solution heat treated condition in accordance with applicable ASME and ASTM specifications. Carbon content was limited to 0.08% maximum, and cooling rates from solution heat treating temperatures were required to be rapid enough to prevent sensitization.

Welding heat input was restricted to 110,000 joules per inch maximum, and interpass temperature to 350° F. High heat welding processes such as block welding and electroslag welding were not permitted. All weld filler metal and castings were required by specification to have a minimum of 5% ferrite. As deposited austenitic welds were controlled to have at least 3% ferrite content. See Subsection 5.2.3.14 for specific reference to control of the use of sensitized stainless steel. Additionally, grinding of field erection welds was prohibited on the reactor fluid side of Class 1 stainless steel pressure boundary pipe.

Whenever any wrought austenitic stainless steel was heated to temperatures over 800° F by means other than welding or thermal cutting, the material was re-solution heat treated.

These controls were used to avoid severe sensitization and to comply with the intent of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

We believe that we comply with the intent ~f this guide via incorporation of the alternate approach cited above.

REGULATORY GUIDE 1.45 – REV. 0

REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEMS

Regulatory Guide 1.45 describes methods acceptable to the NRC to assure that leakage detection and collection systems provide maximum practical identification of leaks within the reactor coolant pressure boundary.

The leak detection equipment consists of temperature, pressure, fission product monitoring and flow sensors with associated instrumentation and alarms. This equipment detects, annunciates, and isolates (in certain cases) leakages in the following systems:

a. Main steamlines

b. Coolant systems within the drywell

c. Reactor water cleanup (RWCU) system

d. Residual heat removal (RHR) system

e. Reactor core isolation cooling (RCIC) system

f. Feedwater system

g. High-Pressure Core Spray and Low-Pressure Core Spray.

Leakage is separated into identified and unidentified categories. The leakage to primary containment from identified sources is collected and isolated so that flow rates are separately monitored. For example, the leakage from the MSIV's is separately identified and measured by localized sensors. Leakage to primary containment from unidentified sources is also collected and this accumulation rate is monitored. Small unidentified leaks (5 gpm and less) inside the drywell are detected by temperature changes, pressure changes, drain pump activities, fission product monitoring, and drywell cooler condensate flow monitoring.

Large leaks are also detected by changes in reactor water level and changes in flow rates in process lines.

Detection methods used to identify or quantify unidentified leakage include:

a. sump level and flow rate monitoring,

b. airborne particulate radioactivity monitoring, and

c. airborne gaseous radioactivity monitoring.

These fission product detection methods are backed up with sump flow rate monitoring and drywell cooler condensate monitoring, thus providing diversity of methods.

The leak detection equipment is capable of monitoring leakage of 1 gpm in less than 1 hour. The airborne activity monitors are used as trend devices only.

Isolation and/or alarm of affected systems and the detection methods used are summarized in Table 5.2-9 of the LSCS UFSAR.

Monitoring of coolant for radiation in the RHR and Reactor Water Heat Exchanger satisfies Position C.4 of the regulator guide. (For system details see Process Radiation Monitoring UFSAR Section 7.6.)

Leakage detection indicators and alarms are provided in the main control room. Operators use standard procedures to convert various indications to leakage equivalents.

The leakage detection systems are equipped with provisions to permit testing for operability and calibration during operation by the following methods:

a. Continuous monitoring of sump level compared to flow rates into sump.

b. Operability checked by comparing one method to ‘another.

c. Simulation of signals into trip monitors.

d. Channel “A” against Channel “B” of the same method.

Limiting conditions for identified and unidentified leakage are established in the Technical Specifications.

At least one particulate radioactivity monitoring channel is expected to remain functional following an SSE. Administrative procedures can be used to verify operability of this type of monitor following an SSE.

Additional information is presented in Sections 5.2, 7.1, 7.6, and 11.5.

The RCS leakage detection systems are consistent with the recommendations of Regulatory Guide 1.45.

REGULATORY GUIDE 1.46 – REV. 0

PROTECTION AGAINST PIPE WHIP INSIDE CONTAINMENT

Regulatory Guide 1.46 describes an acceptable basis for selecting the design locations and orientations of postulated breaks in fluid system piping within the reactor containment and for determining the measures that should be taken for restraint against pipe whipping that may result from such breaks.

The design of the containment structure, component arrangement, Class 1 pipe runs, pipe restraints and compartmentalization was done in consonance with the acknowledgement of protection against dynamic effects associated with postulated rupture of piping. Analytically sized and positioned pipe restraints were engineered to preclude damage based on the pipe break evaluation. See Appendix C for additional information.

This regulatory guide is applicable to recirculation pipe lines.

The design of the containment structure, component arrangement, Class 1 pipe runs, pipe whip restraints and compartmentalization was done in consonance with the acknowledgment of protection against dynamic effects associated with postulated rupture of piping. Analytically sized and positioned pipe whip restraints were engineered to preclude damage based on the pipe break evaluation.

Pipe whip requirements for fluid system piping within the primary containment that, under normal operation, has service temperatures > 200°F or pressures > 275 psig, complied with ANS N176, "Design Basis for Protection Against Pipe Whip," and Regulatory Guide 1.46 except as delineated in the following criteria f or no breaks in Class 1 piping:

a. If Equation 10 of NB—3653—1, ASME Code III results in $S < 2.4\ S_m$ for ferritic or austenitic steels, no other requirements need be met. Stress range should be calculated between any two load sets (including zero load set) according to NB-3600 for upset and an OBE event transient.

b. If Equation 10 results in $2.4 < S < 3.0\ S_m$ for ferritic or austenitic steels, the cumulative usage factor, U, calculated on the basis of Equation 14 of NB-3653.6, must be < 0.1.

c. If Equation 10 results in S > 3.0 Sm for ferritic or austenitic steels, then the stress value in Equations 12 and 13 of NB-3653..6 must not be > 2.4 S_m.

We believe that the recirculation piping is in compliance with the intent of this regulatory guide through incorporation of the alternate approach cited above. The remainder of the piping inside the containment has been evaluated as described in Subsections 3.6.1 and 3.6.2 and meets the intent of this Regulatory Guide.

REGULATORY GUIDE 1.47 – REV. 0

BYPASSED AND INOPERABLE STATUS INDICATION FOR NUCLEAR POWER PLANT SAFETY SYSTEMS

Regulatory Guide 1.47 requires that automatic indication at the system level be provided of the bypass or deliberately induced inoperability of the protection system, of systems actuated by the protection system, and of auxiliary or suppporting systems that effectively bypass or render inoperable the protection system.

Regulatory Guide 1.47 was issued after the Advisory Committee on Reactor Safeguards (ACRS) review of LSCS. The LaSalle County Station PSAR described a bypass and inoperable status indicator which was manually actuated by switches in the control room. This procedure supplements the normal administrative procedures followed by the Commonwealth Edison Company for rendering equipment inoperable.

Any bypass or deliberately induced inoperability of the protection system and those systems actuated or controlled by the protection system will be indicated at those system levels. Likewise, bypassed or deliberately induced inoperability of auxiliary or supporting systems are indicated at the same system level. By manually initiating this status indicator as part of the administrative procedure, or by automatically indicating a bypass or inoperative condition at the system level through the remote digital surveillance system described in Section 7.8 of the UFSAR the control room operators are made aware of potential technical specification violations prior to performing any improper bypass function.

Refer to Subsections 7.2.2, 7.3.1, and 7.6.1 for additional information concerning bypass status indications which are annunciated.

We believe that we comply with the intent of the guidance set forth for LSCS.

REGULATORY GUIDE 1.48 – REV. 0

DESIGN LIMITS AND LOADING COMBINATIONS FOR SEISMIC CATEGORY I FLUID SYSTEM COMPONENTS

Regulatory Guide 1.48 delineates acceptable design limits and appropriate combinations of loadings associated with normal operation, postulated accidents, and specified seismic events the design of Seismic Category I fluid system components.

The design limits specified for Class 1 components are in compliance with those specified in Article NB of Section III of the ASME B&PV Code, based on the following load combinations.

Normal	Normal plant operation conditions.
Upset	a) Normal + 50% SSE
	b) SRV loads + 50% SSE
Emergency	Normal + 100% SSE
Faulted	Normal + 100% SSE + pipe rupture loads (if applicable)

The design basis was representative of good industry practices at the time of design, procurement and manufacture and is shown to be in general agreement with requirements of Regulatory Guide 1.48, with the following clarifications: (a) The probability of an OBE of the magnitude postulated for LSCS is consistent with its classification as an emergency event. However, for design conservatism, loads due to the OBE vibratory motion have been included under upset conditions, loads due to the OBE vibratory motion plus associated transients, such as a turbine trip, have been considered in the equipment design under emergency conditions consistent with the probability of the OBE occurrence, and (ID) the use of increased stress level for Class 2 components is consistent with industry practice as specified in ASME Code Section III.

We believe that the design limits and load combinations for Seismic Category I fluid system components are in compliance with the intent of this guide through the incorporation of the alternate approach cited above.

REGULATORY GUIDE 1.49 – REV. 1

POWER LEVELS OF NUCLEAR POWER PLANTS

Regulatory Guide 1.49 requires that the proposed licensed power level be restricted to a reactor core power level of 3800 megawatts thermal or less and that analyses and evaluations in support of the application should be made at 1.02 times the proposed licensed power level.

The original rated thermal power for LSCS reactors W25 3323 thermal megawatts. The safety analyses and evaluations were made for an LSCS power level of 3454 thermal megawatts.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.50 – REV. 0

CONTROL OF PREHEAT TEMPERATURE FOR WELDING CF LOW-ALLOY STEEL

Regulatory Guide 1.50 requires that weld fabrication for low-alloy steel components should comply with fabrication requirements of Section III and Section IX of the ASME B&PV Code supplemented by the following:

a. procedures qualification should require that a minimum preheat and maximum interpass temperature be specified and that the welding procedure be qualified at the minimum preheat temperature,

b. preheat temperature be maintained until post-weld heat treatment has been performed, and

c. welding should be monitored to verify that limits on preheat and interpass temperatures are maintained.

For the NSSS equipment, the use of low-alloy steel is restricted to the reactor pressure vessel. Other ferritic components in the reactor coolant pressure boundary are fabricated from carbon steel materials.

Preheat temperatures employed for welding of low-alloy steel meet or exceed the recommendations of ASME Section III, Subsection NA. Components were either held for an extended time at preheat temperature to assure removal of hydrogen, or preheat was maintained until postweld heat treatment. The minimum preheat and maximum interpass temperatures were specified and monitored.

All welds were nondestructively examined by radiographic methods. In addition, a supplemental ultrasonic examination was performed.

By meeting and/or exceeding the recommendations of the ASME Code, the intent of this guide is satisfied even though the design was developed prior to its issuance.

REGULATORY GUIDE 1.51 – REV. 0

INSERVICE INSPECTION OF ASME CODE CLASS 2 AND 3 NUCLEAR POWER PLANT COMPONENTS

This regulatory guide was withdrawn by Mr. Robert B. Minogue's letter, dated July 15, 1975. The withdrawal letter states that current licensing practice is based on reference to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The LaSalle County Station, Units 1 and 2, is designed to comply with ASME Section XI.

REGULATORY GUIDE 1.52 – REV. 2

DESIGN, TESTING, AND MAINTENANCE CRITERIA FOR ATMOSPHERE CLEANUP SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

Regulatory Guide 1.52 presents acceptable methods for implementing regulations with regard to the design, testing, and maintenance of air filtration and adsorption units of engineered-safety-feature atmospheric cleanup systems designed to mitigate consequences of postulated accidents.

A detailed description of the design and operation of the gaseous effluent treatment equipment including filters, charcoal, etc. is presented in Subsections 6.2.3, 9.4.1, and 11.3.1.

The ESF atmospheric cleanup units at LaSalle were designed and constructed to comply with the intent of the revision of Regulatory Guide *1.52,* which was current at the time of their procurement.

The ESF atmospheric cleanup units at LaSalle that require in-place testing will be tested to comply with the intent of the now current Revision 2 of Regulatory Guide 1.52.

REGULATORY GUIDE 1.53 – REV. 0

APPLICATION OF THE SINGLE-FAILURE CRITERION
TO NUCLEAR POWER PLANT
PROTECTION SYSTEMS

Regulatory Guide 1.53 requires that protection systems meet the requirements of Section 4.2 of IEEE 279-1971, which is also ANSI-N 42.7-1972 in that any single failure within the protection systems shall not prevent proper protective action at the system level when required. This guide provides guidance on an acceptable method of complying with this requirement.

Compliance is achieved by specifying, designing, and constructing the engineered safeguards systems to meet the single failure criterion, Section 4.2 of IEEE 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," and IEEE 379- 1972, "IEEE Trial-Use Guide for the Application of the single- Failure Criterion to Nuclear Power Generating Station Protection Systems." The reactor protection system has separate and redundant instrument channels and control logic to ensure that a failure in a sensing element or the decision logic or an actuator does not prevent or initiate protective action. Separated channels are employed, so that a fault affecting one channel does not prevent the other channels from operating properly.

The reactor protection system is normally energized, with two motor-generator sets for power, one for each separate trip system. Therefore, a single power failure produces a trip on only one channel. Complete loss of power trips the reactor since its protection system is designed for fail-safe operation.

The ECCS is divided into ADS, HPCS, LPCS, and RHR (LPCI) which meets the Single Failure criterion on a network basis.

Testing logic is provided so that the equipment can be operated in various test modes to confirm that it will operate properly when called upon. Testing incorporates all elements of the system under various test modes including sensors, logic, actuators, and actuated equipment. Tests are performed at intervals so that there is an extremely low probability of failure in the intervening periods. During tests sufficient channels and systems are available to not degrade the protective function.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.54 – REV. 0

QUALITY ASSURANCE REQUIREMENTS FOR PROTECTIVE COATINGS APPLIED TO WATER-COOLED NUCLEAR POWER PLANTS

Regulatory Guide 1.54 states that ANSI N101.4-1972 in conjunction with ANSI N45.2-1971 provides an adequate basis for complying with quality assurance requirements for protective coatings applied to ferritic steels, aluminum, stainless steel, galvanized steel, concrete, or masonry subject to the following:

a. applicability and acceptability of references to be determined elsewhere; and

b. coatings and cleaning materials used with stainless steel shall not contain elements that could contribute to corrosion, intergranular cracking, or stress corrosion cracking.

Protective coatings are not applied to austenitic stainless steels at the LSCS. The provisions of ANSI N101.4-1972 apply to protective coatings applied to other materials within the power plant at LSCS.

We believe that we comply with the intent of guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.55 – REV. 0

CONCRETE PLACEMENT IN CATEGORY I STRUCI'URES

Regulatory Guide 1.55 describes an acceptable method for implementing design control, material control, special processes control, and inspection and test control requirements with regard to the placement of concrete in Category I structures.

The applicant complies with the intent of the guidelines set forth in Regulatory Guide 1.55 and its Appendix. Refer to Appendix E for concrete placement in Category I structures.

Although the applicant complies with the regulatory position, the applicant does not comply with all the codes in the appendix of Regulatory Guide 1.55 in their entirety. Refer to Appendix E for exceptions taken to concrete codes with which the applicant complies.

REGULATORY GUIDE 1.56 – REV. 0

MAINTENANCE OF WATER PURITY IN BOILING WATER REACTORS

Regulatory Guide 1.56 describes an acceptable method of implementing GDC 13, 14, 15, and 31 of 10 CFR 50 Appendix A with regard to minimizing the probability of corrosion-induced failure of the RCPB in BWR's by maintaining acceptable purity levels in the reactor coolant, and acceptable instrumentation to determine the condition of the reactor coolant.

Materials in the primary system are primarily Type 304 stainless steel and Zircaloy cladding. The reactor water chemistry limits have been established to provide an environment favorable to these materials. Technical Specification limits are placed on conductivity and chloride concentrations. Operationally, the conductivity is limited because it can be continuously and reliably measured and gives an indication of abnormal conditions and the presence of unusual materials in the coolant. Chloride limits are specified to prevent stress corrosion cracking of stainless steel.

The water quality requirements are further supported by General Electric Topical Report NEDO-10899.

Conductivity is continuously monitored on the primary coolant with instruments connected to redundant sources, the reactor water recirculation loop and the reactor water cleanup system inlet. The effluent from the reactor water cleanup system is also monitored for conductivity on a continuous basis. These measurements provide reasonable assurance for adequate surveillance of the reactor coolant.

Sufficient instrumentation is provided so that the conductivity of the condensate is known and so that the purity of the demineralizer effluent is known. Other conductivity monitors provide redundant measurements of reactor water purity. Refer to Subsection 5.2.3 in the UFSAR.

The condensate demineralizer system and reactor water cleanup system provide the recorded conductivity measurements and alarms of influents and effluents of the demineralizers and records of the flow rate through each demineralizer. Refer to Subsections 5.4.8 and 10.4.6.

Based upon the assessed capability of the design, we believe we comply with the intent of this regulatory guide through incorporation of the alternate approach cited above.

REGULATORY GUIDE 1.57 – REV. 0

DESIGN LIMITS AND LOADING COMBINATIONS FOR METAL PRIMARY REACTOR CONTAINMENT SYSTEM COMPONENTS

Regulatory Guide 1.57 delineates acceptable design limits and appropriate combinations of loadings associated with normal operation, postulated accidents, and specified seismic events for the design of components of metal primary reactor containment systems.

The applicant complies with the Regulatory Position to the extent described in. Items a and b, below.

a. The design of ASME Code Class MC components of primary metal containment systems, which are completely enclosed within Seismic Category I structures, is in accordance with Subsection NE of Section III and meets the intent of Regulatory Guide 1.57.

b. The design of piping, pump, and valve components of primary metal containment systems and of piping penetration head fittings, which are directly subjected to process pipe internal pressure, is in accordance with Subsection NB or NC (as applicable) of Section III and meets the intent of Regulatory Guide 1.48.

REGULATORY GUIDE 1.58 – REV. 1

QUALIFICATION OF NUCLEAR POWER PLANT INSPECTION, EXAMINATION, AND TESTING PERSONNEL

Regulatory Guide 1.58 describes an acceptable method of complying with regulations with regard to the qualification of nuclear power plant inspection, testing, and examination personnel during the construction, preoperational testing, startup testing, and operational phases.

Station Operational Phase

For the operational phase, onsite inspection, examination and testing personnel are qualified in accordance with ANSI N18-1-1971 as described in UFSAR Section 13.1.3.

Offsite inspection, examination and testing personnel are qualified in accordance with Regulatory Guide 1.58, Revision 1, dated September 1980.

Commonwealth Edison complies with the objectives set forth in the referenced revision of this regulatory guide, as indicated in the textural material described above.

REGULATORY GUIDE 1.59 – REV. 0

DESIGN-BASIS FLOODS FOR NUCLEAR POWER PLANTS

Regulatory Guide 1.59 describes an acceptable method of determining for sites along streams or rivers the design basis floods that nuclear power plants must be designed to withstand without loss of safety-related functions.

Although LSCS is approximately 255 feet above the Illinois River an analysis of design basis floods was made for the Illinois River and the onsite lake and watershed (drainage) areas at the plant for the defined PMF. Design and engineering of the lake structures, and plant structures was based on PMF with maximum wind wave runup. The methodology was consistent with the extensive references provided as guidance. See Subsection 2.4.3.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.60 – REV. 1

DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS

Regulatory Guide 1.60 describes a procedure acceptable to the NRC for defining the response spectra for the design of Seismic Category I structures, components, and systems.

The requirements of current Regulatory Guide 1.60 (Revision 1) are not applicable to this facility as this revision of the regulatory guide was not issued at the time construction permits were issued for LSCS. However, a comparison between the LSCS design spectra and Regulatory Guide 1.60, Revision 1 spectra, corresponding to 0.2g and 0.lg maximum ground accelerations and for appropriate values of dampings is presented in Figures B-1 through B-4. Regulatory guide spectra for OBE 4% damping and SSE 7% damping have been compared with the corresponding 2% and 5% damping LSCS design spectra. This is because conservative values of 2% and 5% dampings were used in actual design whereas 4% and 7% dampings are allowed in Regulatory Guide 1.61.

With acknowledgment of the vintage of this plant vis-a-vis that of the current guidance, we believe we have complied with the intent of this regulatory guide via accepted industry practice at the time of the design. The response spectra used in the design were based on practical earthquake acceleration time-history records for this tectonic region.

REGULATORY GUIDE 1.61 – REV. 0

DAMPING VALUES FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS

Regulatory Guide 1.61 delineates damping values acceptable for use in the elastic modal dynamic seismic analysis of Seismic Category I structures, systems, and components.

With acknowledgement of the vintage of this plant vis-a-vis that of the current guidance, we believe we have complied with the intent of this regulatory guide via alternate but locally conservative damping values used in the seismic analysis of major structures.

REGULATORY GUIDE 1.62 – REV. 0

MANUAL INITIATION OF PROTECTIVE ACTIONS

Regulatory Guide 1.62 requires that manual initiation of each protective action at the system level be provided, that such initiation accomplishes all actions performed by automatic initiation, and that protective action at the system level go to completion once manually initiated. In addition, manual initiation should be by switches readily accessible in the control room, and a minimum of equipment should be used in common with automatically initiated protective action.

Means are provided for manual initiation of reactor isolation, emergency core cooling, and reactor manual scram at the system level through the use of armed pushbuttons, as described below:

Action initiated	Number of Switches	Switch location (Control Room Panel)
Reactor isolation	four	H13-P601
ADS	four, two in Div. 1 and two in Div. 2	H13-P601
HPCS	one switch in Div. 3	H13-P601
RHRA/LPCS	one switch in Div. 1	H13-P601
RHRB/RHRC	one switch in Div. 2	H13-P601
Reactor manual scram	four	H13-P601

Operation of these switches accomplishes the initiation of all actions performed by the automatic initiation circuitry.

The amount of equipment common to both manual and automatic initiation of the above functions is kept to a minimum through implementation of manual activation at the final devices (relays, scram contractor) of the protection system. No failure in the manual, automatic, or common portions of the protection system will prevent initiation of a given function by manual or automatic means.

To prevent manual initiation of vessel depressurization when low- pressure core cooling capability is absent, the ADS manual initiation has an interlock to assure proper conditions for depressurization (AC interlock). One interlock is provided for Division 1 ADS and a second independent interlock is provided for Division 2 ADS.

Manual initiation of any of the above functions, once initiated, goes to completion as required by IEEE 279-1971 Section 4.16.

For electric power systems, manual controls are provided to permit the operator to select the most suitable distribution path from the power supply to the load. An automatic start signal will override the test mode. Provision is made for control of the system from the control room as well as from an external location.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.63 – REV. 0

ELECTRIC PENETRATION ASSEMBLIES IN CONTAINMENT STRUCTURES FOR WATER-COOLED NUCLEAR POWER PLANTS

This regulatory guide describes a method acceptable to the NRC for compliance with GDC 50 and Appendix B to 10 CFR 50 requirements of the design, construction, and installation of all electrical penetrations through the primary containment barrier.

The LaSalle County Station electrical penetrations are designed and applied to withstand, without loss of mechanical integrity, the maximum possible fault current versus time conditions (which could occur because of single random failures of circuit overload protection devices) within the two leads of any one single-phase circuit or the 3 leads of any one three-phase circuit.

The LSCS design meets the objective of Position 1 of Regulatory Guide 1.63 by having two breakers for interruption of fault current for the medium voltage penetrations and by having one breaker and self-fusing circuit characteristics to interrupt fault current for the low voltage penetrations. However, on the low voltage penetrations the self-fusing circuit characteristics are external to the penetrations. The LSCS low voltage penetrations are designed with oversized conductors through the penetration seals such that they can withstand any conceivable fault current versus time condition not interrupted by the primary protective device. Should the backup protective device also fail to clear the fault, the ratio of conductor size within the seals to the conductor size of the external circuit is such that failure of the electrical circuit external to the penetration seals is calculated to occur prior to failure of the penetration conductors. This capability is confirmed by testing (see Subsection 3.8.1. This oversizing of the penetration assembly, so that fault current will not cause overheating of the electrical penetration before fault current is interrupted, meets the requirement of Section 4 of IEEE 317-1972.

However, Commonwealth Edison will, by the end of the first refueling outage, install redundant (backup) circuit breakers at LaSalle on all 480-Vac circuits which penetrate primary containment and which may be energized during plant operation.

These circuit breakers will be directly in series with the primary protection circuit breakers. Also, by the end of the first refueling outage redundant (backup) circuit protection devices will be installed on all control circuits penetrating primary containment which do not already have backup circuit protection devices and which may be energized during plant operation. Note the control circuits to valve operators include their limit switches because their power comes from a dedicated transformer which is fed by the controller's power source hence limit switches will be protected with the same redundant circuit protection devices as the parent controller. Control power is protected with fusing as the primary protection device.

No additional protection is planned for any instrumentation circuit which penetrates primary containment because all of the circuits are of such low energy that potential short circuit energy levels would have no adverse effect on the penetration seals.

Regulatory Guide 1.63 was issued after the LSCS construction permit. However, the LSCS design meets the objective of position 1 of Regulatory Guide 1.63 and meets Positions 2 and 4. Position 3 makes reference to other regulatory guides.

REGULATORY GUIDE 1.64 - REV. 2

QUALITY ASSURANCE REQUIREMENTS FOR THE DESIGN OF NUCLEAR POWER PLANTS

Regulatory Guide 1.64 describes an acceptable method for complying with the NRC's quality assurance requirements for the design of nuclear power plants.

Chapter 17.0 cities Exelon Nuclear's reference QA Topical Report NO-AA-10 which is the updated version of EGC-1-A describing the Exelon Nuclear Quality Assurance Program for design and control for its nuclear generating stations. This document was accepted by the NRC as a docketed record adequate for cross-referencing in licensing matters.

We comply with the objectives set forth in the referenced revision of this regulatory guide.

REGULATORY GUIDE 1.65 – REV. 0

MATERIALS AND INSPECTIONS FOR REACTOR VESSEL CLOSURE STUDS

Regulatory Guide 1.65 gives recommendations for the selection and application of the reactor vessel closure bolting materials, with the intent of preventing failures such as could occur from embrittlement, corrosion, and stress corrosion cracking. The bolting materials used are SA-540, Grades B-23 and B-24, as specified in the regulatory guide.

The maximum reported tensile strength and minimum Charpy impact test properties are as tabulated below, from the certified materials test reports:

	Tensile Strength Sn (psi)	Lateral Expan. (mils)	Cv Energy (ft—lb)
LSCS 1: Studs	174,000	23	43
Nuts and washers	174,500	22	36
LSCS 2: Studs	163,500	25	40
Nuts and washers	156,000	25	43

As shown, the maximum tensile strength values of the LSCS 1 bolting exceed the 170-ksi maximum given in the Guide. Also, the impact test properties are lower than specified by Paragraph IV.A.4 of 10 CFR 50, Appendix G. It is noted, however, that the impact data shown are the lowest values measured from tests at 10°F), instead of the lowest preload or service temperature (70°F) specified for the Appendix G requirements. It is probable that the properties would compare more favorably with the Appendix G minimums of 45 ft-lb and 25 mils lateral expansion, if tested at the higher temperature.

The extent of compliance with the Item C.2 inspection requirements of the Guide is described in the following.

The bolting materials were ultrasonically examined after final heat treatment and prior to threading, as specified. The specific requirement for examination according to SA-388 was not complied with. However, the procedures approved for use in practice are judged to ensure comparable material quality and moreover are considered adequate on the basis of compliance with the applicable requirements of ASME Code Paragraph N-322. Additionally, straight beam examination was performed on 100% of cylindrical surfaces and from both ends of each stud. In addition to the code required notch, the reference standard for the radial scan contained a 1/2-inch-diameter flat bottom hole with a depth of 10% of the thickness, and the end scan standard contained a 1/4-inch-diameter flat bottom hole 1/2 inch deep. Also, angle beam examination was performed on the outer cylindrical surfaces in both axial and circumferential surfaces in both axial and circumferential directions. Surface examinations were performed on the studs and nuts after final heat treatment and threading, as specified in the guide, in accordance with N-626 or N-627 of the applicable ASME Code.

As specified in Item C.1.b (3) of the guide, there are no metal platings applied to the closure bolting. A phosphate coating is applied to threaded areas to studs and nuts and bearing areas of nuts and washers to act as a rust inhibitor and to assist in retaining lubricant on these surfaces.

Item C.3 of the guide requires adequate corrosion protection of studs and flange holes during venting and filling of the vessel and while the head is removed. General Electric practice allows exposure of these surfaces to high-purity fill water. Nuts and washers are stored dry during refueling.

Volumetric examination of the studs is to be in accordance with ASME Code Section XI.

REGULATORY GUIDE 1.66 – REV. 0

NONDESTRUCTIVE EXAMINATION OF TUBULAR PRODUCTS

Regulatory Guide 1.66 describes an acceptable method for implementing the requirements for nondestructive examination of tubular products used for components of the reactor coolant pressure boundary and other safety-related systems.

Wrought tubular products were supplied in accordance with applicable ASTM/ASME material specifications. These specifications require a hydrostatic test on each length of tubing. Additionally, the specification for the tubular product used for CRD housing specified ultrasonic examination to Paragraph N132550 of the ASME Code, Section III.

These components met the requirements of ASME Codes existing at the time of placement of the order, which predated Regulatory Guide 1.66.

This regulatory guide was withdrawn on September 28, 1977 by the NRC because the additional requirements imposed were satisfied by the ASME Code. See Subsections 4.5.2 and 5.2.3.

REGULATORY GUIDE 1.67 – REV. 0

INSTALLATION OF OVERPRESSURE PROTECTION DEVICES

Regulatory Guide 1.67 describes a method acceptable to the NRC for implementing GDC 1 of 10 CFR 50, Appendix A with regard to the design of piping for safety valve and relief valve stations which have open discharge systems with limited discharge pipes and which have inlet piping that neither contains a water seal nor is subject to slug flow of water upon discharge of the valves.

The MSL safety/relief valves of the LSCS plant relieve to closed discharge systems. The guidelines of this guide are therefore not applicable.

REGULATORY GUIDE 1.68

Initial Issue: Revision 0, November 1973

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LaSalle C.P. Issue: September 10, 1973

PREOPERATIONAL AND INITIAL STARTUP TEST PROGRAMS FOR WATER-COOLED POWER REACTORS

Regulatory Guide 1.68 describes the NRC's requirements for the initial startup test programs. It is applicable to such activities as precritical tests and flow power tests.

A test program has been established to ensure that all structures, systems, and components will satisfactorily perform their safety-related functions. This test program provides additional assurance that the plant has been properly designed and constructed and is ready to operate in a manner that will not endanger the health and safety of the public, that the procedures for operating the plant safely have been evaluated and have been demonstrated, and that the operating organization is knowledgeable about the plant and procedures and are fully prepared to operate the facility in a safe manner.

The test program includes simulation of equipment failures and control system malfunctions that could reasonably be expected to occur during the plant lifetime. The test program also includes testing for interactions such as the performance of interlock circuits in the reactor protection system. It also determines that proper permissive and prohibit functions are performed and that circuits normally active and supposedly unaffected by the position of the mode switch perform their function in each mode. Care is taken to ensure that redundant channels of the equipment are tested independently.

We believe that the preoperational and initial startup test programs at LSCS comply with the intent of this regulatory guide

REGULATORY GUIDE 1.69 – REV. 0

CONCRETE RADIATION SHIELDS FOR NUCLEAR POWER PLANTS

ANSI 101.6-1972 basically deals with field practices. It also stresses quality assurance (QA) especially in the mixing and pouring of concrete. The parts of this document that coven shield design are more concerned with the protection and construction of shields than with the radiation protection aspects. The radiation protection is included in the station design criteria.
The shielding for LSCS has the following design:

a. the design density of the concrete used in shielding walls is 140 pounds per cubic foot;

b. the surrounding atmosphere does not exceed 1500 F;

c. all source areas have one on all of the following:
   1. leaktight liner,
   2. sealed shields,
   3. exhaust systems;

d. all shield penetrations and voids are considered;

e. removable walls and plugs are used where appropriate;

f. structural, thermal, missile, shielding load, accident, and operational analyses are made where required;

g. shield walls are designed to maintain a design dose rate; and

h. ordinary concrete is being used in the reactor shield.

Precautions are to be taken to ensure that the pouring of the reactor shield is done correctly. Further discussion of the shielding requirements is found in Section 12.1, and further discussion of the structural requirements is found in Section 3.8.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.70 – REV 2 (09/75)

The previous commitment to the Regulatory Guide is superseded by the following.

As part of the ongoing effort to improve the quality of the UFSAR, the guidelines provided in NEI 98-03 Revision 1, 06/99 and endorsed by the NRC in Regulatory

Guide 1.181, Revision 0, 09/99 will be used to further improve the content of the UFSAR. While the UFSAR will continue to follow the general organizational recommendations specified in this Regulatory Guide, the reformatting options described in NEI 98-03, Revision 1, will be used to simplify information contained in the UFSAR to improve their focus, clarity and maintainability. Examples of the types of improvements that are anticipated are the identification of historical information and the removal or replacement of excessively detailed drawings as described in Appendix A to NEI 98-03 Revision 1.

REGULATORY GUIDE 1.71 – REV. 0

WELDER QUALIFICATION FOR AREAS OF LIMITED ACCESSIBILITY

Regulatory Guide 1.71 requires that weld fabrication and repair for wrought low-alloy and high-alloy steels or other materials such as static and centrifugal castings and bimetallic joints should comply with fabrication requirements of Section III and Section IX supplemented by the following:

a. Performance qualification should require testing of welders under simulated access conditions when welder's access to weld is restricted to 30 to 35 cm. in any direction from the joint.

b. Regulation is required if:

   1. sufficiently different restricted accessibility occurs, or

   2. any of essential welding variables listed in Section IX are changed.

c. Production 'welding should be monitored and adherence to qualification requirements should be certified.

All ASME Section III welds were fabricated in accordance with the requirements of Sections III and IX of the ASME Boiler and Pressure Vessel Code.

Only a few restrictive welds are involved in the fabrication of LSCS pressure boundary components. Welder qualification for welds with the most restrictive access was accomplished via mockup welding. Mockups were examined with radiography or sectioning.

All reactor pressure boundary welding was performed in, accordance with ASME Section IX. Reactor internal components welding was performed in accordance with ASME Section IX or appropriate AWS requirements.

REGULATORY GUIDE 1.72 – REV. 0

SPRAY POND PLASTIC PIPING

This regulatory guide is not applicable to LSCS.

REGULATORY GUIDE 1.73 – REV. 0

QUALIFICATION TESTS OF ELECTRIC VALVE OPERATORS INSTALLED INSIDE THE CONTAINMENT OF NUCLEAR POWER PLANTS

To the extent practicable, the electric and mechanical components integral to the electric valve operator mechanism and required to operate and control valve action inside the containment have been tested in accordance with IEEE 382-1972.

The valve operators for Class lE electrically operated valves have been tested in accordance with the test sequence outlined in Section 4.5.2 of IEEE 382-1972. If the anticipated actual service operating sequence is expected to create a more severe operating condition, then this actual service sequence has been used in the qualifying tests.

The qualifying tests have been made under environmental conditions (temperature, pressure, humidity, radiation, dust) that are at least as severe as those that the valve operator will be exposed to during and following a design-basis accident (LOCA).

The radiological source term is based upon the same source term described in the guide.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.74
QUALITY ASSURANCE TERMS AND DEFINITIONS

Regulatory Guide 1.74 identifies quality assurance terms and acceptable definitions that are important to the understanding of the quality assurance requirements for design, construction, and operation of nuclear power plant structures, systems, and components.

Chapter 17.0 cites Edison's QA Topical Report CE-lA document was accepted by the adequate for cross-referencing in licensing matters.

We comply with the objectives set forth in the referenced revision of this regulatory guide.

REGULATORY GUIDE 1.75 – REV. 1

PHYSICAL INDEPENDENCE OF ELECTRIC SYSTEMS

Regulatory Guide 1.75 describes a method acceptable to the NRC for assuring physical independence of the circuits and electric equipment comprising or associated with Class 1E power systems, protection systems, systems actuated or controlled by the protection system, arid auxiliary or supporting systems that must be operable for the protection systems to perform their safety-related functions.

Regulatory Guide 1.75 was issued after the LaSalle County Station construction permit. However, the LSCS design meets most of the regulatory positions, as described in Subsection 8.3.1. Aside from the two exceptions noted below (Positions 1 and 10) the LSCS design complies with this guide. Position 2 is not applicable.

The LSCS design differs with Position 1 in that the basis for precluding the use of a fault-current-actuated interrupting device as an isolation device within the context of Regulatory Guide 1.75 is questionable. This conclusion seems to preclude the use of a circuit breaker as an isolating device and thus prevents achieving a practical design of an electrical auxiliary system. A qualified (Class 1E) circuit breaker could otherwise fulfill the literal interpretation of Section 3.8 by preventing (by its successful interruption of fault current) a malfunction in one section of a circuit from causing unacceptable influences (i.e., failures) in other sections of the circuit (or other circuits). To preclude Class 1E circuit breakers from these applications casts doubt upon the safety-related portions of the auxiliary power systems in all nuclear plants. Such systems invariably utilize Class 1E metal-clad switchgear and molded case circuit breakers as circuit protection devices.

The LaSalle County Station (LSCS) design complies with Positions 3 through 9.

With respect to the separation of associated circuits, the LSCS design complies with Section 4.5(3) of IEEE 384-1974.

Although separation of associated circuits in some cases differs from that dictated in Sections 5.1.3 and 5.6.2 of IEEE-384, Class 1E circuits are not degraded below an acceptable level for the following reasons:

a. Cables associated with one safety-related division are never routed in cable trays or conduit containing cables of a redundant safety-related division. This is true for all general plant areas. Lesser separation than that dictated by IEEE-384 occurs only within control panels located in the control room and auxiliary equipment room.

b. All cables used to interconnect associated circuits are the same high quality as that utilized in Class lE circuits, (i.e., all associated cables comply with the requirements of IEEE 383-1974). Therefore, this cable has been proven to be highly fire retardant by testing.

c. Cables with separation less than that dictated by Sections 5.1.3 and 5.6.2 of IEEE-384 are constrained to control and instrumentation circuits which, by their very nature, are low energy circuits. Control circuits are generally l20-Vac or 125-Vdc, whereas, the insulation rating of the cable utilized at LSCS is 600-V.

d. There are no power cables in contact with the control and instrumentation cables in the cable spreading area or in the control room and auxiliary equipment room. Also, there are no high energy sources located within control panels installed in these areas.

e. Fire stops are installed in the bottom entrances of all control panels.

With respect to the separation of non-Class 1E from Class 1E control and instrumentation circuits, LSCS complies with Section 4.6.2 of IEEE-384. Although the separation of non- Class 1E from Class 1E control and instrumentation circuits is in some cases, less than that required by Sections 5.1.3 and 5.6.2 of IEEE-384, these circuits have been analyzed to show that Class 1E circuits are not degraded below an acceptable level because:

a. Non-Class lE cables are routed in separate cable trays from Class 1E and associated cables in general plant areas.

b. Non-Class lE cables which come in close proximity to Class 1E and associated cables at one end do not come in contact with redundant Class 1E or associated circuits at their other end. This has been confirmed by a study of installed cables at LSCS.

c. All cables used to interconnect non-Class 1E circuits are the same high quality as that, utilized in Class 1E circuits, (i.e., all associated cables comply with the requirements of IEEE 383-1974).

   Therefore, this cable has been proven to be highly fire retardant by testing.

d. Cable separation less than that required by Sections 5.1 and 5.6 of IEEE-384 is limited to control and instrumentation circuits which by their very nature are low energy circuits. Control circuits are generally 120-Vac or 125-Vdc, whereas, the insulation rating of the cable utilized at LSCS is 600-V.

e. There are no power cables in contact with the control and instrumentation cables in the cable spreading area or in the control room and auxiliary equipment room. Also, there are no high energy sources located within control panels installed in these areas.

f Fire stops are installed in the bottom entrances of all control panels.

Position 10 refers to Section 5.1.2 of IEEE 384-1974 concerning cable and raceway identification. The LSCS design utilizes cable trays with permanent colored identification markers at each routing point which are assigned and alphanumeric code per Table 8.3-6 of the LSCS-FSAR. Each cable is assigned a number and segregation code. This information is placed on a colored tag, of permanent design, which is affixed to each end of the cable. A similar tag is also affixed to the cable where it enters and exits a penetration.

The LSCS design complies with Positions 11, 12, 13, 14, 15, and 16.

IEEE 384-1974, Section 4.6, requires that Non-Class lE cable trays be separated from Class lE cable trays by the following minimum separation requirements:

a. 1 ft horizontally and 3 ft vertically in cable spreading areas.

b. 3 ft horizontally and 5 ft vertically in general plant areas.

The LaSalle County Station (LSCS) criteria specifies that the minimum distance between safety-related and non-safety-related cable trays shall be 3 inches horizontally and 1 foot vertically. Cable trays at LSCS were installed to this specific criterion prior to issuance of IEEE-384.

At LSCS, the routing of safety-related cables and non-safety-related cables was controlled during design such that only a few such instances of close proximity exist in the auxiliary building.

Although LSCS criteria are less than that prescribed by IEEE-384, the LSCS cable tray installation is considered to be adequate for the following reasons:

a. There are no non—safety—related cable trays throughout the entire reactor building and primary containment.

b. There are no safety-related cable trays throughout the entire turbine building and outlying structures (service building, screen house, etc.).

c. Non-safety-related cable trays are installed in close proximity to safety-related cable trays only at a minimal number of locations in the auxiliary building.

d. In those few instances where non-safety-related cable trays come in close proximity to safety- related trays, only one division of safety-related trays are present in the room. The lone exception to this statement is the corridor between the diesel generator rooms and the switchgear room. However, trays in this area will be covered with a flame retardant material as a result of the Fire Hazards Analysis.

e. All cables used to interconnect non-safety-related circuits are the same high quality as that utilized in Class 1E circuits; (i.e., all non-safety-related cables comply with the requirements of IEEE 383-1974). Therefore, this cable has been proven to be highly fire retardant by testing.

f. Fire stops are installed in all cable trays, including non-safety-related cable trays, where they penetrate walls and floors.

g. All cable trays installed at LSCS are, for the most part, solid bottom cable trays. Therefore, where vertical stacking occurs, an additional barrier is provided between each layer of cable trays beyond that recognized by Section 5.1.1.3 of IEEE 384-1974.

<u>Regulatory Guide 1.75</u>

<u>Regulatory Position C6</u> Analysis performed in accordance with Sections 4.5(3), 4.6.2, and 5.1.1.2 should be submitted as part of the Safety Analysis Report and should identify those circuits installed in accordance with these sections.

<u>Applicants Position</u> The referenced analysis, when performed to justify deviation from specific requirements of standard IEEE 384-1974, shall be prepared on a case-by-case basis, shall be documented and be on permanent file, available for NRC review, but will not be an integral part of the safety analysis report.

Justification of Applicants Position The applicant's position is consistent with that taken for other plant design records; e.g., routine design calculations, design document revisions, etc.

REGULATORY GUIDE 1.76 – REV. 0

DESIGN BASIS TORNADO FOR NUCLEAR POWER PLANTS

The tornado design parameters are in accordance with the PSAR. Regulatory Guide 1.76 was issued after the construction permits were granted.

A comparison of the design parameters for the maximum wind velocity (360 mph) and the maximum pressure drop (3 psi) with the regulatory guide values indicates that these parameters are in accordance with the aforementioned guide. The design radius (227 feet) of maximum wind velocity is larger than that of the regulatory guide valued (150 feet) and therefore, the design tornado engulfs a larger portion of the structure.

The rate of pressure drop of the design tornado (1 psi/sec) is smaller than the value given in the regulatory guide (2 psi/sec). However, this difference in the rate of pressure drop has no effect on the design, since the natural periods of the building structures in comparison to either rate of pressure drop allow this load to be treated as a static load.

Thus the tornado design parameters are either equivalent or conservative compared to the parameters given in Regulatory Guide 1.76. Refer to Section 3.3 for additional discussion.

We believe that we comply with the guidance set forth in this regulatory guide.

REGULATORY GUIDE 1.76 – REV. 1

DESIGN-BASIS TORNADO AND TORNADO MISSILES FOR NUCLEAR POWER PLANTS

CONFORMANCE TO THE PROVISIONS OF REGULATORY GUIDE 1.76 – REV. 1 FOR ASSESSMENT OF TORNADO WIND LOAD ON DRY CASK STORAGE CONTAINER STABILITY:

The Plant design, relative to assessment of Dry Cask storage cask stability during handling activities when subjected to tornado loadings, conforms to the wind load definitions defined in Regulatory Guide 1.76 – Rev. 1.

REGULATORY GUIDE 1.77 – REV. 0

ASSUMPTIONS USED FOR EVALUATING A CONTROL ROD EJECTION ACCIDENT FOR PRESSURIZED WATER REAC'IORS

This regulatory guide is not applicable to LSCS.

REGULATORY GUIDE 1.78 – REV. 0

ASSUMPTIONS FOR EVALUATING THE HABITABILITY OF A NUCLEAR POWER PLANT CONTROL ROOM DURING A POSTUATED HAZARDOUS CHEMICAL RELEASE

Regulatory Guide 1.78 describes assumptions acceptable for use in assessing the habitability of the control room during and after a postulated external release of hazardous chemicals and describes criteria that are generally acceptable for the protection of the control room operators.

The requirements of this regulatory guide (for those plants that have received a construction permit prior to the issuance of this regulatory guide) are that the applicant should be in conformance to the criteria that were in effect at the time of the granting of the construction permit. The LaSalle County Station, Units 1 and 2, conform to all the criteria in this area that were in effect on the date that the construction permits were granted.

REGULATORY GUIDE 1.79 – REV. 1

PREOPERATIONAL TESTING OF EMERGENCY CORE COOLING SYSTEMS FOR PRESSURIZED WATER REACTORS

This regulatory guide is not applicable to LSCS.

REGULATORY GUIDE 1.80

PREOPERATIONAL TESTING OF INSTRUMENT AIR SYSTEMS

The preoperational testing of instrument air systems for the LaSalle County Station, Units 1 and 2, is in conformance with Regulatory Guide 1.80 in that all components of the instrument air system are tested to demonstrate that they will perform satisfactorily in service. The tests are designed to verify design response of safety-related equipment to a loss of instrument air accident and to verify that the failure of nonsafety—related components will not jeopardize safety-related components. The tests will also demonstrate the adequacy of operating procedures.

REGULATORY GUIDE 1.81 – REV. 1

SHARED EMERGENCY AND SHUTDOWN ELECTRIC SYSTEMS FOR MULTI-UNIT NUCLEAR POWER PLANTS

Regulatory Guide 1.81 describes an acceptable method for complying with GDC 5 of 10 CFR 50 Appendix A with respect to the sharing of onsite emergency and shutdown electric systems for multiunit nuclear power stations.

The criteria utilized in designing the two unit LaSalle County Station is that each unit shall operate independently of the other and any malfunction of equipment or operator error in one unit will not initiate a malfunction or error in the other unit nor affect the continued operation of the other unit.

The separate HPCS diesel-generators for LSCS Units 1 and 2 do not share other loads within each unit nor with other external units, therefore this guide is fulfilled. The HPCS is in electrical Division 3 and is isolated from electrical Divisions 1 and 2.

No safety-related systems, structures, or components are shared unless such sharing has been evaluated to ensure that there will be no significant adverse impact on safety functions.

Regulatory Guide 1.81 was issued after the LaSalle County Station construction permit. However, the design generally meets the intent of the positions of Regulatory Guide 1.81.

REGULATORY GUIDE 1.82 – REV. 0

SUMPS FOR EMERGENCY CORE COOLING AND CONTAINMENT SPRAY SYSTEMS

Regulatory Guide 1.82 applies to pressurized water reactors; therefore, it is not applicable to the LaSalle County Station.

REGULATORY GUIDE 1.83 – REV. 1

INSERVICE INSPECTION OF PRESSURIZED WATER REACTOR STEAM GENERATOR ‘IUBES

This regulatory guide is not applicable to LSCS.

REGULATORY GUIDE 1.84 – REV. 6

CODE CASE ACCEPTABILITY - ASME SECTION III
DESIGN AND FABRICATION

This guide provides a list of ASME Design and Fabrication Code Cases that have been generically approved by the Regulatory Staff as of June 1975. Code Cases on this list may, for design purposes, be used until appropriately annulled. Annulled cases are considered “active” for equipment that has been contractually committed to fabrication prior to the annulment.

The guide, Revision 6 and later, revisions, requires specific approval by the NRC on all ASME Section III, Class 1, 2, and 3 Code Cases not listed. NRC regulation 10 CFR 50.55a requires NRC approval only for Code Cases for Class 1 Components.. GE’s procedure for meeting the regulatory requirements is to obtain NRC approval for Code Cases applicable to Class 1 components only. NRC approval of Class 2 and 3 Code Cases ‘was not required at the time of the design of LSCS and is not required by 10 CFR 50.55a, however, all Class 2 and 3 equipment has been designed to ASME Code or ASME approved Code Cases.

Conformance for Class 1 components is accomplished by the use of only those ASME Code Cases which are listed as being acceptable in this Guide. Design and construction of the reactor vessel complies with Regulatory Guide 1.84 to the extent described as follows: Code Cases which were used were 1332-5, 1359-1, 1401, 1420, 1441-1 and 1459-1. For Code Case compliance, 1332-4 was authorized under provisions of Regulatory Guide 1.85 to allow use of a Code Case for which a subsequent revision (1332-6) has been approved in the Guide. Similarly, 1401 is allowed under the provision of Regulatory Guide 1.85 for use of a previously listed Code Case which has been subsequently annulled. Code Cases 1359-1, 1420, 1441-1, and 1459-1 were incorporated in the AMSE Code prior to issuance of the regulatory guides, so their application would also be consistent with the intent of the above provisions of the guides.

In addition to the approved Code Cases listed above, for Class 2 and 3 components LSCS is utilizing Code Cases N-237, N-240, and N-241 for which NRC approval has been requested.

Based upon the assessed capability of the Class 1 equipment utilized at LSCS, we believe that this facility is in compliance with the intent of this guide through incorporation of the alternate approach cited above.

Regulatory Guide 1.84

The NRC has approved the use of ASME Code Case N-411 for class 1,2, and 3 piping at LaSalle in a letter from E.G. Adensam (NRC) to D. L. Farrar (CECo) dated April 1, 1986.

REGULATORY GUIDE 1.85 – REV. 6

CODE CASE ACCEPTABILITY - ASME SECTION III MATERIALS

This guide provides a list of ASME Materials Code Cases that have been generically approved by the Regulatory Staff as of June 1974.

Code Cases on this list may, for design purposes, be used until appropriately annulled. Annulled cases are considered "active" for equipment that has been contractually committed to fabrication prior to the annulment.

The guide, Revision 6 and later revisions, require specific approval by the NRC on all ASME Section III, Class 1, 2, and 3 Code Cases not listed. NRC regulation 10 CFR 50.55a requires NRC approval only for Code Cases for Class 1 components. GE's procedure is to obtain NRC approval of Code Cases on Class 1 components only. NRC approval of Class 2 and 3 Code Cases was not required at the time of the design of LSCS and is not required by 10 CFR 50.55a; however all Class 2 and 3 equipment has been designed to ASME code or ASME approved Code Cases.

Conformance is accomplished by the use of only those ASME Code Cases which are listed as being acceptable in this guide. Design and construction of the reactor vessel complies with Regulatory Guide 1.85 to the extent described as follows. Code Cases which were used were 1332-5, 1359-1, 1401, 1420, 1441-1 and 1459-1. Of these, 1332-4 was authorized under provisions of Regulatory Guide 1.85 which allow use of a Code Case for which a subsequent revision (1332-6) has been approved in the Guide. Similarly, 1401 is allowed under the provision to regulatory guide 1.85 for use of a previously listed Code Case which has been subsequently annulled. Code Cases 1359-1, 1420, 14141-1, and 1459-1 were incorporated in the ASNE Code prior to issuance of the regulatory guides, so their application would also be consistent with the intent of the above provisions of the guides.

Based upon the assessed capability of Class 1 materials utilized in the RCPB at LSCS, we believe that this facility is in compliance with the intent of this guide via the alternate approach cited above.

Regulatory Guide 1.85

Code cases 1713 (endorsed in Revision 5) and N-242-1 (endorsed in Revision 18) of Regulatory Guide 1.85 has been adopted for use at LaSalle in a letter from H. L. Massin (BWR Engineering) to G. J. Diederich dated February 5, 1987.

REGULATORY GUIDE 1.86 - REV. 0

TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS

Regulatory Guide 1.86 describes acceptable methods and procedures for the termination of operating licenses for nuclear reactors.

The LaSalle County Station, Units 1 and 2, will conform to the legal requirements of the appropriate NRC regulations when the operating license is terminated.

REGULATORY GUIDE 1.87 - REV. 0

GUIDANCE FOR CONSTRUCTION OF CLASS 1 COMPONENTS IN ELEVATED TEMPERATURE REACTORS

Regulatory Guide 1.87 applies to high-temperature gas-cooled reactors (HTGRS), liquid metal fast-breeder reactors (LMFBR's) and gas—cooled fast—breeder reactors (GCFBR's); therefore, it does not apply to the La Salle Ccunty Station.

REGULATORY GUIDE 1.88 - REV. 2

COLLECTION, STORAGE, AND MAINTENANCE OF NUCLEAR POWER PLANT QUALITY ASSURANCE RECORDS

Regulatory Guide 1.88 describes a method acceptable to the NRC for complying with the requirements to collect, store, and maintain quality assurance records for nuclear power plants.

Per U.S. NRC letter from E.S. Beckjord dated June 17, 1991 (Subject: Withdrawal of Regulatory Guides), Revision 2 of Regulatory Guide 1.88 was withdrawn due to obsolescence.

The ANSI standards endorsed by the associated guides were incorporated into ANSI/ASME NQA-1-1983, "Quality Assurance Program Requirements for Nuclear Facilities." ANSI/ASME NQA-1-1983 was endorsed by Revision 3 of Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)", which was issued in August 1985.

Reference to Regulatory Guide 1.88 Revision 2 is being maintained for archival purposes.

UFSAR Chapter 17.0 cities Exelon Nuclear's QA Topical Report NO-AA-10 which is the updated version of EGC-1-A that described the QA program. That document outlines the Exelon document practices. The station fire protection afforded the archives storage area conforms to NFPA 232 criteria with the exception of documented deviations (See NFPA Code Report). This document was accepted by the NRC as a docketed record adequate for cross-referencing in licensing matters. The station central files is an area for the retention and reference of semicurrent records pending their ultimate disposition. It does not conform to NFPA 232 criteria. Fire protection is afforded by a fire detection system which automatically initiate preaction fire extinguishing system that has a local alarm, and an alarm in the control room. Additional protection is afforded by personnel occupancy during normal working hours and periodic patrols on nights, weekends and holidays.

The following information is also applicable to the station central file:

Completed Quality Assurance records will reside in this area for a nominal period of 1 year.

Completed Quality Assurance records are stored in fully enclosed metal cabinets. Adequate access and aisle ways are maintained at all times throughout the facility.

Smoking is prohibited.

Although a copying machine and two machines for making hard copy prints of aperture cards are located within the area, they are segregated from the records area. Another copying machine is located in a separate enclosed room. The machines for drawing reproduction are located in the front room and are separated from the records area by a central file service counter. The area is used for microfilming.

Access to the records area is restricted to authorized personnel.

The station central file is located on the fourth floor of the South Service Building.

Additional storage of records is provided for on the 2nd floor of the South (New) Service Building. Historical hard copies of safety related design calculations are maintained in this area for reference purposes. Filmed copies of these records are maintained on microfiche for quality records retention purposes in a QA vault in building 30.

The copies of more recent design calculation records are filed in this area while the records await filming to microfiche. These calculations are maintained for reference following filming.

The original hard copies of older non-safety related design calculation records are also maintained in this area.

The fire protection is a wetpipe system, which automatically initiates, and alarms locally and in the control room.

Exelon Nuclear complies with the objectives set forth in the referenced revision of this regulatory guide, with the exception noted above.

REGULATORY GUIDE 1.89 – REV. 1

ENVIROMENTAL QUALIFICATION OF CERTAIN ELECTRIC EQUIPMENT IMPORTANT TO SAFETY FOR NUCLEAR POWER PLANTS

Regulatory Guide 1.89, Revision 1 describes a method acceptable to the NRC staff for complying with 10 CFR 50.49 with regard to qualification of electric equipment important to safety for service in nuclear power plants to ensure that the equipment can perform its safety function during and after a design basis accident. LaSalle complies with the intent of RG 1.89, Revision 1 as follows.

a. Equipment installed prior to February 22, 1983 is either:

   1. Qualified in accordance with RG 1.89, Revision 1 (which endorses IEEE Standard 323-1974), or

   2. Qualified in accordance with NUREG-0588, Category II and IEEE Standard 323-1971, which meet the intent of RG 1.89, Revision 1

b. New and replacement equipment is qualified in accordance with RG 1.89, Revision 1 unless there are sound reasons to the contrary.

c. The guidance does not include requirements for qualifying equipment in mild environment.

REGULATORY GUIDE 1.90- REV. 0

INSERVICE INSPECTION OF PRESTRESSED CONCRETE CONTAINMENT STRUCTURES WITH GROUTED TENDONS

This regulatory guide is not applicable to the LaSalle County Station, Units 1 and 2, since this guide does not cover the type of containment utilized on LSCS.

REGULATORY GUIDE 1.91- REV. 0

EVALUATION OF EXPLOSIONS POSTULATED TO OCCUR ON TRANSPORTATION ROUTES NEAR NUCLEAR POWER PLANT SITES

Regulatory Guide 1.91 describes a method acceptable to the NRC for determining safe distances from a nuclear power plant to a transportation route over which explosive material (not including gases) may be carried.

The applicant complies with Regulatory Guide 1.91 in the sense that the tornado design pressure is higher than the overpressures of the postulated explosions on the nearby transportation routes. For details, refer to Section 2.2.

We believe that we comply with the objectives set forth in this regulatory guide.

REGULATORY GUIDE 1.92- REV. 1

COMBINATION OF MODES AND SPATIAL COMPONENTS IN SEISMIC RESPONSE ANALYSES

This regulatory guide describes methods acceptable to the NRC for combining the values of the response spectrum modal dynamic analyses, and combining maximum values of a time-history dynamic analyses or in combining maximum values of a spectrum dynamic analyses.

a. Combination of Modal Response

   1. In response spectrum method of analysis, the probable ma>~imum response is obtained by using the double sum criterion as explained in Subsection 3.7.2.7 for BOP structures, and in Subsection 3.7.3.7 for BOP and NSSS subsystems. The method takes into account the separated modes as well as the closely spaced modes.

2. All piping and equipment analyzed or supplied by General Electric are evaluated to the requirements of this regulatory guide as described in Subsection 3.7.3.7.

b. Combination of Spatial Components

1. Time-History Method

When the time—history dynamic analysis is used and the three spatial components are statistically independent, the response due to each spatial component is algebraically added at each time instant and the maximum is obtained from the combined response.

2. Response Spectrum Method

For NSSS components and equipment, the calculated maximum responses due to one horizontal directional earthquake excitation are combined with the responses due to the vertical earthquake by the sum of the absolute values method. The maximum responses due to another perpendicular horizontal earthquake are also combined with the responses due to the vertical earthquake in the same manner. The larger of the two values is used for the design,. (i.e., the larger of [x+y] and [y+z] where x and z are horizontal and y is vertical.

REGULATORY GUIDE 1.93 – REV. 0

AVAILABILITY OF ELECTRIC POWER SOURCES

Regulatory Guide 1.93 describes acceptable operating procedures and restrictions which should be implemented if the available electric power sources are less than the limiting conditions of operation.

Regulatory Guide 1.93 was issued after the LaSalle County Station construction permit. The LSCS design generally meets the intent of the regulatory positions, as outlined in Chapter 8.0. A specific network stability analysis for LSCS is included in Section 8.2.

REGULATORY GUIDE 1.94 – REV. 1

QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION, AND TESTING OF STRUCTURAL CONCRETE AND STRUCTURAL STEEL DURING THE CONSTRUCTION PHASE OF NUCLEAR POWER PLANTS

The construction of LaSalle County Station was well underway at the time Regulatory Guide 1.94 was published. However, all quality assurance programs for LSCS are in line with ANSI N45.2.5-1974 except that in some cases the type and frequency of these tests are not identical. For LSCS quality control programs on structural concrete and steel see Appendix E.

LaSalle County Station has cadwelded all positive connections for reinforcing bars; therefore the section on welding reinforcing bar splices is not applicable.

REGULATORY GUIDE 1.95 – REV. 0

PROTECTION OF NUCLEAR POWER PLANT CONTROL ROOM OPERATORS AGAINST AN ACCIDENTAL CHLORINE RELEASE

The only potential source for accidental chlorine release are chlorine-carrying barges which occasionally pass about 5 miles north of the Site. The effects of this accident are discussed in Subsection 2.2.3.2.

The detailed description of the control room HVAC system covered under Section 6.4 and Subsection 9.4.1 of LSCS-UFSAR.

The detection systems installed at LSCS are consistent with the recommendation of this regulatory guide.

REGULATORY GUIDE 1.96 – REV. 1

DESIGN OF MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEMS FOR BOILING WATER REACTOR NUCLEAR POWER PLANTS

LSCS Unit 1 MSIV-LCS has been abandoned in place, and Unit 2 MSIV-LCS has been removed.

REGULATORY GUIDE 1.97 – REV. 2

Regulatory Guide 1.97 Rev. 2, Instrumentation for Light-Watered-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions during and following an Accident, describes a method acceptable to the NRC for monitoring plant variables and systems during and following an accident. LaSalle is committed to compliance with Reg. Guide 1.97 Rev. 2 as documented in reference 1 with noted exceptions and deviations. Additional information and clarification was provided to the NRC in response to an interim report to CECo from the NRC. (References 2 and 3). A final report summary of Human Factors Review of Reg. Guide 1.97, Rev. 2 was later submitted to the NRC (Ref. 4). The NRC accepted the Commonwealth Co. position for LaSalle County Station Units 1 and 2 as complying with the recommendations of Reg. Guide 1.97 Rev. 2 with the exception that CECo commit to installing and having operational monitoring instrumentation for neutron flux which fully meets the recommendations of Reg. Guide 1.97, Rev. 2 (Reference 5). Commonwealth Edison responded in later correspondence that LaSalle would follow the recommendations of the BWR Owners Group in regard to upgrade of the neutron monitoring system when available. (Reference 6). The NRC accepted ComEd's submittal of compliance to Reg. Guide 1.97 Rev. 2 for neutron flux instrumentation with justification for non-conformance from the NEDO 31558-A that is consistent with the BWROG position (References 7 and 8).

In response to an NRC inspection of LaSalle for compliance with Reg. Guide 1.97, additional clarification was provided in regard to the noted deviations of the instrumentation ranges of the Containment Hydrogen Concentration Monitors and the Reactor Vessel Water Level (References 9 and 10). In addition a concern for the acceptability of electrical isolation devices between Reg. Guide 1.97 instrument circuitry and the non-safety process computer was addressed by CECo with analysis and testing (References 11 and 12)

References

1. 6/29/82 Letter from C.W. Schroeder CECO to A. Schwencer NRC on Commonwealth Edison's position of Reg Guide 1.97 for LaSalle Station with noted exceptions.
2. 12/13/84 NRC letter from A Schwencer to Dennis L. Farrar CECO, Interim Report on CECo response to Reg Guide 1.97 Rev 2.
3. 2/22/85 letter from G.L. Alexander CECO to H. R. Denton NRC on CECO response to NRC interim report on CECO compliance to Reg Guide 1.97 Rev 2.
4. 8/1/86 letter from C. M. Allen CECO to H. R. Denton NRC on LaSalle Human Factors Review of Reg Guide 1.97 Instrumentation.
5. LaSalle Reg Guide 1.97 SER transmitted to L. D. Butterfield on 8/20/87 from D. Muller on Emergency Response Capability, Conformance to Reg Guide 1.97 Rev 2 for LaSalle Unit 1 and 2.

6. 10/7/87 letter from C.M. Allen CECO to NRC Document Control Desk on Neutron Flux Monitoring.
7. 6/21/99 letter from Jeffery A. Benjamin to the NRC on Compliance with Reg. Guide 1.97 - BWR Neutron Flux Monitoring.
8. 9/7/99 NRC letter from Donna M. Skay NRC to O. Kingsley ComEd on Reg Guide 1.97 - BWR Neutron Flux Monitoring - LaSalle County Station Units 1 and 2 (TAC No. M77660).
9. 12/5/88 NRC letter from Hubert J. Miller NRC to Commonwealth Edison, Notice of violation for items identified by RG 1.97 Inspection. Unresolved items.
10. 3/2/89 CECO Response to NRC from W. E. Morgan CECo to A Bert Davis NRC.
11. 11/9/90 letter from W. E. Morgan CECo to A Bert Davis, NRC on Response to Inspection Report 373-90022 and 374-90023.
12. 6/21/91 letter from Peter Piet CECo to NRC on Supplement to response on Inspection Report 373-90022 and 374-90023.

Regulatory Guide 1.99 Rev. 2

Regulatory Guide 1.99 Rev. 2, “Radiation Embrittlement of Reactor Vessel Material,” date May 1988, required updating Technical Specification Pressure/Temperature Curves and have been approved by NRC in Amendments 71 and 55 dated for LaSalle Unit 1 and Unit 2 respectively.

REGULATORY GUIDE 1.100, – REV. 1

SEISMIC QUALIFICATION OF ELECTRIC AND MECHANICAL EQUIPMENT FOR NUCLEAR POWER PLANTS

Regulatory Guide 1.100, Rev. 1, describes a method acceptable to the NRC for seismic qualifications testing of electric equipment. LaSalle complies with the intent of the guidance provided in that:

a) Equipment qualified prior to the issuance of the Facility Operating License was either:

1) Qualified in accordance with Regulatory Guide 1.100, Rev. 1, (which endorses IEEE 344-1975, with a few limitations), or

2) Qualified in accordance with IEEE 344-1971 and evaluated and found to meet the intent of Regulatory Guide 1.100, Rev. 1.

b) New and replacement (except for "like-for-like" replacement) equipment is qualified in accordance with Regulatory Guide 1.100, Rev. 1.

c) The guidance is also applied to mechanical equipment and instrumentation that is required to be seismically qualified.

REGULATORY GUIDE 1.106, – REV. 1

THERMAL OVERLOAD PROTECTION FOR ELECTRIC MOTORS ON MOTOR-OPERATED VALVES

Regulatory Guide 1.106, Rev. 1, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves," describes a method acceptable to the NRC for design, erection, and testing of thermal overload protection for electric motors on motor-operated valves important to safety.

Except for the Hydrogen Recombiner System motor-operated valves, both NSSS and Non-NSSS safety-related equipment meet the requirements of the Regulatory Position, C.1, using both alternate positions, as determined to be appropriate based on the design function of each motor-operated valve. The Hydrogen Recombiner System motor-operated valves meet Regulatory Position C.2. The methodology used to meet Regulatory Position C.2 is in accordance with IEEE-741-1990. More information concerning actual implementation of this Regulatory Guide is discussed in Subsection 6.3.2.2.13.

REGULATORY GUIDE 1.108

PERIODIC TESTING OF DIESEL GENERATORS USED AS ONSITE ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS

Regulatory Guide 1.108 describes a method acceptable to the NRC for preoperational and periodic testing of diesel electric power units to ensure their availability requirements (0.99 reliability at 50% confidence).

This Regulatory Guide has been superseded by Regulatory Guide 1.9, Revision 3.

REGULATORY GUIDE 1.116 – REV. 0

QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION, AND TESTING OF MECHANICAL EQUIPMENT AND SYSTEMS

This guide describes a method acceptable to the NRC for fulfilling QA requirements on the installation, inspection, and testing of mechanical equipment and systems for water- cooled nuclear power plants. ANSI N45.2-1975 is used to delineate preoperational tests and other functional tests.

The Exelon Nuclear Topical Report which outlines the Exelon Nuclear QA program with respect to inspection, and testing of safety-related equipment applies to these mechanical systems.

We comply with the objectives of the referenced revision of this regulatory guide as indicated by the practices referenced above.

REGULATORY GUIDE 1.118 – REV. 1

PERIODIC TESTING OF ELECTRIC POWER AND PROTECTION SYSTEMS

Regulatory Guide 1.118 describes a method acceptable to the NRC for periodic testing of the protection system and electrical power systems important to safety. This is the implementation of IEEE Standard 338-175 which is ancillary to IEEE 279-1971 and IEEE 380-1974.

The NSSS design complies with this guideline to enable response time testing based on relay contact internals, but not sensor responses to stimulating media. Relative to the regulatory position given in paragraphs C.l through C.7, the LSCS design generally meets the intent of this guide with the exception of provisions for response time testing. Because of the issue date of this document, no specific design features were originally included in the non-NSSS systems to facilitate response time testing.

NRC Implementation: To be used in the evaluation of submittals with construction permit applications docketed after January 1, 1978.

REGULATORY GUIDE 1.123 – REV. 1

QUALITY ASSURANCE REQUIREMENTS FOR CONTROL OF PROCUREMENT OF ITEMS AND SERVICES FOR NUCLEAR POWER PLANTS

This regulatory guide describes a method acceptable to the NRC for fulfilling QA requirements relative to the procurement of items and services during design, construction and operations phases of nuclear power plants.

The LSCS FSAR was docketed with the OL application prior to issuance of this regulatory guide. Nevertheless, the Edison "QA Progran for Nuclear Generating Stations," CE-1-A app1ies to procurement of modification items and services for safety-related systems as outlined in that Topical Report.

Edison complies with the objectives of the referenced revision of this regulatory guide.

REGULATORY GUIDE 1.145, – REV. 1

ATMOSPHERIC DISPERSION MODELS FOR POTENTIAL ACCIDENT CONSEQUENCE ASSESSMENTS AT NUCLEAR POWER PLANTS

Regulatory Guide 1.145 describes the methodology for determining the offsite atmospheric dispersion relative concentrations (X/Q) values on a directional-dependent basis and overall site basis for both stack and ground-level releases.

The computer code PAVAN implements the guidance provided in this regulatory guide and was utilized for calculating the Alternative Source Terms (AST) offsite X/Q values, and is in full compliance with this regulatory guide.

REGULATORY GUIDE 1.166

PRE-EARTHQUAKE PLANNING AND IMMEDIATE NUCLEAR POWER PLANT OPERATOR POST-EARTHQUAKE ACTIONS

Regulatory Guide 1.166 provides guidance acceptable to the NRC staff for a timely evaluation after an earthquake of the recorded instrumentation data and for determining whether plant shutdown is required by 10 CFR Part 50. Refer to subsection 3.7.4.5 for further information.

Lasalle station has voluntarily implemented the methods described in this regulatory guide, as allowed by Section D of the regulatory guide.

REGULATORY GUIDE 1.167

RESTART OF NUCLEAR POWER PLANT SHUT DOWN BY A SEISMIC EVENT

Regulatory guide 1.167 provides guidance acceptable to the NRC staff for performing inspections and tests of nuclear power plant equipment and structures prior to restart of a plant that has been shut down by a seismic event. Refer to subsection 3.7.4.5 for further information.

Lasalle station has voluntarily implemented the methods described in this regulatory guide, as allowed by Section D of the regulatory guide.

REGULATORY GUIDE 1.181, – REV. 0 (09/99)

CONTENT OF THE UPDATED FINAL SAFETY ANALYSIS REPORT IN ACCORDANCE WITH 10 CFR 50.71(e)

As part of the ongoing effort to improve the quality of the UFSAR, the guidelines provided in NEI 98-03 Revision 1, 06/99 and endorsed by the NRC in Regulatory Guide 1.181, Revision 0, 09/99 will be used to further improve the content of the UFSAR. While the UFSAR will continue to follow the general organizational recommendations specified in Regulatory Guide 1.170, Rev. 2 (09/75), the reformatting options described in NEI 98-03, Revision 1, will be used to simplify information contained in the UFSAR to improve their focus, clarity and maintainability. Examples of the types of improvements that are anticipated are the identification of historical information and the removal or replacement of excessively detailed drawings as described in Appendix A to NEI 98-03 Revision 1.

REGULATORY GUIDE 1.183, – REV. 0

ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS

Regulatory Guide 1.183 provides acceptable assumptions that may be used in evaluating the radiological consequences of design basis accidents for a boiling water reactor using Alternative Source Terms (AST). The assumptions used for evaluating the radiological consequences of a LOCA and FHA in this UFSAR for AST are in full conformance with the requirements of this regulatory guide except ventilation filter efficiencies and fuel rod burnup limits for ATRIUM-10 partial length rods. For details see Section 6.5.1, 15.6.5 and 15.7.4, respectively. The requirements of this guide were incorporated in the evaluation of LSCS Loss of Coolant and Fuel Handling Accidents.

REGULATORY GUIDE 1.190, DATED MARCH 2001

CALCULATIONAL AND DOSIMETRY METHODS FOR DETERMINING PRESSURE VESSEL NEUTRON FLUENCE

Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence” provides state of the art calculation and measurement procedures that are acceptable to the NRC for determining Reactor Pressure Vessel (RPV) neutron fluence.

In Technical Specification Amendments 160 (Unit 1) and 146 (Unit 2), the NRC approved LaSalle’s participation in the Integrated Surveillance Program. Specific details are provided in UFSAR section 4.1.4.5. However, as a condition of granting the Amendments, LaSalle committed that all future evaluations of RPV fluence will be completed using a method in accordance with the recommendations of Regulatory Guide 1.190.

REGULATORY GUIDE 1.194 – REV. 0

ATMOSPHERIC RELATIVE CONCENTRATIONS FOR CONTROL ROOM RADIOLOGICAL HABITABILITY ASSESSMENTS AT NUCLEAR POWER PLANTS

Regulatory Guide 1.194 presents guidance for utilizing the computer code ARCON96 for calculating ground-level and stack atmospheric dispersion relative concentrations (X/Q) values for Control Room Habitability Assessments.

The guidance presented in this regulatory guide was utilized for Alternative Source Terms (AST) onsite X/Q calculation, and is in full compliance with this Regulatory Guide.

REGULATORY GUIDE 1.196 – REV. 0

CONTROL ROOM HABITABILITY AT LIGHT WATER NUCLEAR POWER REACTORS

The licensee has only committed to section C.2.7.3 "Degraded and Nonconforming Condition" of Regulatory Guide 1.196.

The Control Room Envelope Habitability Program is governed by the Technical Specification 5.5.15 approved under License Amendment No. 186 to Unit 1 and 173 to Unit 2. The Technical Specification Program 5.1.15 is a result of a commitment in response to NRC Generic Letter 2003-01 by implementing the requirements of Consolidated Line Item Improvement Process for implementation of TSTF-448 revision 3.

Survey of chemical sources is to be performed on a 6 year frequency as part of the periodic assessment of CREH program. References to Regulatory Guides 1.78 and 1.95 are to be interpreted as Regulatory Guides 1.78 and 1.95, revision 0.

<u>REGULATORY GUIDE 1.197 – REV. 0</u>

<u>DEMONSTRATING CONTROL ROOM ENVELOPE INTEGRITY AT NUCLEAR POWER REACTORS</u>

The Licensee has only committed to the requirement for determining the unfiltered inleakage into the CRE in accordance with the testing methods and at frequencies specified in section C.1 and C.2 of Regulatory Guide 1.197, revision 0 as described by Technical Specification 5.5.15 "Control Room Envelope Habitability Program."

APPENDIX C

ANALYSIS OF EFFECTS OF HIGH-ENERGY LINE BREAKS OUTSIDE PRIMARY CONTAINMENT

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APPENDIX C

ANALYSIS OF HIGH-ENERGY LINE BREAKS OUTSIDE PRIMARY CONTAINMENT

The analytical methods and results of the high-energy line break analysis for the LaSalle plant is documented in this Appendix.

APPENDIX C

ANALYSIS OF EFFECTS OF HIGH-ENERGY LINE BREAKS OUTSIDE PRIMARY CONTAINMENT

C.1 ABSTRACT

The purpose of this appendix is to record in detail the methods used in the pipe whip analysis for the LaSalle County Station (LSCS). It also contains the results of the pipe whip and jet impingement analysis of systems outside the containment vessel at LSCS, as well as an exposition of the method of analysis and the criteria for postulating breaks. The conclusions obtained for LSCS Unit 1 are also appropriate for LSCS Unit 2.

The appendix concludes with the action taken in the specific areas or components needing protection and in the design of restraints required for this purpose.

C.2 INTRODUCTION

Piping systems which exceed temperatures above 200°F and/or pressures above 275 psi during "normal plant conditions" are classified as high-energy lines according to NRC guidelines. These high-energy lines are considered as possible sources of pipe whip. Some systems, such as the RHR system, exceed these criteria less than 2% of the time that the systems operate as moderate energy systems, and therefore, do not qualify as high energy systems. See MEB 3-1, 8/5/77, paragraph B.2.c.

The postulation of circumferential and longitudinal breaks on high-energy lines of 4-inch diameter or more is based on the A. Giambusso letter of December 15, 1972 which recommends that the breaks be considered at points where stresses exceed 0.8 ($S_A + S_h$).

However, per Branch Technical Position MED 3-1 (Revision 0), the postulation of Class 2 and 3 piping (ASME Code, Section III) pipe ruptures in areas other than containment penetration is postulated at the following locations in those portions of each piping and branch run:

a. At terminal ends.
b. At intermediate locations selected by the following criteria:

   At each location where the stresses exceed 0.8 (1.2 $S_h + S_A$) but at not less than two separated locations chosen on the basis of highest stress. Where the piping consists of a straight run without fittings, welded attachment, or valves, and all stresses are below 0.8 (1.2 $S_h + S_A$), a minimum of one location chosen on the basis of highest stress.

Normally break locations are assumed at locations that would maximize EQ (Environmental Qualification Values) impact not on pipe stress. The above break exclusion criteria is applied when the postulated break analysis produced consequences which are too conservative.

The following systems were analyzed:

a. feedwater heater drain lines,
b. reactor core isolation cooling lines,
c. reactor water cleanup lines,
d. condensate booster lines,
e. extraction steam lines,
f. control rod drive hydraulic lines,
g. main steamlines, and
h. feedwater lines.

The effects of pipe rupture in all of the above systems on essential components and structures were determined for the case of both pipe whip and jet impingement.

The method of analysis for pipe whip is obtained in Attachment C1.

There are no safe shutdown SSCs in the immediate vicinity of breaks postulated for the extraction steam system and the feedwater heater drain lines. Therefore, no further evaluation is necessary of these systems.

Double-ended guillotine breaks are postulated in Rooms 20A, 21A, 22A, 23A, 26, 27, 28A, 29A, 30A, 30B, 30C, 36, 45A, 48, 49, and 50 throughout the reactor water cleanup (RT) piping.

C.3 DEFINITIONS

See Attachment C1.

C.4 METHOD AND CRITERIA USED TO DETERMINE THE NECESSITY OF A COMPLETE ANALYSIS

Breaks are postulated by marking the high-energy lines on a set of composite drawings at every pipe fitting and identifying the breaks in numerical sequence. (The original composite drawings are historical and have not been maintained.) If it is determined that there is no component or structure affected by a particular break, no further analysis is required. If a component or structure is affected, further analysis is only required when:

a. The component is essential to ensure safe shutdown of the plant or is required to mitigate the consequences of the accident.
b. The ruptured pipe collides with or otherwise affects another pipe of equal or smaller size.
c. The component is a cable pan carrying electrical safety-related hardware.
d. The structure is Seismic Category I or safety-related Class I.
e. Examination of the composite drawings may or may not provide sufficient information to enable the determination of whether or not any of the conditions (above items a through d) are applicable. In case sufficient information is not available, it is assumed that conditions of items a through d pertain.

C.5 PIPE WHIP

Based on the information contained in the “PWUR” computer output, certain areas in the steam tunnel are provided with pipe whip protection. Restraints RMS-1 through 8 and RFW-1 through 9 were located at selected points for this purpose. In order to meet the valve vendors allowable loading, it was necessary to include additional restraints RFW-10 and RFW-11 and design restraints RMS-1, RMS-2, RMS-3, RMS-4, RFW-1, and RWF-2 to allow deflections of 0.3 inch or less at that location. These restraints are located as shown in Figure C-1.

In the above-grade areas of the Turbine Driven Reactor Feed Pumps (TDRFP) rooms, additional restraints were provided to protect structural components.

The postulated pipe rupture orientation and number of design basis breaks used in the analysis of high energy systems outside containment are shown in Table C-1. The individual break locations within each system and the whip restraints are provided in Table C-2. Figures C-2 through C-5 schematically illustrate the information provided in Table C-2.

It is normally recommended that a pipe whip restraint be connected to a pipe at a distance from the elbow less than the minimum distance to plastic hinge formation. This ensures that a hinge cannot form ahead of the restraint or at the restraint itself.

C.6 METHOD FOR CALCULATING THE JET IMPINGMENT FORCE

The models used for calculating jet impingement forces are shown in Figures C-6 and C-7.

The impingement force on a target is found by first calculating the average pressure of the jet at the location of the target according to the equation.

$$P_{jet} = \frac{F}{A_{jet}}$$

Where

P_{jet}	=	average jet pressure at the required location
F	=	blowdown force
A_{jet}	=	area of the jet at the required location,

The impingement force is then

F_{imp}	=	impingement force on target

Where

F_{imp}	=	impingement force on target
K_o	=	shape factor
P_{jet}	=	average jet pressure as defined above
A_{target}	=	area of the target.

Affected cable pans that cannot withstand these forces will be rerouted or shielded from the jet.

C.7 CONCLUSIONS

Analysis of the effects of high-energy line breaks is performed to determine where pipe whip restraints are necessary. Based on the information provided by the analysis, pipe whip restraints are provided where needed to protect essential components or structures. Likewise, analysis of jet impingement due to high-energy line breaks is performed to determine if shields are necessary to protect essential components or structures.

Subsequently to the original analysis presented here, the Class 1 piping stress reports were completed. At that time the piping stress and fatigue factors were used to identify potential HELB's in accordance with the appropriate guidelines (USNRC Branch Technical Position MEB 3-1). The use of the pipe stress results reduced the number of required whip restraints and also eliminated 19 of the original 22 "jet impingement hits" on cable pans. With this reduced number of pipe breaks, combined with the Safe Shutdown Analysis (UFSAR Appendix H) and the EQ System Safety Evaluation, it is concluded that neither jet impingement shields nor equipment shields are required. This conclusion is based on the physical independence or alternate safety systems designed for safe reactor shutdown and removal of decay heat from the shutdown reactor system.

ATTACHMENT C1

DESCRIPTION OF "PWUR" COMPUTER PROGRAM - ENERGIES AND LOADS FOR UNRESTRAINED PIPE WHIP

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ATTACHMENT C1

ENERGIES AND LOADS FOR UNRESTRAINED PIPE WHIP

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C1.0 INTRODUCTION

According to NRC regulatory guides, essential (safety-related) systems and components required to safely shutdown the plant need to be protected from the effects of high energy pipe breaks, including pipe whip.

Depending on the nature of the break, four different kinds of protection can be provided against pipe whip:

a. reinforcement, such as in the case of a structure;
b. dissipation of energy by a restraint;
c. shielding plates such as in the case of jet impinging upon a cable pan; and
d. separation criteria for systems which operate in different areas of the containment.

The study of the effect of pipe whip considering an individual component leads to the following conclusions:

a. The component is capable of assimilating the impact energy generated by pipe whip.
b. The component would be able to assimilate the impact energy if reinforcement is provided.
c. The component cannot be reinforced and is not suited to assimilate the energy of impact.

The object of the PWUR computer program is to provide information and necessary documentation in aspects such as:

a. source and nature of break;
b. break number and type;
c. affected component, whether it is a wall, another pipe, valve, electrical or mechanical equipment;
d. location of the break and affected component;
e. energies and jet impingement loads resulting from circumferential or longitudinal breaks;
f. plastic hinge location; and
g. minimum distance to plastic hinge formations.

C2.0 DEFINITIONS

C2.1 Pipe Breaks

C2.1.1 Circumferential Break

Circumferential breaks are perpendicular to the pipe axis. Dynamic forces resulting from such breaks are assumed to separate the pipe axially and cause whipping in the plane determined by the pipe axis and the next pipe run.

C2.1.2 Longitudinal Break

Longitudinal breaks are parallel to the pipe axis. Dynamic forces resulting from such breaks act perpendicular to the pipe axis. The nature of this break is two-fold:

a. The pipe deforms plastically and impacts surrounding components. This situation is referred to as "longitudinal impact."

b. A jet is created in the opposite direction of the force deforming the pipe and impinges upon other component(s). This situation is referred to as "jet impingement."

Longitudinal and circumferential breaks do not occur simultaneously at the same location on the pipe, although for analysis purposes, they may be postulated as such when essential components can be affected by either one.

C2.2 Blowdown Force

The sudden change of pressure when a break occurs has a two-fold effect:

a. A thrust force of magnitude equal to the initial internal pressure times the break area is developed.

b. The flow velocity increases from zero to a limiting terminal velocity, that is, if it is assumed that the pressure P_o inside the pipe remains constant for as long as it takes the flow to reach V_T. Therefore, the initial stage of the flow where only the static pressure P_o exerts a total force of P_o times the blowdown area changes to a stage where the total blowdown force is the result of the static pressure plus dynamic pressure times the blowdown area. F. J. Moody (Reference 1) has shown that for an ideal gas under isentropic conditions this total force is

$$F = 1.26 \times P_o \times A$$

P_o = internal operating pressure
A = break area.

The blowdown thrust for subcooled water was derived using $2P_oA$ as the steady-state value for frictionless non-flashing water. The magnitude of the steady-state thrust coefficient varies from 1.26 to 2.0 for frictionless fluid depending on the degree of subcooling.

However, the steady-state blowdown forces are reduced by taking frictional effects into consideration.

Therefore, the blowdown thrust values used for calculating pipe motion take into account frictional effects of the piping system.

C2.3 Plastic Hinge

A plastic hinge is the point on the pipe where the pipe will deform plastically under stress caused by bending moments.

C2.4 Plastic Hinge Distance

When a force is considered acting at a point of a pipe with infinite length, the distance from the force action point to the plastic hinge is defined as the plastic hinge distance.

C2.5 Minimum Distance to Plastic Hinge Formation

If the pipe has a terminal end or a restraint located along its axis, then the pipe will become stiffer than a pipe of infinite length. However, if the constraint is not applied within the plastic hinge distance, it will behave as a pipe of infinite length. When the constraint is within this distance, the plastic hinge will form exactly at the constraint point. As the constraint is moved toward the force application point, the pipe becomes stiffer until it reaches a point when the couple induced by the constraint and the force is equal to the plastic moment of the pipe. The distance from the point of application of the force to this point is labeled as “The Minimum Distance to Plastic Hinge Formation” and is given by

$$L_S = \frac{M_O}{F_B}$$

where:

M_o = plastic moment of the pipe.

C2.6 Terminal Ends

Terminal ends of pipe runs originate at any point of maximum constraint such as components rigidly anchored to structures, anchor, penetrations, elbows, or restraints.

C2.7 Restraint

A restraint is a structural component located around the pipe for the purpose of constraining the pipe. In some cases the restraint is designed to yield under the load caused by the break, but the energy assimilated by the restraint in the deformation process decreases the effect of the pipe impact on the component.

C2.8 Pipe Run

Usually taken as the distance between two consecutive elbows or between the elbow adjacent to the break and the next possible place where a hinge could form, such as a terminal end. See Figures C1-1 and C1-2.

C2.9 Distance to Affected Structure

For a circumferential break, the distance to the affected structure is taken as the space separating the elbow adjacent to the break and the structure or component located along the line of action of the blowdown force. See Figure C1-1.

For a longitudinal break, this distance is the space separating the point where the break occurs and the structure located along the line of action of the thrusting force. See Figure C1-2.

C3.0 PROCEDURE

The simple model shown in Figure C1-4 is used (Reference 2) to calculate energies imparted by two objects colliding plastically. The mass of the pipe shown in the longitudinal position is lumped into an equivalent mass concentrated at the tip of the pipe along with the existing tip weight, which is the weight of the elbow plus the portion of pipe where the circumferential break occurs.

The velocity at which the pipe moves is assumed to be linear, which implies constant acceleration, which in turn implies that the limiting blowdown force is constant.

C4.0 INTERPRETATION OF RESULTS

Depending on the nature of the break and the local configuration of the pipe, the following cases are possible:

Under the circumferential break conditions (see Figure C1-5):

a. If the pipe run provided is too small for a plastic hinge to form there will be a message generated "No plastic hinge formed along pipe run." This condition exists when the pipe run length is less than the minimum distance for plastic hinge formation.
b. If the pipe run provided is large and exceeds the value at which the pipe will hinge under unrestrained conditions. In this case, a message "Hinge formed at a distance 'S' from break elbow" will appear.
c. If the pipe run provided is large and exceeds the minimum distance for hinge formation, but not the 'S' value. The message "Hinge formed at the end of the run" will depict this condition.

Under longitudinal break conditions:

d. For a given pipe length there is a force F that will deform the pipe plastically. If this condition is not met, the message "Force is not large enough to deform pipe plastically" will appear.
e. The longitudinal break model considers the force applied at the midsection of the pipe. If this is not the case, i.e., the force is applied closer to one end, the smaller distance to the elbow or possible hinge point is compared with the minimum distance for hinge information. In the event that the force application point clears this last distance, then the force is considered as if it were applied at the midsection. But if the point of application does not clear the minimum distance for hinge formation, then the message "L break too close to elbow and must be considered separately, see documentation" will appear.

Since the program is not designed to store the entire configuration of the pipe and/or the surrounding components, pipes structures, etc., changes cannot be made when the aforementioned conditions of items a or e arise. These cases are reformulated after the results are available according to the following considerations:

For case a:

Reformulate the c-break by considering the pipe run, input previously, as part of the portion of pipe where the break occurs such as indicated in Figure C1-6. Thus, this "new" portion will be composed of the "old" portion of pipe where the break occurs;

the “old” pipe run plus the additional pipe portion ranging between the end of the “old” run and the nearest elbow. Close observation of this pipe configuration indicates that a longitudinal break in impact may need to be formulated as well, if there are components at a radial distance from pipe. See Figure C1-7.

For case e:

Two possible considerations are applied in this case:

a. The portion of pipe labeled “D”, shown in Figure C1-8, is less than the minimum distance for plastic hinge formation. Therefore, the two elbows adjacent to “D” are disregarded as possible hinge formations and the break is reformulated as shown in Figure C1-9.
b. The portion of pipe labeled “D” is greater than L_s, as shown in Figure C1-10; therefore, no longitudinal break in impact can exist at this location, and no formulation is required.

C5.0 REFERENCES

1. F. J. Moody, “Prediction of Blowdown Thrust and Jet Forces,” ASME Trans., 69-HT-31, 1969.

2. H. G. Hopkins, “On the Behavior of Infinitely Long Rigid Beams Under Transverse Concentrated Load,” J Mech. Phys. Solids, Vol. 4, 1955.

TABLE C-1

POSTULATED RUPTURE ORIENTATION AND NUMBER OF DESIGN BASIS BREAKS USED IN RUPTURE ANALYSIS OUTSIDE CONTAINMENT

	NUMBER OF BREAKS			
	CIRCUMFERENTIAL		LONGITUDINAL	
SYSTEM	UNIT 1	UNIT 2	UNIT 1	UNIT 2
Main Steam	8	8	--	--
Feedwater	10	10	3	3
Condensate Booster	7	4	--	--
RCIC	1	1	1	1

TABLE C-2

(Sheet 1 of 2)

RESULTS OF DYNAMIC ANALYSIS FOR POSTULATED PIPE RUPTURE

OUTSIDE CONTAINMENT

SUB-SYSTEM	LINE NO.	RESTRAINT NO.	BREAK LOCATION	BLOWDOWN LOAD (KIPS)	BREAK TYPE
FW-03	1FW02HA-24 2FW02HA-24	RFW-2	C-444	254.4	C
FW-03	1FW02BA-24 2FW02BA-24	RFW-10	C-430	240.3	C
FW-03	1FW02BA-24 2FW02BA-24	RFW-11	C-429	167.4	C
FW-04	1FW02HB-24 2FW02HB-24	RFW-1	C-435	254.4	C
FW-04	1FW02BB-24 2FW02BB-24	RFW-3	C-448	240.3	C
FW-05	1FW02AB-30 2FW02AB-30	RFW-5	C-396	562.6/480	C/L
FW-05	1FW02AA-30 2FW02AA-30	RFW-6	C-407	587.2/480	C/L
FW-05	1FW-34.5 2FW-34.5	RFW-7	C-412	667.6	C
FW-05	1FW02A-30 2FW02A-30	RFW-8	C-417	415.6	C
FW-05	1FW02BB-24 2FW02BB-24	RFW-9	C-421	216/385	C/L
MS-05	1MS01BA-26 2MS01BD-26	RMS-3	C-17	334.5	C
MS-05	1MS01BA-26 2MS01BD-26	RMS-7	C-19	334.5	C
MS-06	1MS01BB-26 2MS01BD-26	RMS-4	C-25	334.5	C
MS-06	1MS01BB-26 2MS01BC-26	RMS-8	C-27	334.5	C

TABLE C-2

(Sheet 2 of 2)

RESULTS OF DYNAMIC ANALYSIS FOR POSTULATED PIPE RUPTURE

OUTSIDE CONTAINMENT

SUB-SYSTEM	LINE NO.	RESTRAINT NO.	BREAK LOCATION	BLOWDOWN LOAD (KIPS)	BREAK TYPE
MS-07	1MS01BC-26 2MS01BB-26	RMS-1	C-3	334.5	C
MS-07	1MS01BC-26 2MS01BB-26	RMS-5	C-5	334.3	C
MS-08	1MS01BD-26 2MS01BA-26	RMS-2	C-9	334.34	C
MS-08	1MS01BD-26 2MS01BA-26	RMS-6	C-11	334.34	C
CB-01	1CB06CA-30	R-134	C-134	119.89	C
CB-01	1CB06CB-30 2CB06CA-30	R-135	C-135	119.89 129.1	C
CB-01	1CB06CB-30	R-136	C-136	337.89	C
CB-01	1CB06CB-30	R-137	C-137	84.77	C
CB-01	1CB06CB-30 2CB06CB-30	R-138	C-138	334.5 286.7	C
CB-01	1CB06CA-30	R-139	C-139	353.61	C
CB-03	1CB06B-36 2CB06B-36	RCB-2	C-658	545	C
CB-03	2CB06D-24	RCB-140	C-696	262	C
RI-01	1RI01C-4 2RI01C-4	R-141	C-77B	9.748/9.748	C/L

[redacted]

LASALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE C-1

WHIP RESTRAINTS ON THE STEAM TUNNEL AREA

(SHEET 1 OF 5)

REV. 15, APRIL 2004

[redacted]

LASALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE C-1

WHIP RESTRAINTS ON THE STEAM TUNNEL AREA

(SHEET 2 OF 5)

REV. 15, APRIL 2004

M-466

SEE M-465

[redacted]

LASALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE C-1

WHIP RESTRAINTS ON THE STEAM TUNNEL AREA

(SHEET 3 OF 5)

M-403

[redacted]

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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE C-1

WHIP RESTRAINTS ON THE STEAM TUNNEL AREA

(SHEET 4 OF 5)

M-572

[redacted]

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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE C-1

WHIP RESTRAINTS ON THE STEAM TUNNEL AREA

(SHEET 5 OF 5)

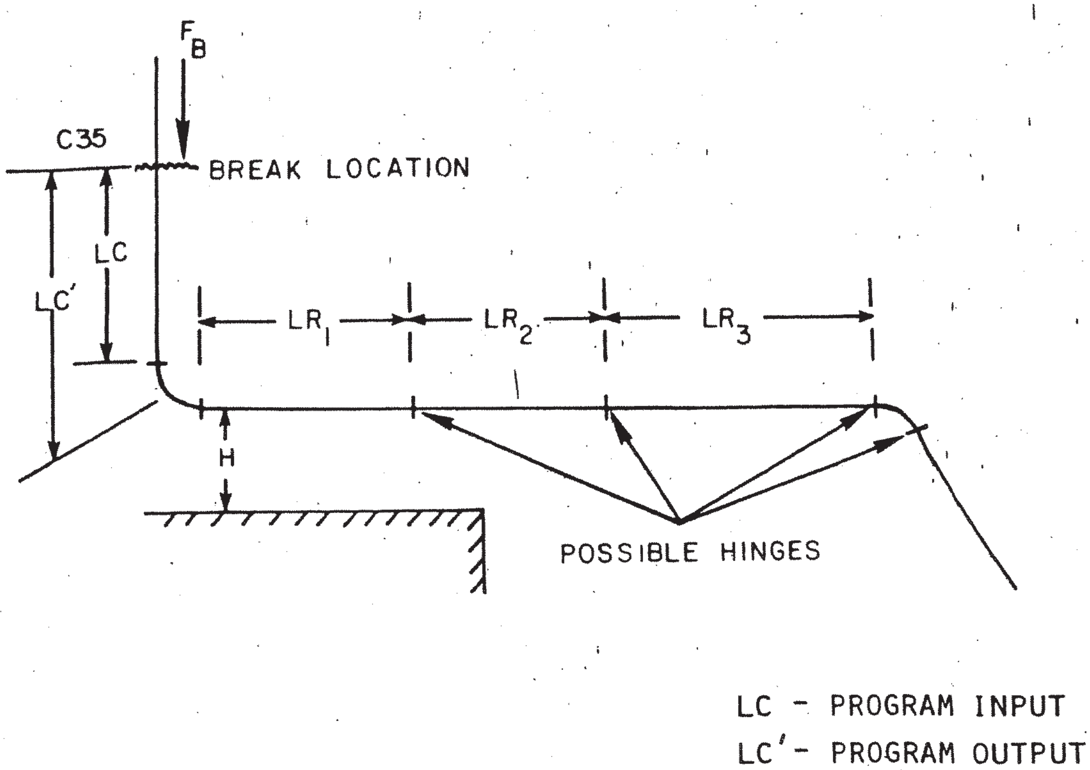

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FIGURE C1-1

PHYSICAL LAYOUT FOR
CIRCUMFERENTIAL BREAK

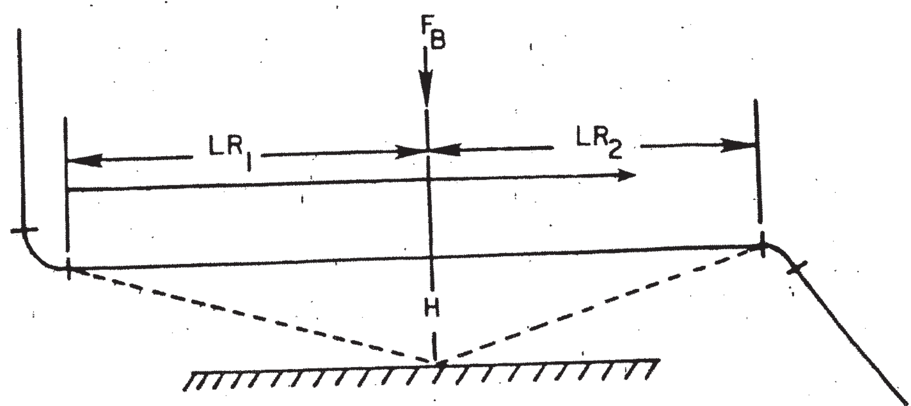

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FIGURE C1-2

PHYSICAL LAYOUT FOR
LONGITUDINAL BREAK

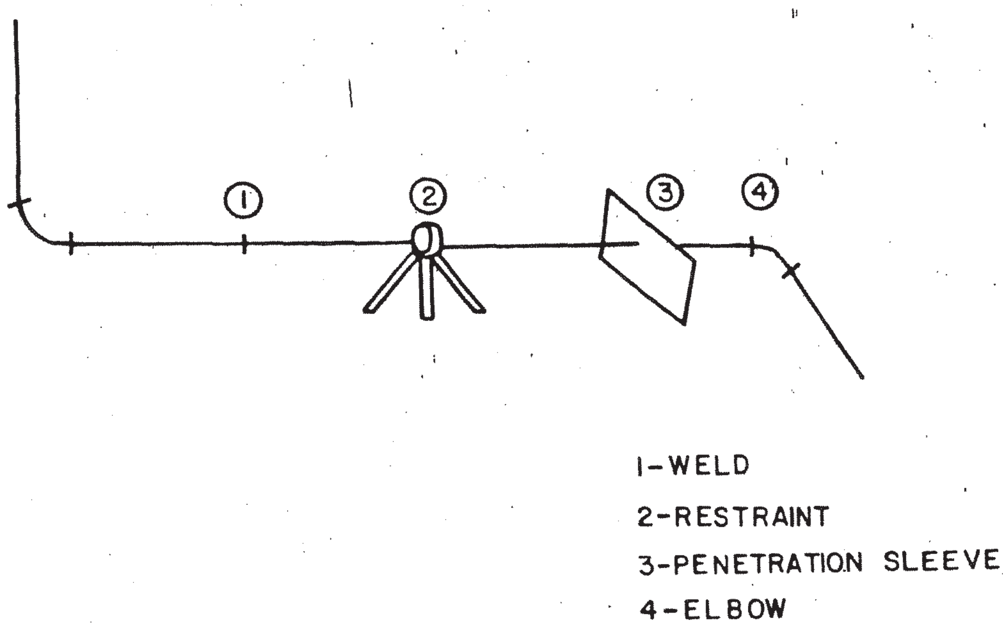

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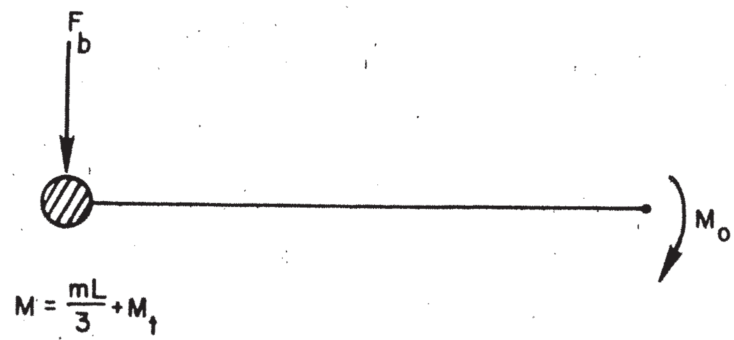
FIGURE C1-3

POSSIBLE HINGE FORMATIONS



NOTE: 1. DISTRIBUTED MASS ALONG PIPE HAS BEEN LUMPED IN ONE END.

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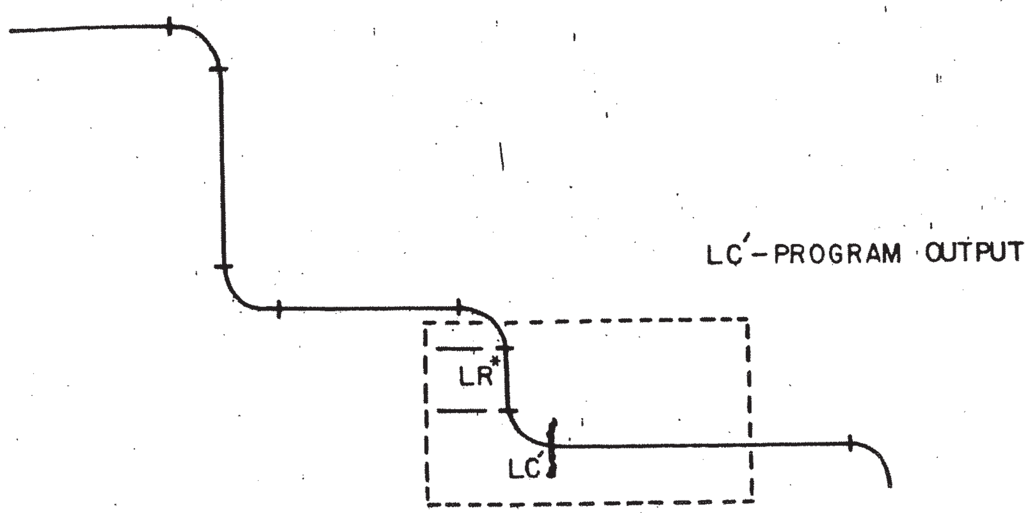
FIGURE C1-4 MODEL FOR PROBLEM FORMULATION



NOTE: 1. PROGRAM ONLY KNOWS INFORMATION FOR CONFIGURATION ENCLOSED BY DOTTED LINE.

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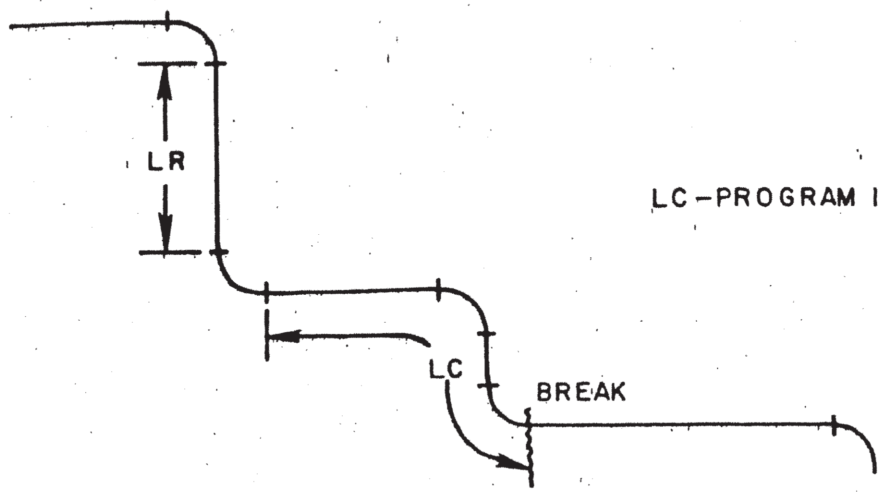
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE C1-5

CIRCUMFERENTIAL BREAK MODEL – ORIGINAL FORMULATION



NOTE: 1. APPLIES WHEN $LR^* < L_S$.

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FIGURE C1-6

CIRCUMFERENTIAL BREAK MODEL – REFORMULATION

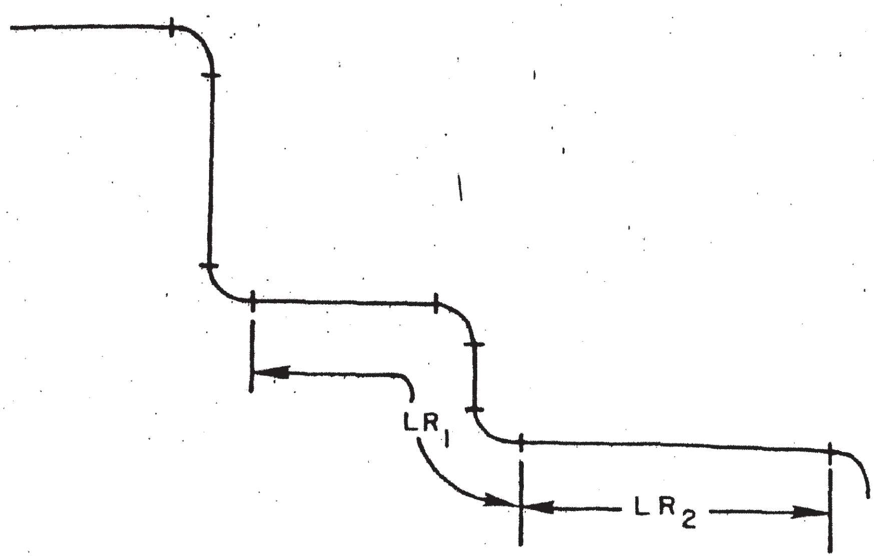

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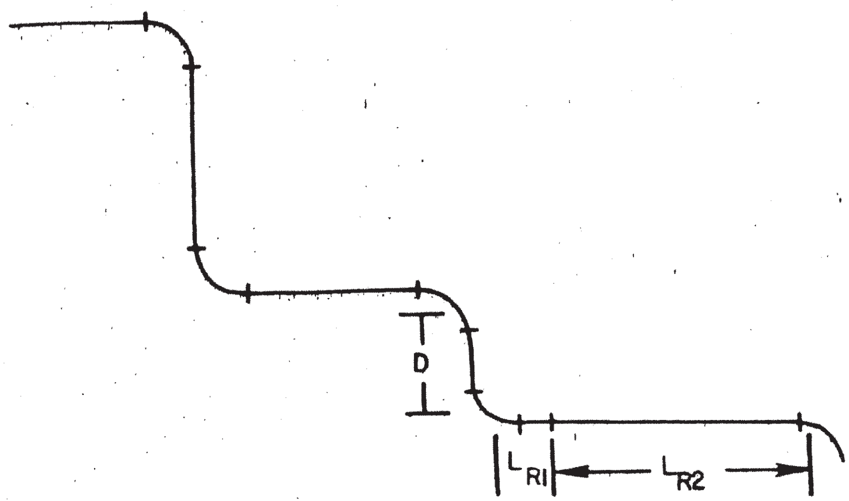
FIGURE C1-7

CIRCUMFERENTIAL BREAK MODEL – FORMULATE L - BREAK



NOTE: 1. L BREAK IS TOO CLOSE TO ELBOW.

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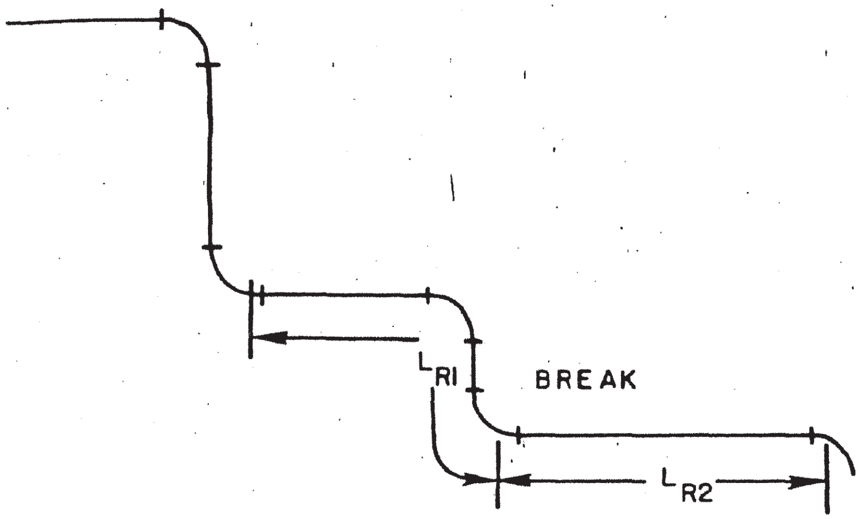
FIGURE C1-8

LONGITUDINAL BREAK MODEL – $L_{r1}<L_s$



NOTE: 1. APPLIES WHEN $D < L_S$.

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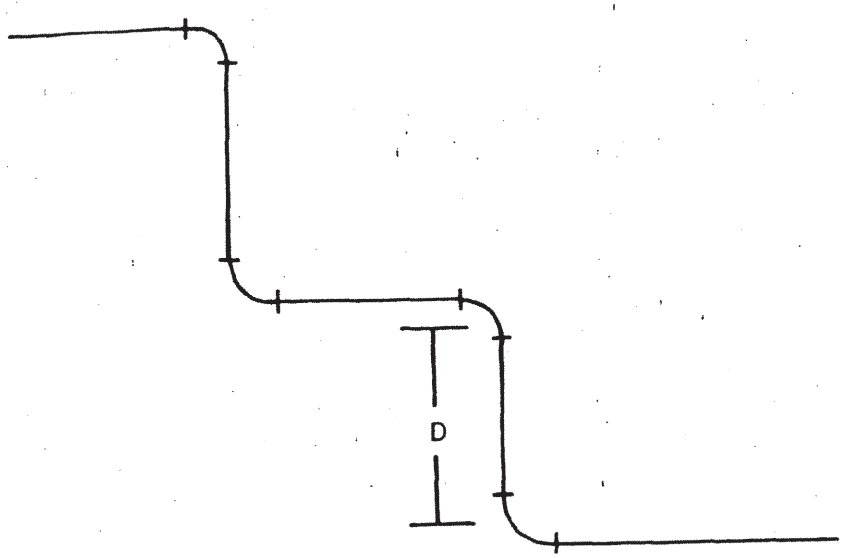
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE C1-9

LONGITUDINAL BREAK MODEL – REFORMULATION



NOTE: 1. APPLIES WHEN D>L_S.

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FIGURE C1-10

LONGITUDINAL BREAK MODEL – FORMULATE C – BREAK

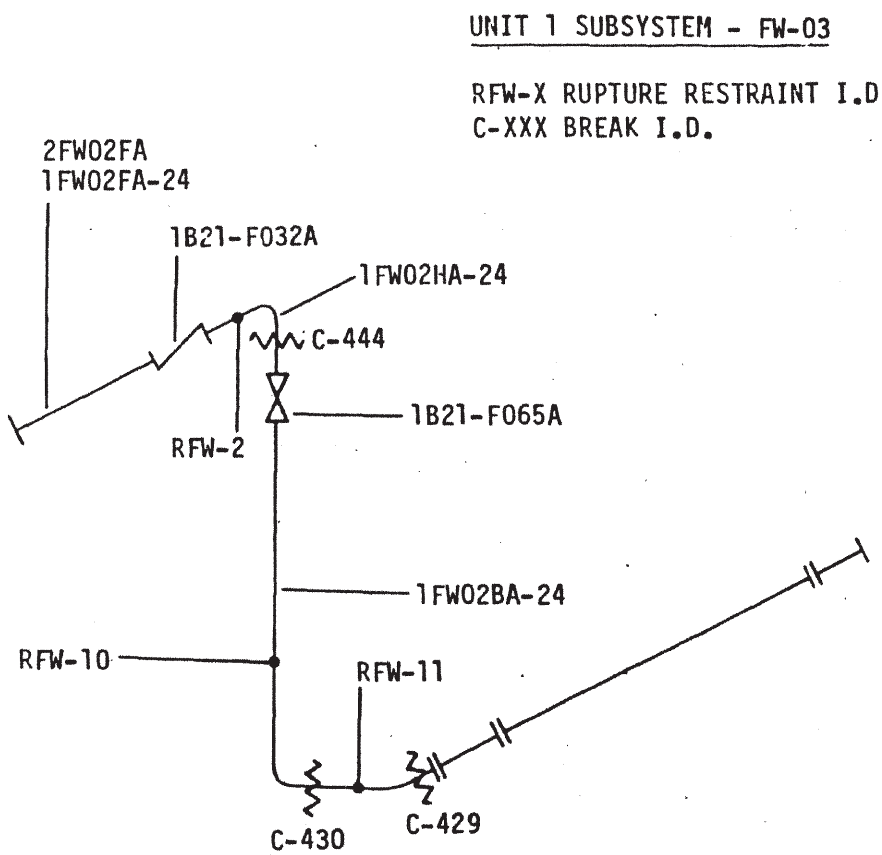

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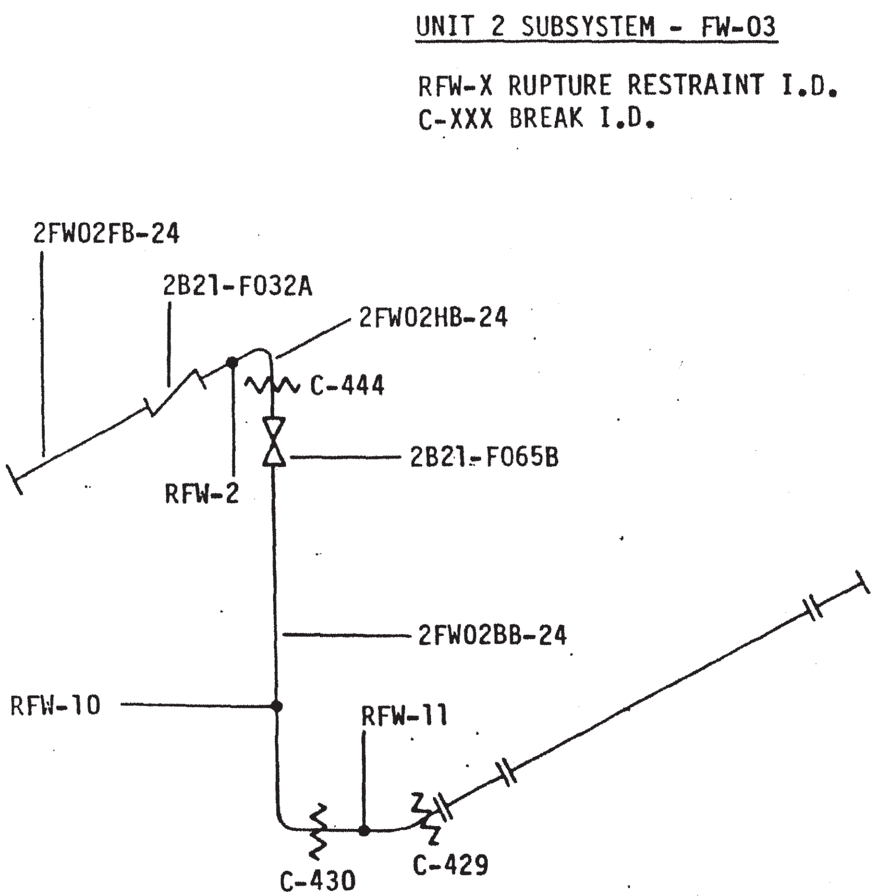
FIGURE C-2

FEEDWATER SYSTEM – POSTULATED BREAKS AND RESTRAINT LOCATIONS

(SHEET 1 OF 6)



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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE C-2

FEEDWATER SYSTEM – POSTULATED BREAKS AND RESTRAINT LOCATIONS

(SHEET 2 OF 6)

UNIT 1 SUBSYSTEM - FW-04

RFW-XX RUPTURE RESTRAINT I.D.
C-XXX BREAK I.D.

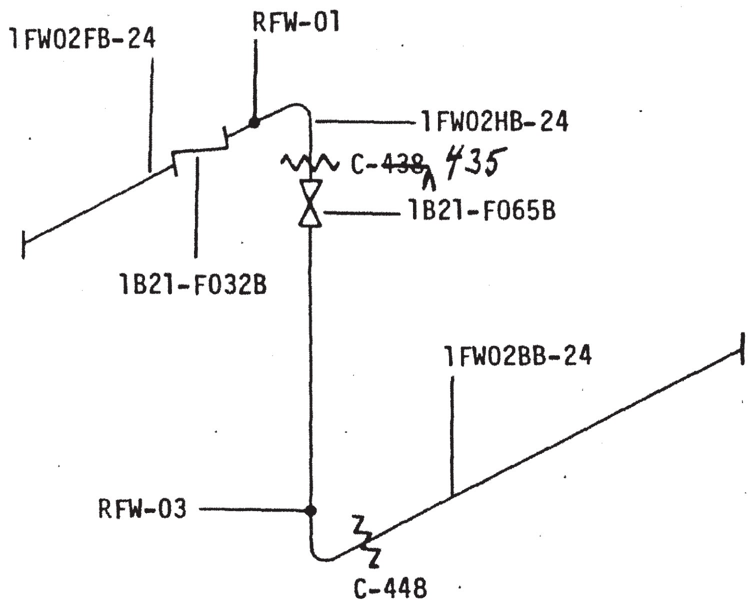

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FIGURE C-2 FEEDWATER SYSTEM – POSTULATED BREAKS AND RESTRAINT LOCATIONS (SHEET 3 OF 6)

UNIT 2 SUBSYSTEM - FW-04

RFW-XX RUPTURE RESTRAINT I.D.
C-XXX BREAK I.D.

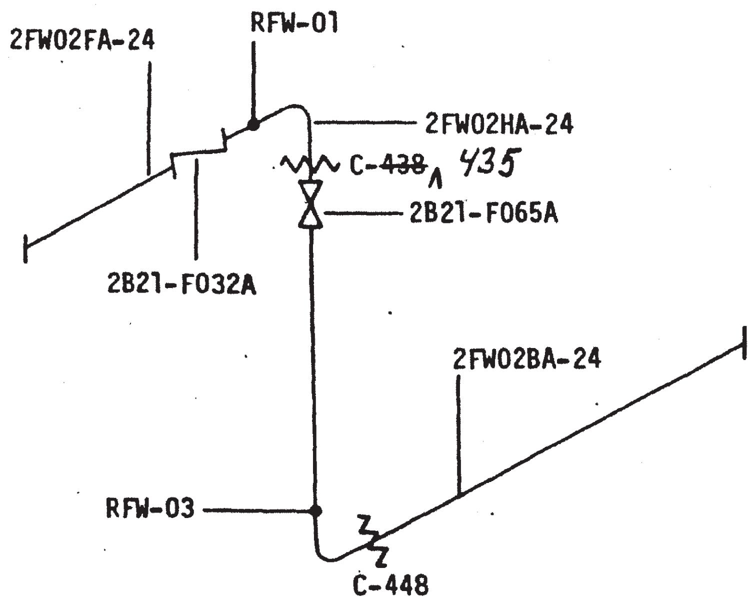

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FIGURE C-2

FEEDWATER SYSTEM – POSTULATED BREAKS AND RESTRAINT LOCATIONS

(SHEET 4 OF 6)

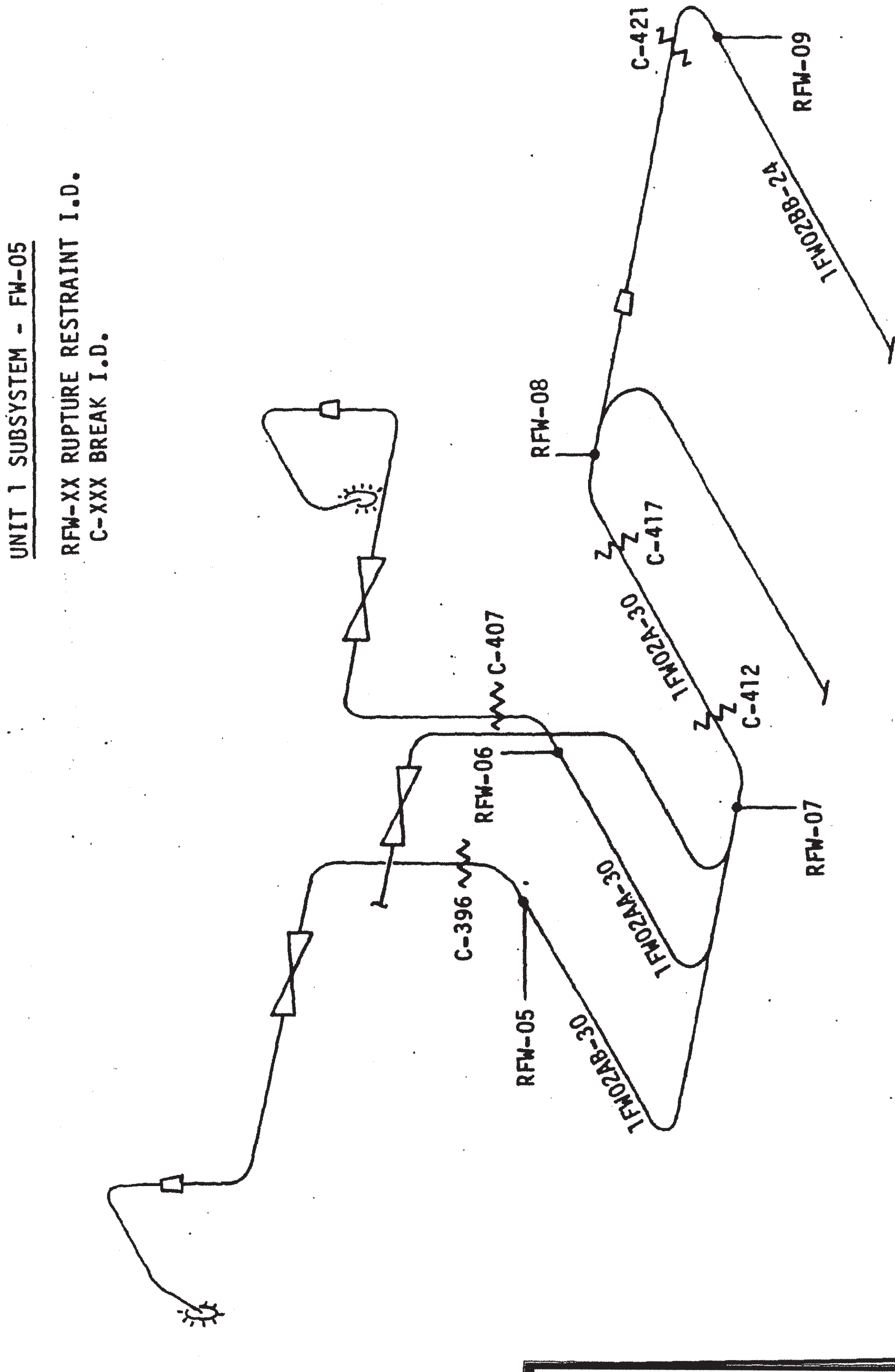

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FIGURE C-2

FEEDWATER SYSTEM – POSTULATED BREAKS AND RESTRAINT LOCATIONS

(SHEET 5 OF 6)

REV 15 APRIL 2004

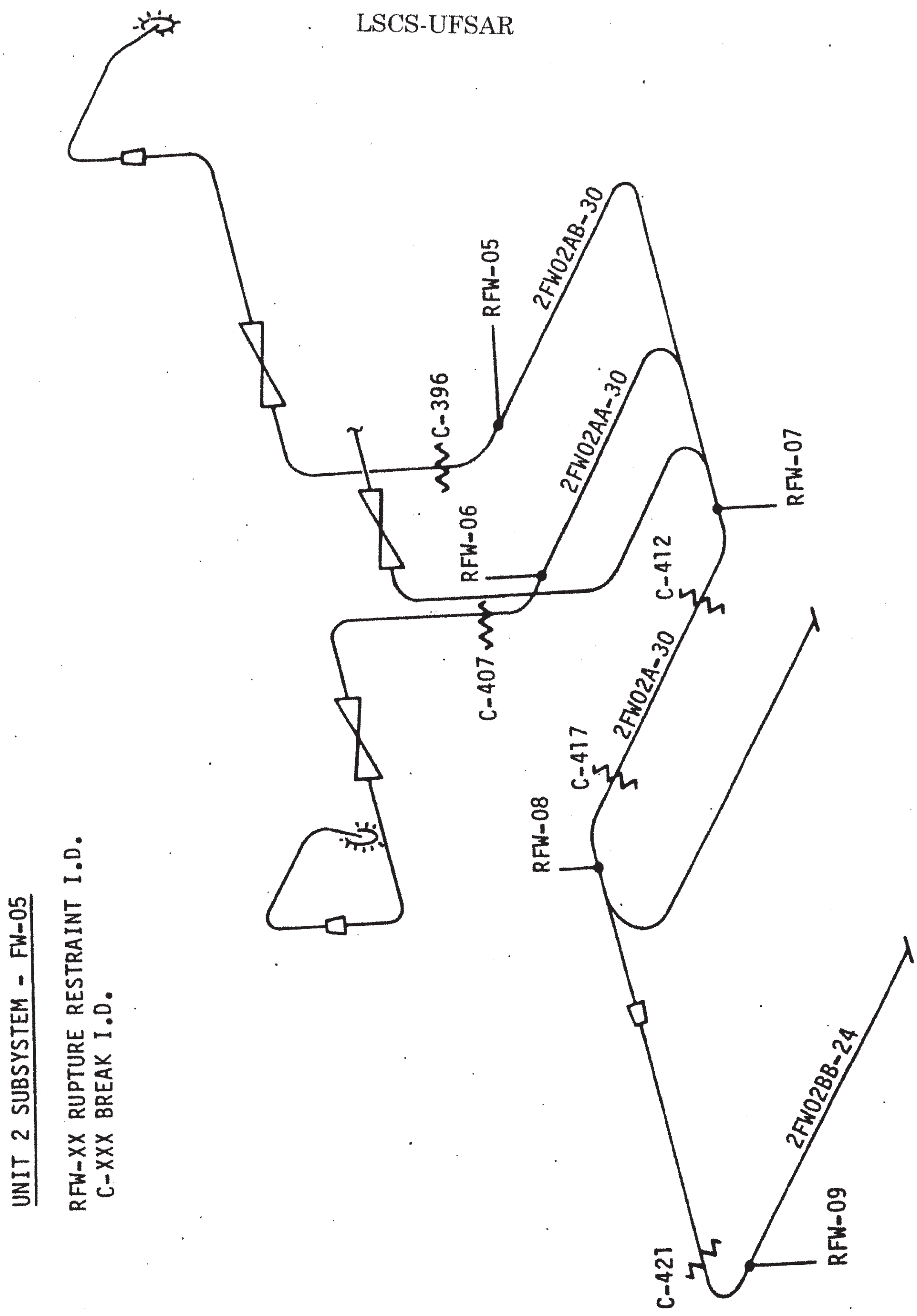

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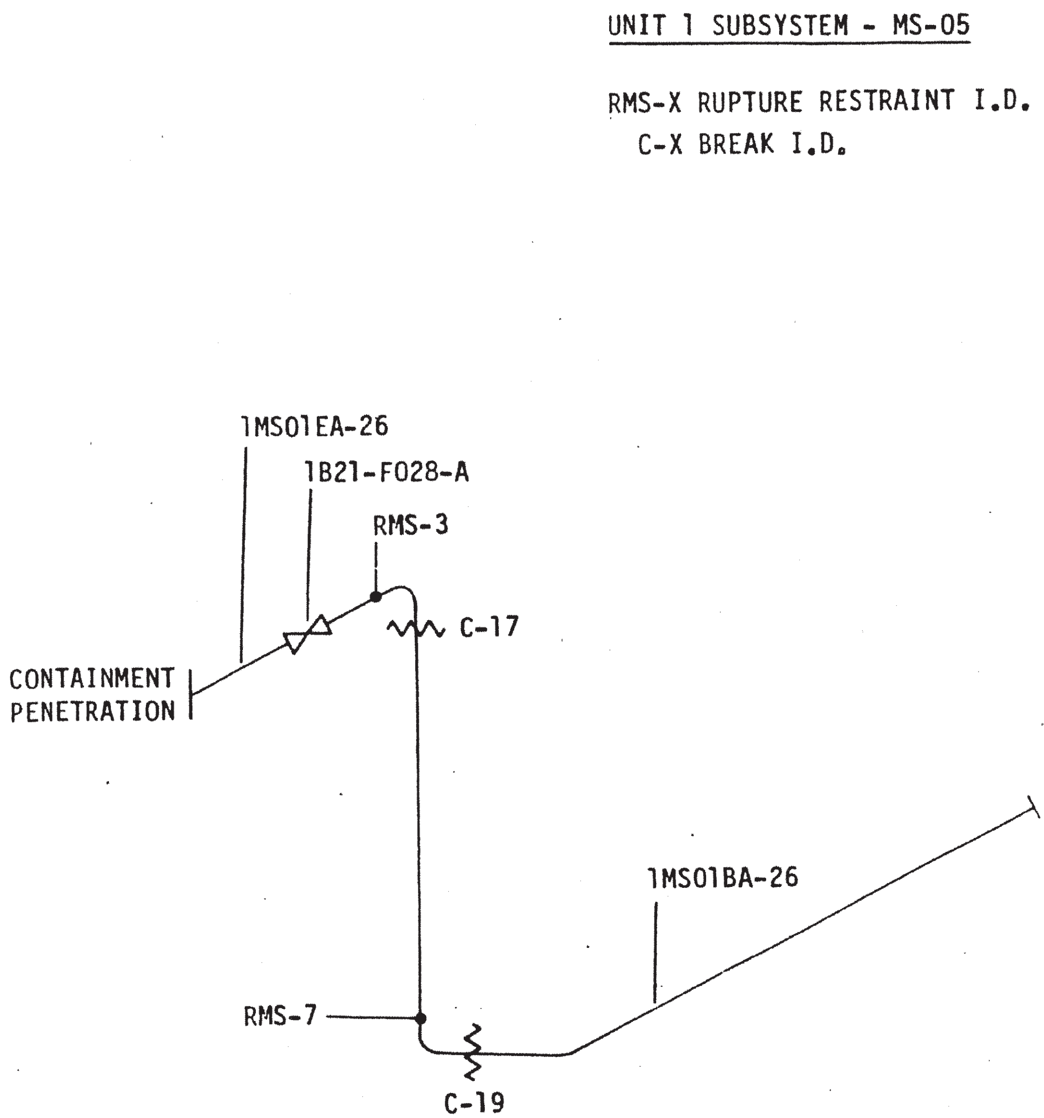
FIGURE C-2 FEEDWATER SYSTEM – POSTULATED BREAKS AND RESTRAINT LOCATIONS (SHEET 6 OF 6)



LASALLE COUNTY STATION
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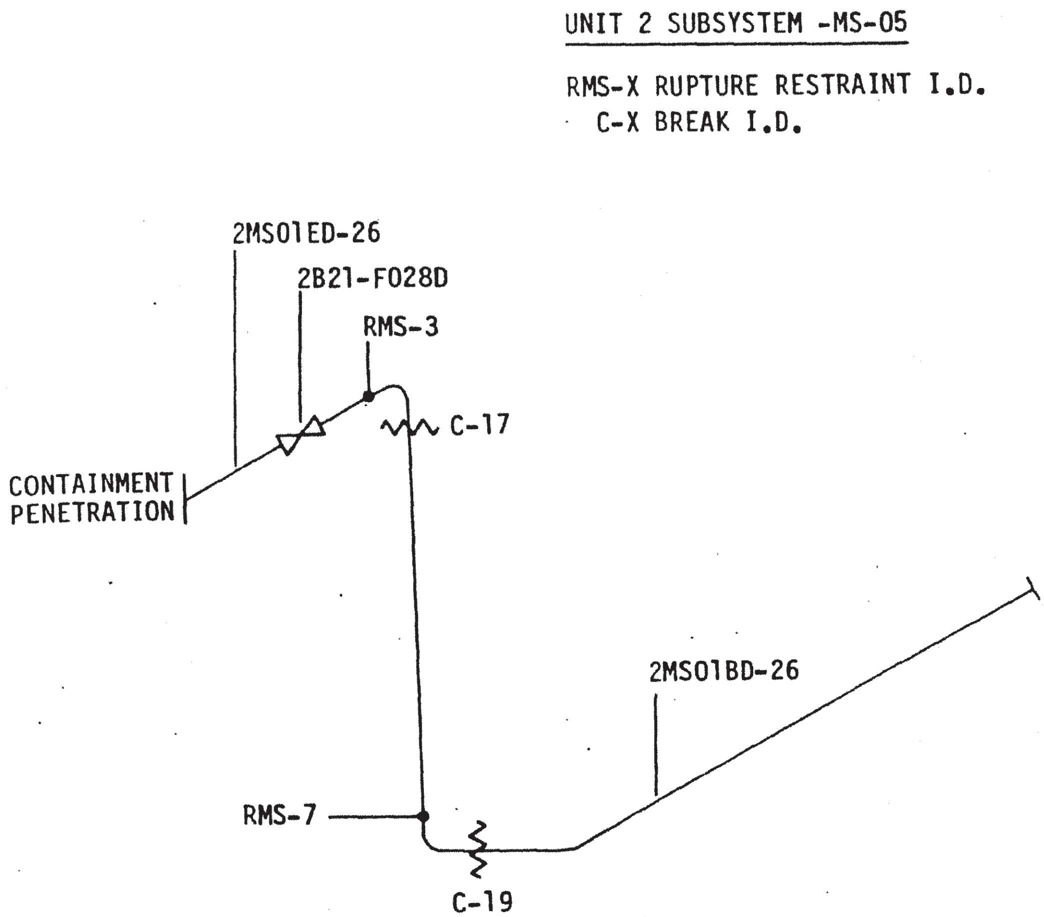
FIGURE C-3

MAIN STEAM – POSTULATED BREAKS AND RESTRAINT LOCATIONS

(SHEET 1 OF 8)



LASALLE COUNTY STATION
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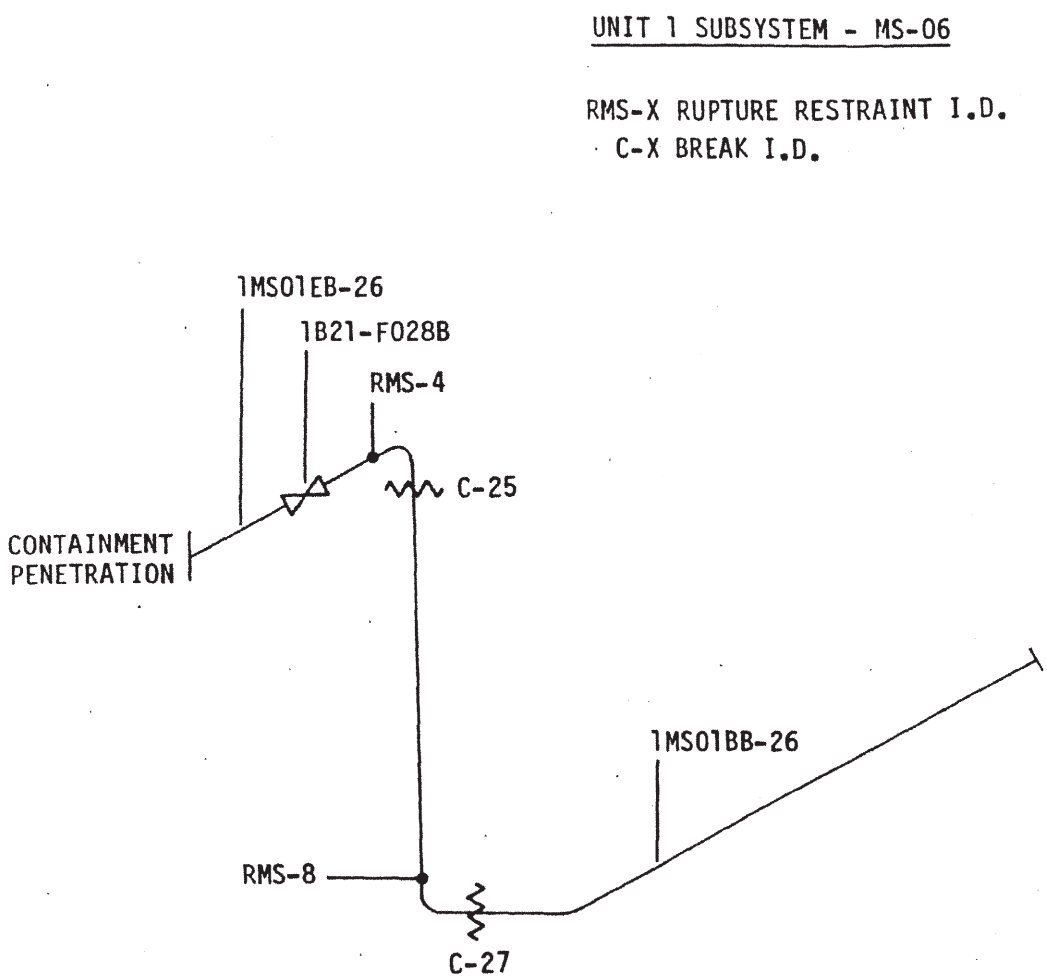
FIGURE C-3

MAIN STEAM – POSTULATED BREAKS AND RESTRAINT LOCATIONS

(SHEET 2 OF 8)



LASALLE COUNTY STATION
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FIGURE C-3

MAIN STEAM – POSTULATED BREAKS AND RESTRAINT LOCATIONS

(SHEET 3 OF 8)

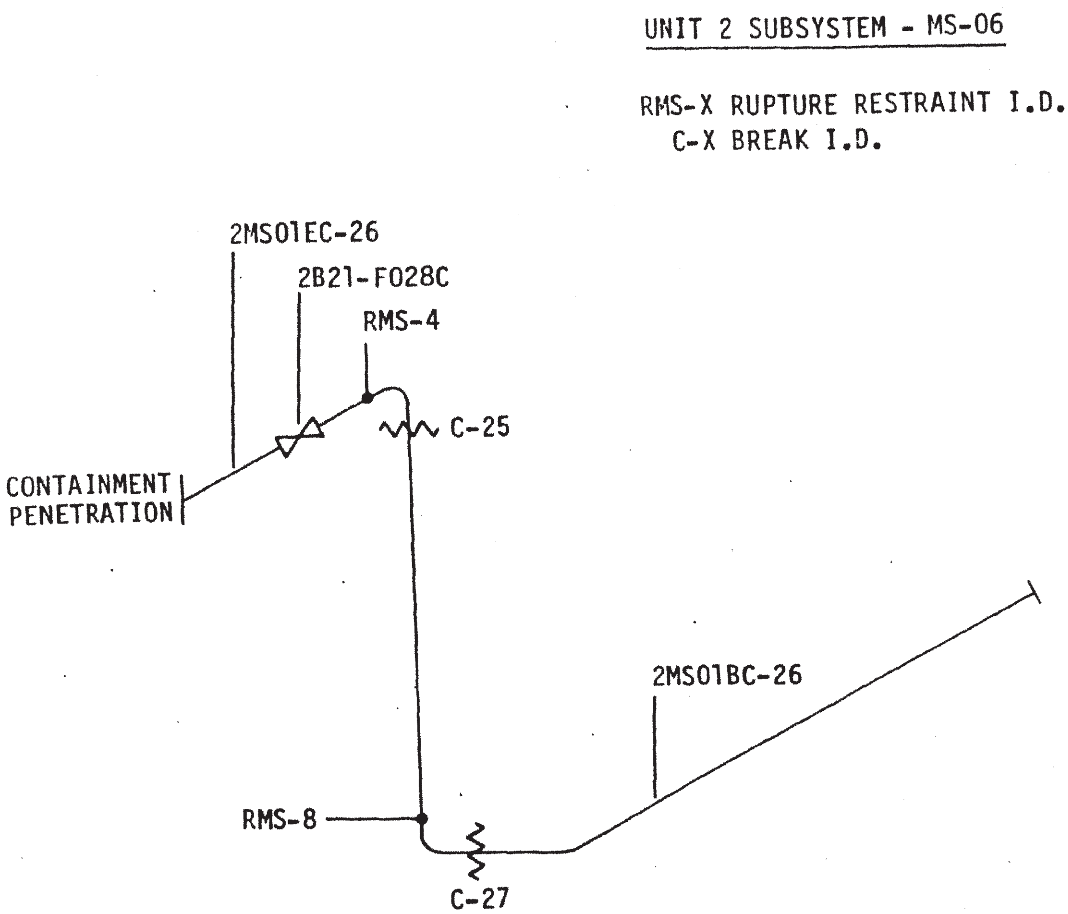

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FIGURE C-3

MAIN STEAM – POSTULATED BREAKS AND RESTRAINT LOCATIONS

(SHEET 4 OF 8)

UNIT 1 SUBSYSTEM - MS-07

RMS-X RUPTURE RESTRAINT I.D.
C-X BREAK I.D.

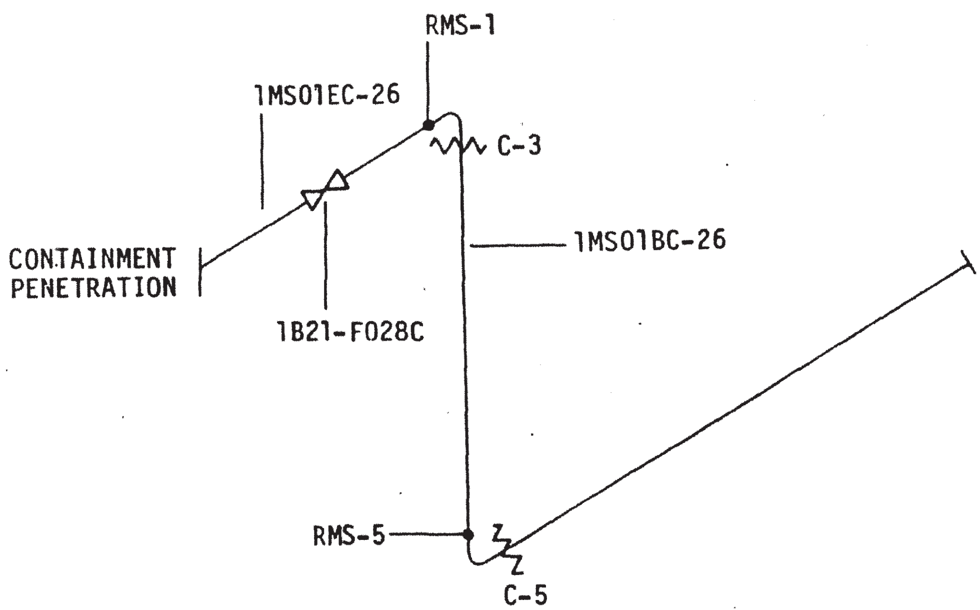

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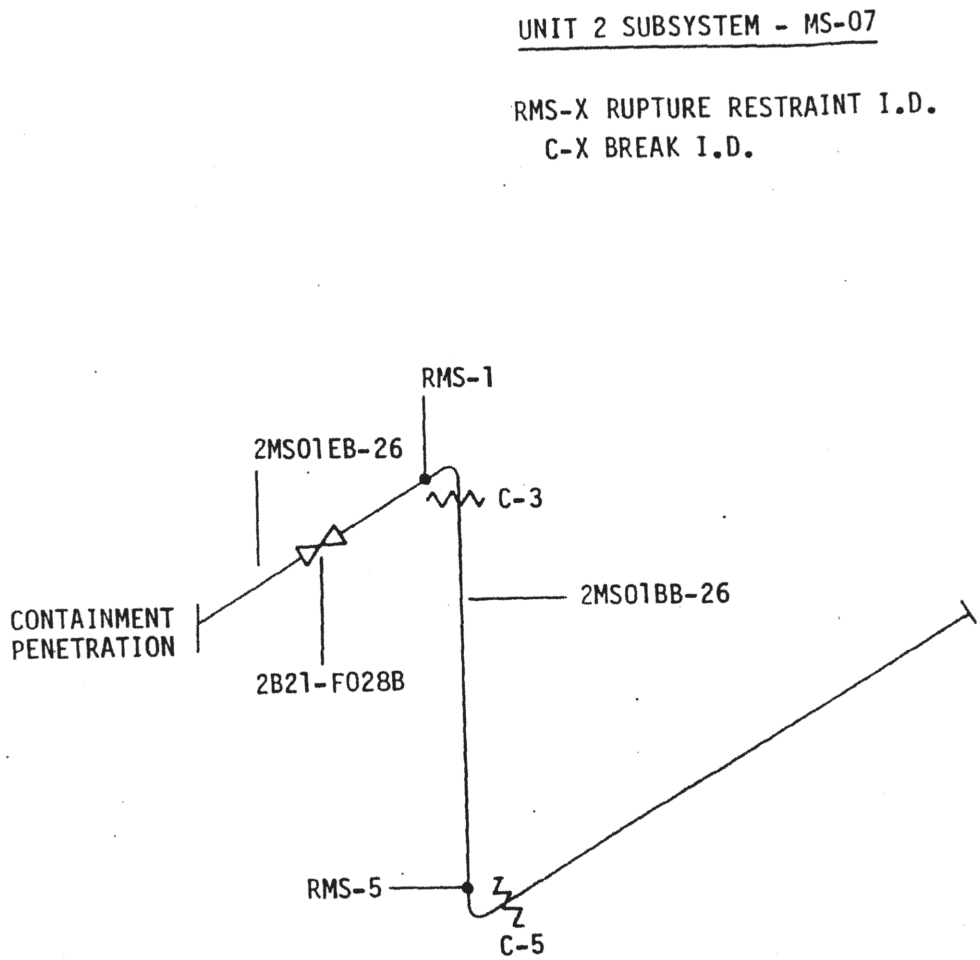
FIGURE C-3 MAIN STEAM – POSTULATED BREAKS AND RESTRAINT LOCATIONS (SHEET 5 OF 8)



LASALLE COUNTY STATION
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FIGURE C-3

MAIN STEAM – POSTULATED BREAKS AND RESTRAINT LOCATIONS

(SHEET 6 OF 8)

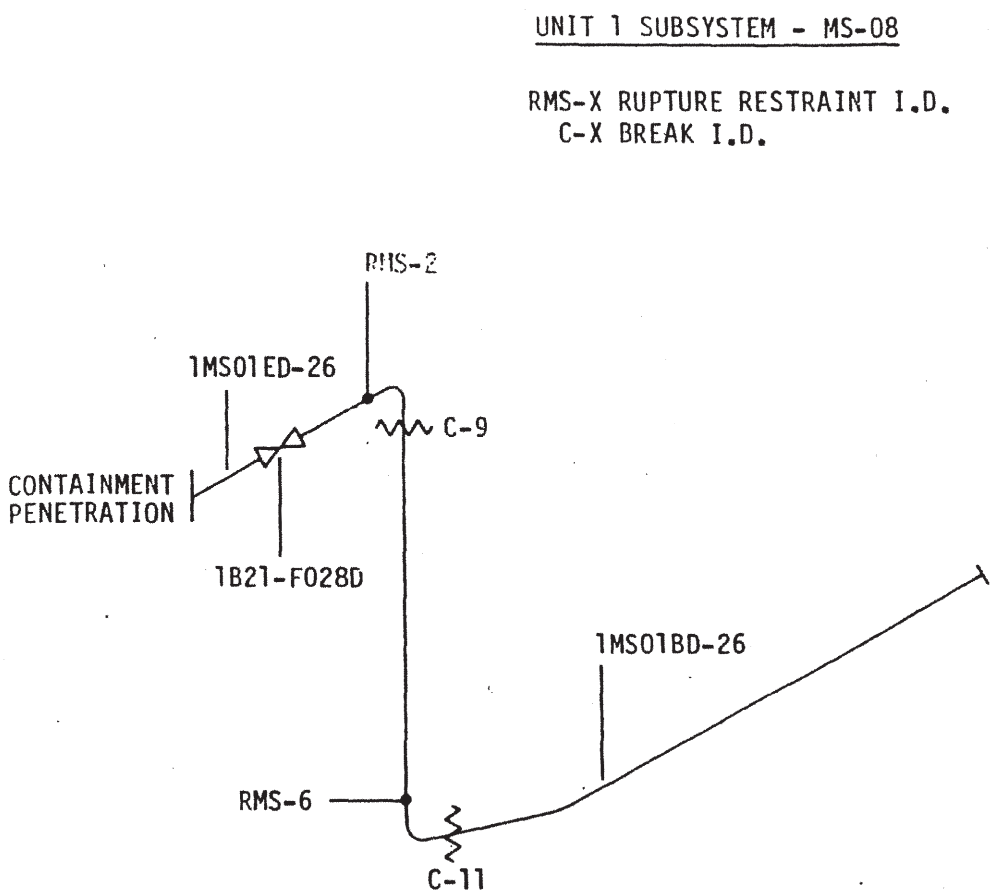

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FIGURE C-3

MAIN STEAM – POSTULATED BREAKS AND RESTRAINT LOCATIONS

(SHEET 7 OF 8)

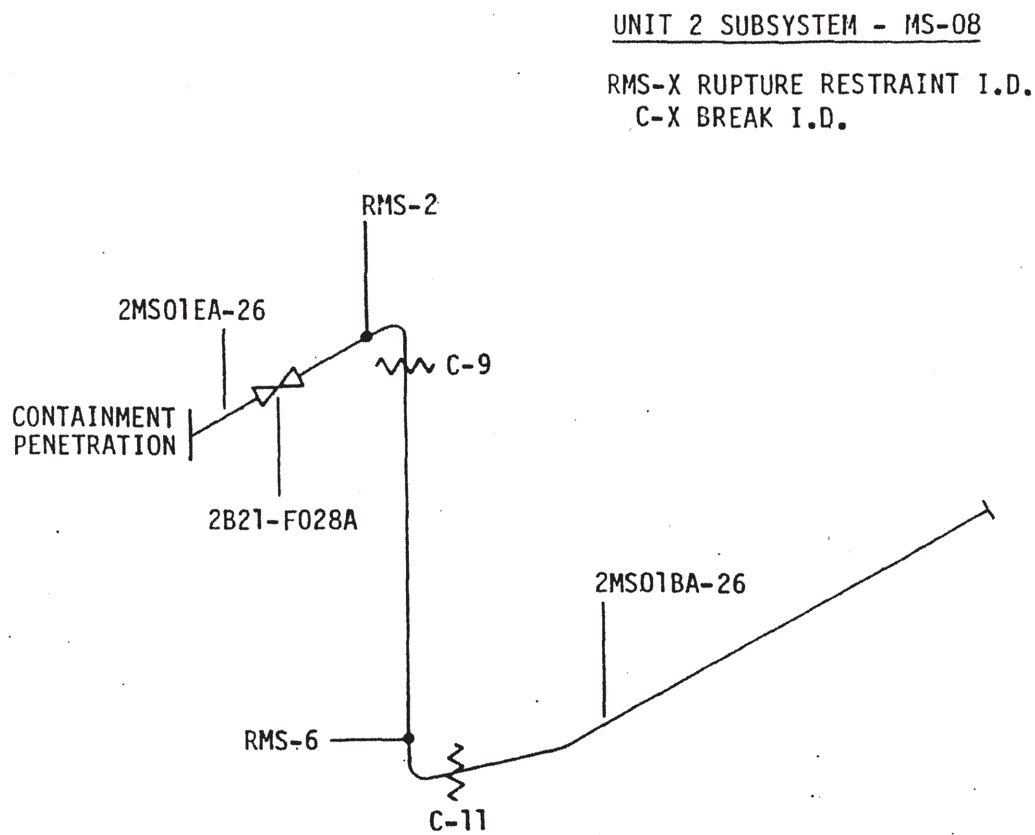

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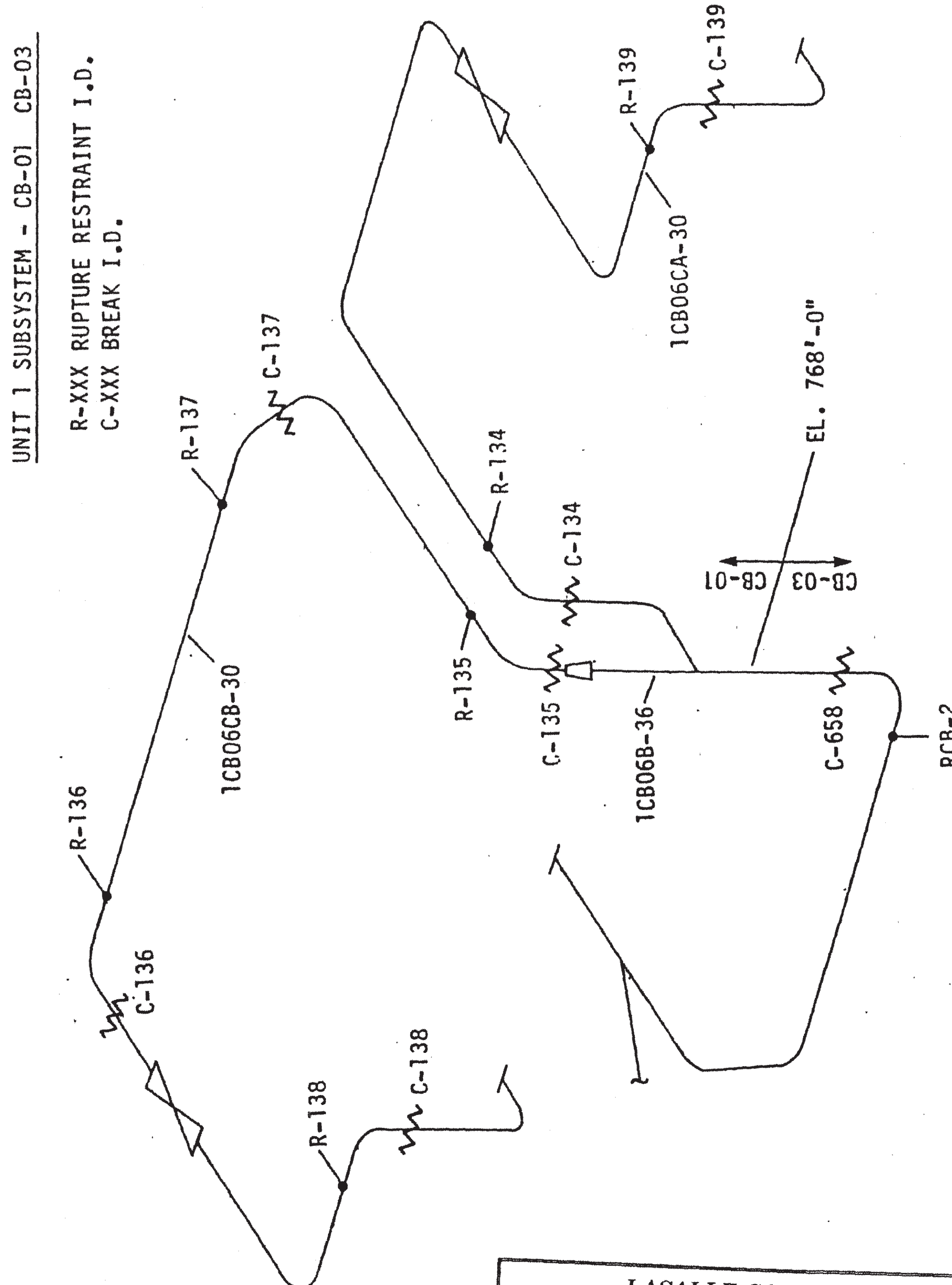
FIGURE C-3

MAIN STEAM – POSTULATED BREAKS AND RESTRAINT LOCATIONS

(SHEET 8 OF 8)



LASALLE COUNTY STATION
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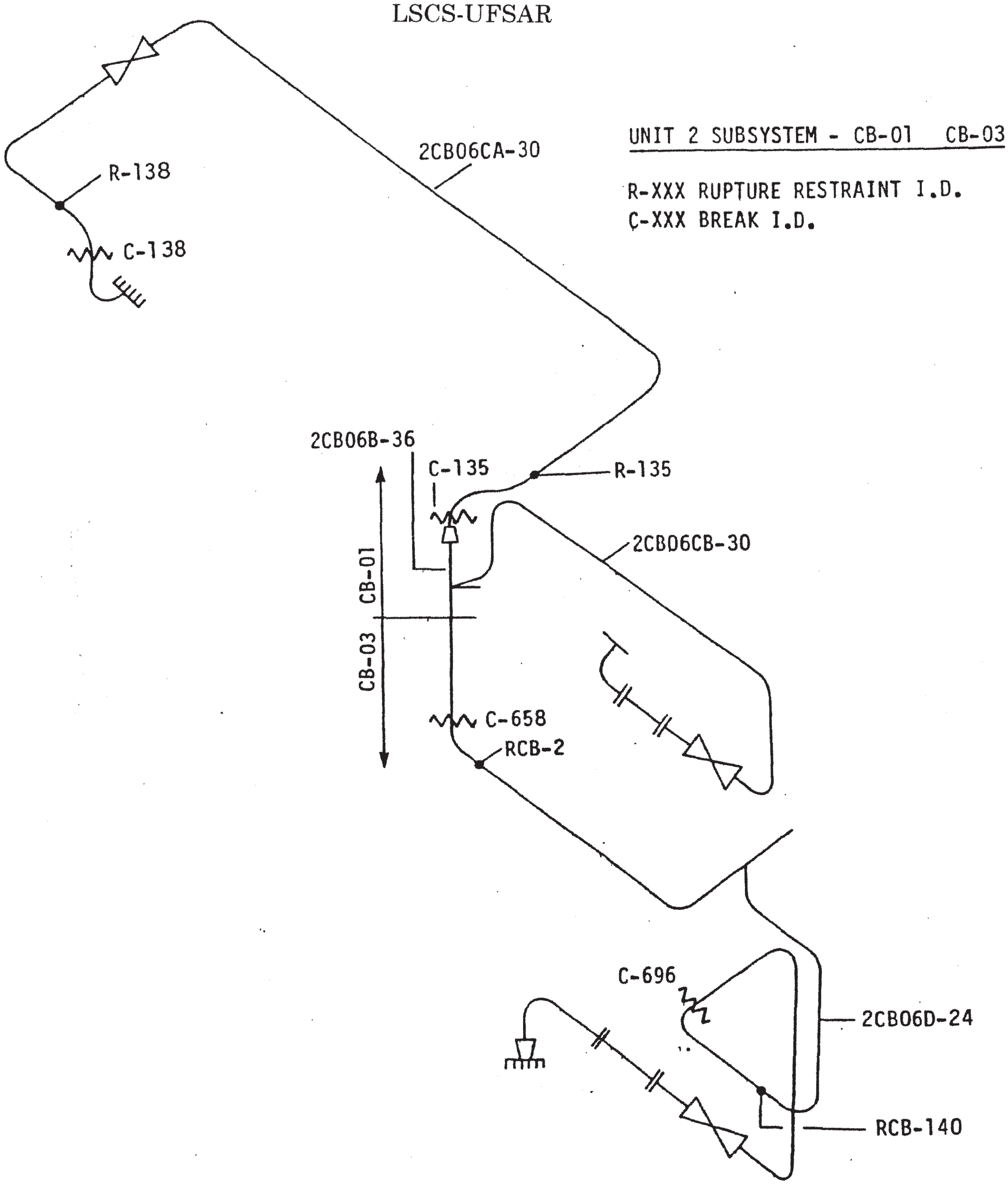
FIGURE C-4

CONDENSATE BOOSTER – POSTULATED BREAKS AND RESTRAINT LOCATIONS

(SHEET 1 OF 2)



LASALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE C-4

CONDENSATE BOOSTER – POSTULATED BREAKS AND RESTRAINT LOCATIONS

(SHEET 2 OF 2)

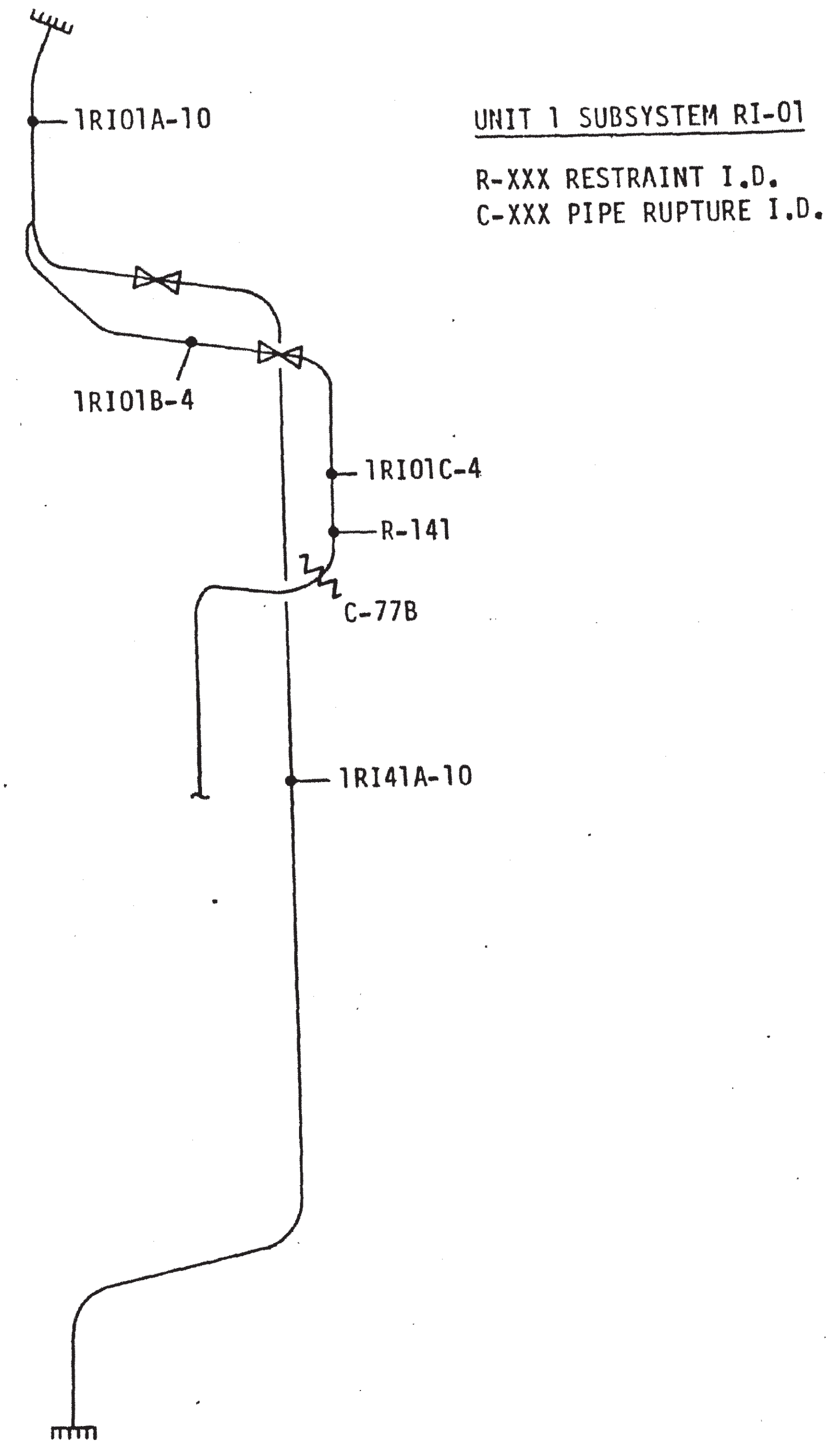


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FIGURE C-5

RCIC – POSTULATED BREAKS AND RESTRAINT LOCATIONS

(SHEET 1 OF 2)

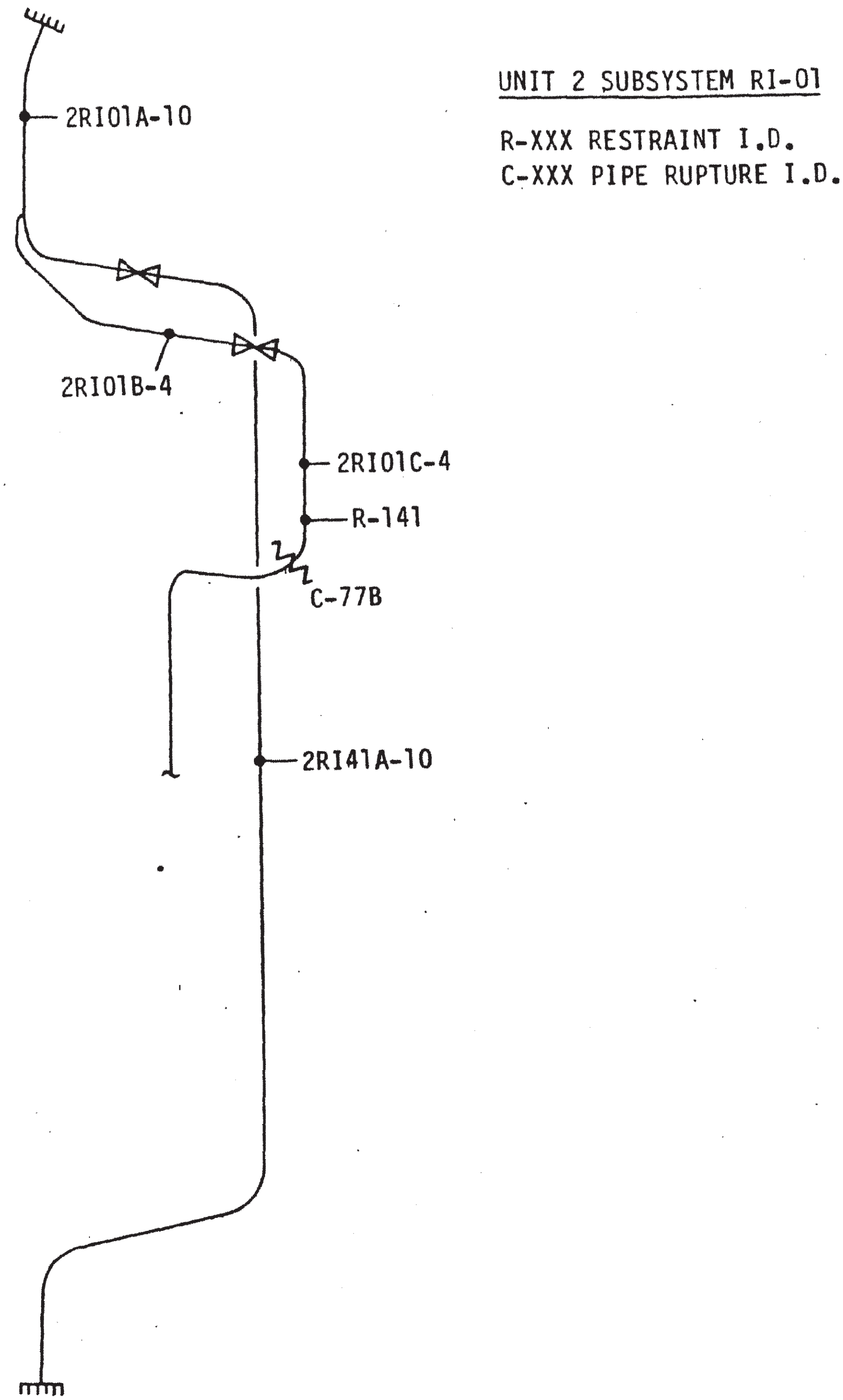

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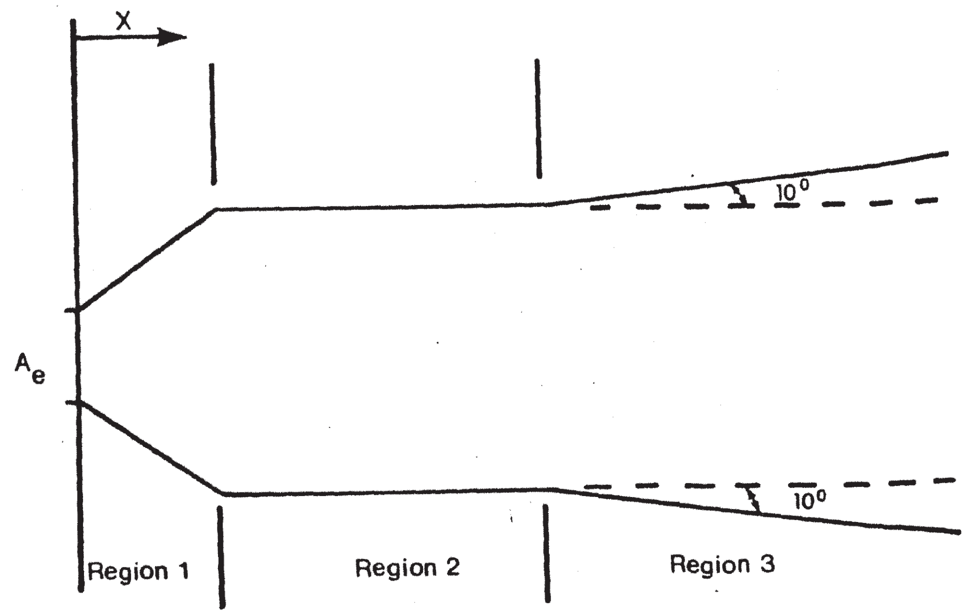
FIGURE C-5 RCIC – POSTULATED BREAKS AND RESTRAINT LOCATIONS (SHEET 2 OF 2)



REGION 1

$0 \leq x \leq 5D_e$

$A_{jet} = A_e + \frac{x}{B}\left(A_{asyp} - A_e\right)$

REGION 2

$5D_e \leq x \leq 1/2\left(\sqrt{\frac{4\ A_{asyp}}{\pi}} - D_e\right)\ \text{COT } 10°$

$A_{jet} = A_{asyp}$

REGION 3

$1/2\left(\sqrt{\frac{4\ A_{asyp}}{\pi}} - D_e\right)\ \text{COT } 10° < x \leq \infty$

$A_{jet} = A_e\left(1 + \frac{2x}{D_e}\ \text{TAN } 10°\right)^2$

where:

A_{asyp} = asymptotic area = $\frac{A_e\ Gv}{g_c\ F}$

A_e = break area

D_e = break diameter

x = axial distance from break

B = $5D_e$

G = mass velocity rate

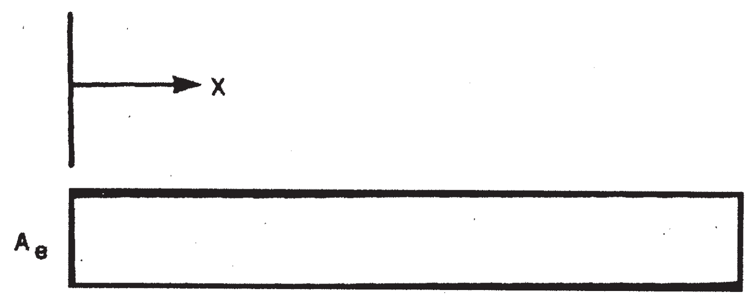
F = thrust

v = momentum specific volume

LASALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE C-6

JET MODEL – FLASHING/EXPANDING FLUID



$0 \leq x \leq \infty$

$A_{jet} = A_e$

where x = axial distance from break, and

A_e = jet area

LASALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE C-7

JET MODEL – NON-FLASHING/NON-EXPANDING FLUID

APPENDIX D

PLANT NUCLEAR SAFETY OPERATIONAL ANALYSIS

Detailed analyses of the "worstcase" sequence of the "Nuclear Safety Operational Analysis" (NSOA) for the LaSalle plant is documented in Chapter 15 of the UFSAR.

APPENDIX E-CONSTRUCTION MATERIAL STANDARDS AND QUALITY CONTROL PROCEDURES

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APPENDIX E

CONSTRUCTION MATERIAL STANDARDS AND QUALITY CONTROL PROCEDURES

E. 1 CONCRETE STANDARDS

E.1.1 General

All concrete work done is in accordance with ACI 318-71, "Building Code Requirements for Reinforced Concrete," and ACI 301-72, "Specifications for Structural Concrete for Buildings," with the exceptions indicated in Subsection E.1.7. The concrete work also complies with all the sections of ACI 349 and ACI 359 which are based on ACI 318-71.

E.1.2 Strength Requirements

The specified minimum compressive strengths are:

Prestressed concrete and the drywell floor	f 'c = 4,500 psi at 91 days
Seismic Category I structures	f 'c = 4,000 psi at 91 days
All others	f 'c = 3,500 psi at 91 days

E.1.3 Material Requirements and Quality Control

E.1.3.1 Cement

Cement is Type I portland cement. The sampling and qualification tests are done in conformance with ASTM C150-72.

Control tests are made for every 1000 tons or once a month on car or truck samples taken at the batch plant. These tests are ASTM Standard Tests: C109-70T, C266-71, C191-71, and C151-71.

E.1.3.2 Aggregates

Aggregates consist of gravel or crushed stones that are clean, hard, and durable. The sampling and qualification tests conform with ASTM C33-71a. Tests that are done from this specification are the following ASTM Standard Tests: D75-71, C136-71, C117-69, C40-66, C87-69, C88-71a, C142-71, C123-69, C131-69, C535-69, C227-71, Cl28-68, and C29-71.

Control tests in accordance with ASTM C136-71 and C40-66 are done daily. ASTM C29-71 and C117-69 are done weekly on samples taken at the batch plant.

E.1.3.3 Fly Ash

Fly ash is used in a portion of the construction. It conforms to ASTM C618-72, Pozzolan Class F, except that in the Table I chemical requirements the maximum percent of sulfur trioxide is 7%. Qualification and control tests are in accordance with the previously mentioned specification. Control tests for approximately 200 tons or about once a month were made on batch plant samples. The control test set consists of the sulfur trioxide, sodium oxide, loss on ignition, and fineness tests of A8TM C618-72.

E.1.3.4 Admixtures

Air entraining admixtures used in all construction conform to ASTM C260-69. Retarding and water-reducing admixtures when used conform to ASTM C494-71.

E.1.3.5 Water

An initial qualification test in accordance with AASHO T26-70 is made on samples taken from the water source to be used in the concrete mix. A monthly control test in accordance with ASTM D512-67 is taken to determine the chloride ion content in the mix.

E.1.3.6 Concrete Testing

During construction, concrete is sampled and tested to ascertain conformance to ACI 318-71 and ACI 301-72. Using the previously specified ingredients the following physical properties are determined for each sample:

a. Temperature of concrete

b. Temperature of air

c. Slump of concrete (ASTM C143-71)

d. Air content (ASTM C231-72T)

The physical properties stated above are determined for each control test performed. Samples are secured in accordance with ASTM C172-71 at the forms when concrete is pumped or at the mixer truck when a conveyor or bucket is used. The samples are taken per each 50 yd^3 of concrete, or for each pour, or one out of five consecutive trucks.

e. Compressive strength of concrete

Six cylinders are filled with fresh concrete and molded into test cylinders. The set of six test cylinders is taken with the following frequencies:

Total Cubic Yards of Concrete in Each Continuous Placement of Concrete	Frequency
100 yd^3 or less	each day's placement
100 to 500 yd^3	each 100 yd^3
500 to 2000 yd^3	each 200 yd^3
over 2000 yd^3	each 300 yd^3

The cylinders are molded and cured in accordance with ASTM C31-69. Two cylinders are tested at 7 days, two cylinders at 28 days, and two cylinders at 91 days in accordance with ASTM C39-71. Evaluation of compression test results and acceptance of concrete is done in accordance with ACI 214-65 and with Subsection 4.3.3 of ACI 318-71.

f. Tensile splitting strength

For every ten samples taken of BA45 concrete, two additional cylinders are molded. These are tested at 91 days for the tensile splitting strength in accordance with ASTM C496-71.

E.1.4 Concrete Mix,Design

The concrete mix is designed in accordance with ACI 613-64 using the previously specified concrete materials. The trial mixes are tested in accordance with ASTM standards listed as follows:

	Test	ASTM Designation
a.	Making and curing of the test cylinders in the laboratory	C192-69
b.	Air content	C231-72T or C173-71
c.	Slump	C143-71
d.	Compressive strength	C39-71

Six cylinders are cast from each design mix. Compressive strength tests are made at 7, 28, and 91 days,. Two cylinders are used for each test.

Test cylinders are cast from the mix proportions selected for the containment and the following concrete properties determined:

a. uniaxial creep,

b. modulus of elasticity and Poisson's ratio,

c. plastic shrinkage (both autogenous and paste shrinkage),

d. thermal diffusivity,

e. volume changes (expansion and contraction), and

f. coefficient of thermal expansion.

E.1.5 Batching, mixing, Delivery, and Placement

Batching, mixing, and delivery equipment, including their operation, conforms to the requirements of' ASTM C94-72, Chapters 7, 8, and 9. The extent to which placement complies with the requirements of Regulatory Guide 1.55 is discussed in Appendix B.

E.1.6 Witness and Inspections

A testing agency is retained to make inspections of the manufacturer's, suppliers, and contractor's plants or installations to investigate their testing facilities and procedures.

The testing agency inspects the batch plant and stationary and truck mixers or agitators prior to production operation to verify conformance with ASTM C94-72, Sections 7, 8, and 9 of ACI 304-73 (Chapter 4 - Mixing), and the "Check List for Certification at Ready-Mixed Concrete Production Facilities" of the Material Ready-Mixed Concrete Association. After the concrete batch plant and mixers are placed in successful production operation, the testing agency inspects the production facilities and secures samples to determine that the concrete is being produced in accordance with Section 7.2 of ACI 301-72.

E.1.7 Exceptions Taken to Concrete Codes

ACI 301-72 Specifications for Structural Concrete for Buildings
Chapter 5- Reinforcement

Section 5.4 - Fabricating and Placing Tolerances: The following requirements apply in place of the requirements specified for items 5.4.2.1 and 5.4.2.4 of Paragraph 5.4.2, "Bars are placed to the following tolerances":

5.4.2.1 Clear distance to formed surfaces:

For #3 through #11 = ± 1/4 inch for straight bars.
± 1/2 inch for bent bars.
For #14 and #18 = ± 1 inch with minimum cover 1-1/4 inches.
± 1/2 inch with minimum cover 1-1/2 inches.

5.4.2.4 Spacing tolerance between bars parallel to the neutral axis for the following bar sizes is as follows, providing the minimum spacing is not violated:

For #3 through #11, #14, and #18 bars ± 2 inches.

Chapter 12 - Curing and Protection
Section 12.2 - Preservation of Moisture: The following requirements apply in place of all requirements specified in Section 12.2.

Initial Curing: Initial curing immediately follows the finishing operation. Concrete is kept continuously moist at least overnight in accordance with the following requirements:

a. Unformed Surfaces at Construction Joints: These surfaces are protected and cured by one of the following methods:

1. absorptive mat or fabric kept continuously moist, and

2. any one of the sheet materials specified in ASTM C171.

This protection and curing is placed over the entire surface immediately after completion of placing concrete and of any required surface treatment.

b. Unformed Surfaces for Rough or Finished Slabs, Floors, or Roofs: These surfaces are protected and cured with membrane curing compound applied immediately after completion of required surface treatment.

c. Formed Surfaces: All forms are protected, on either the face in contact with concrete or on the opposite face, with a suitable sealer to prevent moisture escape from the forms.

Final Curing: Immediately following the initial curing and before the concrete has dried, additional curing is accomplished in accordance with the following requirements:

a. Unformed-Surfaces at Construction Joints: Continue the same method used for initial curing until the next placement of concrete is made or until completion of the entire curing periods specified, whichever is the lesser time interval.

b. Unformed Surfaces for Rough or Finished Slabs Floors or Roofs: The requirements for initial curing provide the necessary protection for final curing.

c. Formed Surfaces:

1. If forms are left in place for the entire curing periods herein stated under "Exceptions" for Section 14.5 of ACI 301, then no further curing is required.

2. If forms are stripped before completion of the specified curing periods, membrane curing compound is applied to these stripped surfaces immediately after completion of specified surface treatment, or these stripped surfaces are kept continuously moist with a continuous water spray for the balance of the specified curing periods.

Duration of curing is stated under "Exceptions" for Section 14.5 of ACI 301.

Chapter 14 - Massive Concrete

Section 14.4 - Placing: The following requirements apply in place of Subsections 14.4.1 and 14.4.2:

14.4.1 Slump: Slump is not less than 2 inches, nor more than 4 inches, in place of a 2 inch maximum.

14.4.2 Placing Temperature: Maximum temperature of concrete when deposited is 75° F, in place of 70° F.

14.4.3 Layers: Restriction to 18 inch heights does not apply. Section 14.5 - Curing and Protection: The following requirements apply in place of all requirements of Section 14.5:

14.5.1 Minimum Curing Period: In place of a minimum curing period of 2 weeks, the following curing periods apply:

Curing Periods for Temperatures Above Freezing:

The following curing periods apply when the temperature of the surrounding air is above 40° F and is expected to remain above this temperature, for the entire curing period indicated:

a.	Concrete floors and stair treads with monolithic or steel trowel finish	10 days
b.	Roof slabs of job-placed concrete	7 days
c.	All other concrete work	5 days

Curing Periods During Freezing weather:

a. When the temperature of the surrounding air is 40° F or below, or is expected to drop to 40° F or below in the next 12 hours, provide adequate means for maintaining the following temperatures of surrounding air for the curing periods indicated:

1.	Concrete floors and stair treads with monalitic or steel trowel finish	10 days at 50° F or 8 days at 70° F
2.	Roof slabs of job-placed concrete.	7 days at 50°F or

5 days at 70° F

3. All other concrete work

5 days at 50° F or
3 days at 70°F

The design mix follows the standards of ACI 301-66.

ACI 306-66 - <u>Recommended Practice for Cold Weather Concreting:</u> The following addition applies:

Requirements for winter concreting conform to ACI 306-66, with the added requirement that curing periods are as herein stated under "Exceptions" for Section 14.5 of ACI 301-72.

ACI 605-59 - <u>Recommended Practice for Hot Weather Concreting:</u> The following exceptions apply:

<u>Temperature of Concrete as Placed</u>: The maximum temperature is 75° F when deposited, in place of 90° F maximum.

<u>Curing:</u> Is as herein stated under "Exceptions Taken to Concrete Codes", for Subsection 14.5.1 of ACI 301-72.

E.2 REINFORCING STEEL STANDARDS

E.2.1 Material Requirements

Reinforcing bars for- concrete consist of full size deformed bars meeting the requirements of ASTM A615-72.

Grade classifications are as follows:

Sizes	Grade
#3 to #11	60
#14 to #18	50

Grade 50 reinforcing bars are specially ordered reinforcement modified to obtain a guaranteed minimum yield strength of 50,000 psi and a minimum ultimate strength of 75,000 psi but maintaining a ductility of 12% elongation in an 8-inch gauge length.

E.2.2 Quality Control

E.2.2.1 Testing

Tests done on reinforcing bars measure the deformations, tensile and bending properties, and the chemical composition of the reinforcing bars in accordance with ASTM A615-72 and A370-72.

At least one full-diameter specimen from each bar size is tested for each 50 tons or fraction thereof from each heat. However, only one chemical analysis is done per heat.

E.2.2.2 Reinforcirq Bar Fabrication

Reinforcing bar fabrication conforms with the requirements of ACI 318-71, Chapter 7.

E.2.3 Cadwelding

Splices in reinforcing bar sizes #11 and smaller are lapped in accordance with ACI 318-71 or by cadwelding. Bar sizes #14 and #18 are spliced by cadwelding. The splice is designed to develop the specified minimum ultimate strength of the reinforcing bar.

E.2.3.1 Qualification of Operators

Prior to production splicing, each cadweld crew member prepares two joints for each position used in his work. These are tested and meet the joint acceptance standards of this subsection for workmanship, visual quality, and minimum tensile strength.

E.2.3.2 Procedure Specifications

All joints are made in accordance with the manufacturer's instruction sheets "Rebar Instructions for vertical Column Joints," plus the following additional requirements:

A manufacturer's representative, experienced in cadweld splicing of reinforcing bars, is present at the jobsite at the outset of the work to demonstrate the equipment and techniques used for making quality splices. He is present for the first 25 production splices to observe and verify that the equipment is being used correctly and that quality splices are being obtained.

The splice sleeves, exothermic powder, and graphite molds are stored in a clean dry area with adequate protection from the elements to prevent absorption of moisture.

Each splice sleeve is visually examined immediately prior to use to ensure the absence of rust and other foreign material on the inside diameter surface.

The graphite molds are preheated with an oxyacetylene or propane torch to drive off moisture at the beginning of each shift when the molds are cold or when a new mold is used.

Bar ends to be spliced are powerbrushed to remove all loose mill scale, rust, concrete, and other foreign material. Prior to power-brushing, all water, grease, and paint are removed by heating the bar ends with an oxyacetylene or propane torch.

A permanent line is marked 12 inches back from the end of each bar for a reference point to confirm that the bar ends are properly centered in the splice sleeve.

Immediately before the splice sleeve is placed into final position, the previously cleaned bar ends are preheated with an oxyacetylene or propane torch to ensure complete absence of moisture.

Special attention is given to maintaining the alignment of sleeve and guide tube to ensure a proper fill.

When the temperature is below freezing or the relative humidity is above 65%, the splice sleeve is externally preheated with an oxyacetylene or propane torch after all materials and equipment are in position.

E.2.3.3 Joint Testing

To ensure that the strength of each cadweld joint is equal to or greater than the specified minimum tensile strength for the particular bar size and ASTM specification, tensile tests are made at the following intervals:

a. one production splice out of the first ten production splices is tested,

b. one production and three sister splices for the next ninety production splices is tested, and

c. three splices, either production or-sister splices for the next subsequent 100 splices are tested. At least 1/4 of the total number of splices tested are production splices.

E.2.3.4 Joint Acceptance Standards

Sound, nonporous filler metal is visible at both ends of the splice sleeve and at the tap hole in the center of the splice sleeve. Filler metal is usually recessed 1/4 inch from the end of the sleeve due to the packing material and is not considered a poor fill.

Splices which contain slag or porous metal in the riser tap hole or at the ends of the sleeves (generally porosity) are rejected. A single shrinkage bubble present below the riser is not detrimental and is distinguished from general porosity as described above.

There is evidence of filler material between the sleeve and bar for the full 360°, however, the splice sleeves need not be exactly concentric or axially aligned with the bars.

The tensile strength of each sample tested is equal or exceeds 125% of the minimum yield strength specified in the ASTM standard appropriate for the grade of reinforcing bar using loading rates set forth in ASTM A370-72.

The average tensile strength of each group of 15 consecutive samples is equal to or exceeds the guaranteed ultimate tensile strength specified for the reinforcing bar.

E.2.3.5 Repairs

Splices that fail to pass visual inspection are discarded and replaced and are not used as tensile test samples.

If any production or sister splice tested fails to meet the tensile test specification and the observed rate of splices that fail the tensile test at that time does not exceed 1 for each 15 consecutive test samples, the sampling procedure is started anew.

If any production or sister splice used for testing fails to meet the tensile test specification and the observed rate of splices that fail the tensile test exceeds 1 for each 15 consecutive test samples, mechanical splicing is stopped. In addition, the adjacent production splices on each side of the last failed splice and 4 other splices distributed uniformly throughout the balance of the 100 production splices under investigation is tested, and an independent laboratory analysis is made to identify the cause of all failures.

If 2 or more splices from any of these 6 additional splice samples fail to meet the tensile test specification, the balance of the 100 production splices under investigation is rejected and replaced.

When mechanical splicing is resumed, the sampling procedure is started anew.

If the average tensile strength of the 15 consecutive samples fails to meet the provisions of Paragraph 6.5, the acceptability of the reduced average tensile strength is evaluated with respect to the required strength at the location from which the samples were taken.

E.3 POSTTENSIONING TENDONS

E.3.1 General

A BBRV posttensioning system is used. Tendons consist of 90 ¼-inch-diameter parallel lay wires. Positive anchorage at ends is provided by button-heading. The materials, erection and fabrication procedures, and testing requirements are in general compliance with the technical provisions of Sections CC-2400, CC-4400, and CC-5400 of the ASME B&PV Code, Section III, Division 2, July 1, 1977 edition with exceptions noted in Section E.9.

E.3.2 Materials

E.3. 2. 1 Tendon Materia1

A 1/4-inch-diameter wire conforms to cold-drawn ASTM A421-65, Type BA, stress-relieved, having a guaranteed minimum ultimate tensile strength, (f_{PU}), of 240,000 psi and a minimum yield strength not less than 0.80 f_{PU} as measured by the 1.0% PU extension under load method.

E.3.2.2 Buttonheads

The positive anchorage of tendons to anchor heads is provided by button-heading of the wires.

All buttonheads are cold-formed after threading wires through wire holes of anchor heads. Buttonheads are formed symmetrically about the axis of wires and are free from harmful seams, fractures, and flaws.

E.3.2.3 Tendon Sheathing

Tendon sheathing through the foundation consists of black seamless steel pipe, ASTM A106-72a, Grade A and/or ASTM A53. Wall sheathing is a black interlocked steel strip conduit, 24-gauge minimum wall thickness, fabricated to be watertight. The outside diameter of the sheathing is approximately 4 inches. All splices are sealed to prevent intrusion of cement paste. The tendon sheath splice is made using a snugly fitting coupling approximately 1 foot long. The joints between the sheath and the coupling are taped. The minimum radius of curvature used is 20 feet.

E.3.2.4 Grease

A corrosion-preventing grease is used as a tendon casing filler.

E.3.2.5 Anchor Heads

The anchor heads conform to AISI C1141 or HR 1144/11L44.

E.3.2.6 Bearing Plate

The bearing plate material conforms to ASTM A36-70a.

E.3.3 Quality Control

E.3.3.1 Testing

E.3.3.1.1 Tendon Tests

Tensile tests are done on the full 90-wire tendon. Tensile tests are done in accordance with applicable ASTM standards. These tests are done on 1 out of every 100 tendons fabricated.

E.3.3.1.2 Tests on Wires and Buttonheads

Sample tests are done on the wires in accordance with ASTM A421-65. All buttonheads are inspected visually. In addition, 10% are tested with a go and no-go gauge.

E.3.3.1.3 Grease Tests

Tests are made on each shipment of grease. These tests are ASTM D512-67, ASTM D992-71, and APHA tests on the sulphide content.

E.3.3.1.4 Anchor Hardware Test

A minimum of 10% of the anchor heads are subjected to the Rockwell Hardness Test.

E.4. STRUCTURAL STEEL

E.4.1 Structural Steel Materials

Structural support steel is ASTM A36-70a, ASTM A572-72, Grade 50, and ASTM 588-71 high strength, low alloy corrosion-resistant steel. Round tubular bracing is ASTM A53-72a Type E or S Steel.

E.4.2 Structural Steel Connections and Connection Materials

E.4.2.1 Bolted Connections

Structural steel bolted connections use ASTM A325-71 Type 1 and ASTM A490-71 friction-type high-strength bolts, except at sliding connections, where high strength bolts in bearing type are used. High-strength bolted connections conform to "Specification for Structural Joints Using ASTM A325 or A490 Bolts" issued by the Research Council on Riveted and Bolted Structural Joints of the Engineering Foundation and endorsed by the AISC, and to Table I or 11 of AISC Manual-69. ASTM A36 nuts are used with all ASTM A36 threaded rods and all ASTM A307-68 headed bolts. A194-71, Grade 2 (or better) ANSI series nuts are used with all ASTM A193-71, Grade B7 (or better), threaded rods.

Structural steel galleries have been provided at seven elevations in the drywell area of the containment and in various areas of the plant where platforms are needed for access to the valves and other equipment. Sliding connections are provided to these gallery framings, to accommodate movement due to thermal expansion or movement due to containment wall growth under pipe break condition. The gallery framing inside containment consists of radial beams spanning between the sacrificial shield and containment wall and tangential beams spanning between the radial beams. Both radial and tangential beams have a sliding connection at one end to allow for expansion and a fixed connection at the other end. Bearing type connections have been used to transfer the load at the sliding end of beams. Friction type connections have been provided at the fixed end. The bolts at the sliding connection are provided with a jam nut in addition to the regular high strength hexagonal nut to ensure against loosening or loss of the high strength nut. To bring connection angles and connection plates to a snug position, the sliding connection bolts are tightened to a pretension load of 5.0k to 10.0k.

E.4.2.2 Welded Connections

Standard welded beam connections conform to Table III or IV of AISC Manual-69.

Welding procedures for shop and field welding are in accordance with the AWS specifications listed in Table 3.8-2. Selection of electrodes and recommended minimum preheat and interpass temperature are in accordance with AWS requirements. All welders and welding operators are certified by an approved testing laboratory and are qualified under AWS procedure set forth in AWS specifications.

E.4.3 Quality Control

E.4.3.1 General

Quality assurance requirements apply to the fabrication and testing of structures and components. Certified material test reports are furnished stating the actual results of all chemical analyses and mechanical tests required by ASTM specifications. Identifying heat numbers are furnished on all structural steel to trace the steel to the specific heat in which the steel was made.

E.4.3.2 Testing and Inspection of Weldments

The following test methods are used for the inspection of welds:

a. radiographic inspection,

b. ultrasonic inspection,

c. magnetic particle inspection, and

d. liquid penetrant inspection.

The above nondestructive test methods are in compliance with the following ASTM specifications: E94-68, E142-72, E114-63, E164-65, E109-63, E138-63, and E165-65.

Visual weld inspection is in accordance with guidelines prepared by the Nuclear Construction Issues Group, NCIG-01, Rev. 2, titled "Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants".

E.4.3.3 Fabricaticn

The fabrication of structural steel conforms to AISC specifications.

E.5 CONTAINMENT LINER, THE REACTOR SUPPORT PEDESTAL LINER, AND THE DRYWELL FLOOR LINER WITHIN THE CONTAINMENT BACKED BY CONCRETE

E.5.1 General

The materials, erection and fabrication procedures, and testing requirements are in general compliance with the technical provisions of Sections CC-2500, CC-4500, and CC-5500 of the ASME B&PV Code, Section III, Division 2, July 1, 1977 edition with exceptions as noted in Section E.9.4.

E.5.2 Materials

The containment liner materials performing only a leaktight function (excluding leak test channels), the reactor support pedestal liner and the drywell floor liner within the containment backed by concrete meet the requirements of the 1971 ASME Boiler and Pressure Vessel Code (ASME B&PV Code), Summer 1972 Addenda, Section III, Paragraph NE-2300, and comply with the following specifications:

APPLICATICN	SPECIFICATION
Drywell liner	SA 516-72 GRADE 60
Suppression chamber, reactor support pedestal, and drywell floor liners	SA 240-72 TYPE 304
Containment liner anchors	A36-70a
Reactor support pedestal and drywell floor anchors	ASTM A108-69

E.5.3 Quality Control

E.5.3.1 Testinq of welds

E.5.3.1.1 General

All nondestructive examination procedures are in accordance with Appendix X of section III and Section V of the 1971 ASME B&PV Code. Acceptance criteria for nondestructive examinations are in accordance with the 1971 ASME B&PV` Code, Section III, NE-5300.

E.5.3.1.2 Liner Plate Seam welds

E.5.3.1.2.1 Radiograohic Examinations

The first 10 feet of weld for each welder and welding position is 100% radiographed. Thereafter one spot radiographing of not less than 12 inches in length is taken for each welder and welding position in each additional 50-foot increment of weld. In any case, a minimum of 2% of linear seam welds is examined by radiography. All radiographic examinations are performed as soon as possible after the weld is placed. The spots selected for radiography are randomly picked. No two spots chosen for radiographic examination are closer than 10 feet apart. If a weld fails to meet the acceptance standards specified in NE-5120, Section III of the ASME B&PV Code, two additional spots are radiographically examined. These two additional spots are at least 1 foot from the spot of initial examination. If either of these two additional welds fails to meet the acceptance standards, then the entire weld test unit is considered unacceptable. Either the entire unacceptable weld is removed and the joint rewelded, or the entire weld unit is completely radiographed and the defective welding repaired. The repaired areas are spot radiographed. Records of all radiographs are stored in accordance with provisions of the 1971 ASME B&PV Code, Section III.

E.5.3.1.2.2 Ultrasonic Examinations

Ultrasonic examinations are done on 100% of the liner seam welds where radiographic examinations of liner seam welds are either not feasible or the welds are not accessible for radiographic examination. If a weld fails to meet the acceptance standards specified in NE-5120 of Section III of the-ASME B&PV Code, the weld is repaired and reexamined according to the above code using an ultrasonic examination.

E.5.3.1.2.3 Magnetic Particle Examination

The magnetic particle examination is performed on 100% of liner seam welds for ferritic material. If a weld fails to meet the acceptance standards specified in NE-5120 of Section III of the ASME B&PV code, the weld is repaired and reexamined according to the above code using the magnetic particle examination.

E.5.3.1.2.4 Liquid Penetrant Fxamination

The liquid penetrant examination is performed on 100% of liner seam welds for austenitic material. If a weld fails to meet the acceptance standards specified in NE-5120 of Section III of the ASME B&PV code, the weld is repaired and reexamined according to the above code using the liquid penetrant method of examination.

E.5.3.1.2.5 Vacuum Box Soap Bubble Test

The vacuum box soap bubble test is performed on 100% of liner seam welds for leaktightness. If leakage is detected, the test is done again after the weld is repaired.

E.5.3.1.3 Leak Test Channels

The leak-chase-system channels are installed over the liner welds. The channel-to-liner plates are tested for leaktightness by pressurizing the channels to the containment design pressure and doing a pneumatic test on 100% of the welds. A 2-psi change in pressure over a 2-hour hold period is allowed due to the variation in temperature during the holding period.

E.5.3.2 Fabrication and Erection

E.5.3.2.1 General

The fabrication and erection of the containment steel boundaries backed by concrete are in accordance with the 1971 ASME B&PV Code, Section III, Division I, Subsection NE-4000.

E.5.3.2.2 Qualification of welders

The qualifications of welders and welding procedures are in accordance with Sections III and IX of the 1971 ASME B&PV Code.

E.6 CONTAINMENT STEEL BOUNDARY NOT BACKED BY CONCRETE

E. 6.1 Materials

The materials comply with the requirements of the 1971 ASME B&PV Code, Summer 1972 Addenda, Section III, Division I, Paragraph NE-2300, and also to the following specifications:

APPLICATION	SPECIFICATION
Drywell head	SA-516 Grade 70
Personnel access airlock and equipment hatch	SA-516 Grade 70
Penetration Sleeves	SA-333 Grade 1 or 6 SA-516 Grade 70 SA-240 Type 304 SA-312 Grade TP 304
Penetration Head Fittings	SA-516 Grade 60 SA-240 Type 304 SA-240 Type 316 SA-350 Grade LF1

E.6.2 Quality Control

E.6.2.1 Testing

E.6.2.1.1 General

The testing of the containment leak boundaries not backed by concrete is in accordance with the 1971 ASME B&PV Code, Section III Division I, Subsection NE-5000.

E.6.2.1.2 Testing of welds

One hundred percent of all welds between penetration and flued fitting and flued fittings and pipelines is examined by radiographic examinations. One hundred percent of all welds in the drywell head, equipment hatch, personnel airlock, and penetration sleeves is inspected also by radiographic examination. If a weld-fails to meet the acceptance standards specified in NE-5120, Section III of the ASME B&PV Code, two additional spots are radiographically examined. These two additional spots are at least 1 foot from the spot of initial examination. If either of these two additional welds fails to meet the acceptance standards, then the entire weld test unit is considered unacceptable. Either the entire unacceptable weld is removed and the joint rewelded, or the entire weld unit is completely radiographed and the defective welding repaired. The repaired areas are spot radiographed. Records of all radiographs are stored in accordance with provisions of Subsection NE-5000 of the 1971 ASME B&PV Code, Section III, Division I.

E.6.2.2 Fabrication and Erection

E.6.2.2.1 General

The fabrication and erection of the containment steel boundaries not backed by concrete are in accordance with the 1971 ASME B&PV Code, Section III, Division I, Subsection NE-4000.

E.6.2.2.2 Qualification of Welders

The qualif ications of welders and welding procedures are in accordance with Section III and IX of the 1971 ASME B&PV Code.

E.7 STAINLESS STEEI, ,POOL LINERS

E.7.1 Materials

Stainless steel pool liners are fabricated from SA-240-72, Type 304 material.

E.7.2 Welding

Welding and fabrication of the liners is in accordance with the applicable parts of the 1971 ASME B&PV Code, Unfired Pressure Vessels, Subsection B, Part UW. All liner seam welds have complete penetration welds. Acceptance criteria f or nondestructive examinations are in accordance with the 1971 ASME B&PV Code, Section III, NE-5210, NE-5220, and NE-5230.

The liner seam welds are examined and tested as follows:

a. radiographic examination on 2% of liner seam welds,

b. ultrasonic examination on 100% of liner seam welds wherever the joint detail does not permit radiographic examination to be performed in accordance with ASME Code,

c. liquid penetrant examination on 100% of liner seam welds for the austenitic material, and

d. vacuum box soap bubble test on 100% of liner seam welds.

These welds are tested in the same manner as the welds discussed in Subsection E.5.3.

E.8 OTHER ELEMENTS

E.8.1 Stainless Steel Requirements

Stainless steel plate bars and flats, angles, and pipe are hot rolled, annealed, and pickled. The finish is comparable to No. 2B sheet finish in accordance with ASTM A480-71.

For further discussion on austenitic stainless steel, refer to Subsection 5.2.3.4.

E.9 EXCEPTIONS TO ASME B&PV CODES

E.9.1 Concrete

E.9.1.1 Cement

For ASTM testing methods C114, C185, C204, C115, C451, and C186, one initial test is made by each test supplier.

E.9.1.2 Fly Ash

For ASTM testing methods C618 and C311, one initial test is made by each supplier. Sulfur trioxide, sodium oxide, loss on ignition, and fineness tests are done on every 200 tons.

E.9.1.3 Aggregates

ASTM tests C142, C117, C123, C128, C88, C131, C535, C87, and C227 are done once by each supplier. ASTM tests C117 and C29 are done weekly. ASTM tests C235, C125, C39, C78, C666, C289, C295, C342, C586, and CRDC119 are done on request.

E.9.1.4 Admixtures

ASTM tests C260 and C949 are done on request. ASTM test C512 is done monthly.

E.9.1.5 Water and Ice

Tests C109, C191, C151, and D1888 are not done. ASHO Designation T-26-70 is done initially by each supplier.

E.9.1.6 Concrete Testing

Test C94 is not done.

E.9.2 Steel Reinforcement

ASTM test A615 for deformations is done on every 50 tons or one time per heat.

E.9.3. Prestressing System

E.9.3.1 Materials for Prestressing

CC2423, CC2424, CC2433.2.2, CC2442.3.2

E.9.3.2 Fabrication and Installation of Prestressing Svstems

CC4431.1, CC4432.5, CC4451, CC4464.1

E.9.3.3 Examination of Prestressing System

CC5421, CC5423.1, CC5423.2

E.9.4. Containment Liner, Reactor Support Liner and Drywell Floor Liner

CC2511, CC2532, CC4534, CC4542.1, CC4543.5, CC4544, CC4552, CC4560, CC5521, CC5524, CC5525, CC5531, CC5535, CC5536, CC5546.1.1, and CC5547.

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APPENDIX F

COMPUTER PROGRAMS

The computer programs referred to in Sections 2.5, 3.7, and 3.8 by their acronyms are described herein. All programs are verified, within the stated assumptions and limitations, for correctness of utilized theory and validity of obtained results for a variety of typical problems. Results are checked against known solutions, solutions obtained from other programs, or hand calculations. Examples of validation problems are included with the program descriptions. Whenever applicable, internal checks such as equilibrium and orthogonality checks are included as an aid in checking the validity of the results obtained from the computer program for each problem analyzed. Additionally this appendix describes some computer programs that are not referenced in the UFSAR, but that have historical significance. This is not an all-inclusive list of computer programs, but highlights those of historical significance.

F.1 CONCRETE

CONCRETE is a computer program used for statistical evaluation of concrete strength. It sorts and analyzes the field data collected on concrete samples and presents it in a convenient form for interpretation.

CONCRETE was developed by Sargent & Lundy.

The compressive strength test results of concrete cylinders are statistically analyzed to obtain the mean, standard deviation, coefficient of variation, moving averages, and other statistical parameters required in the quality appraisal of concrete according to ACI 214-65. The strength results are also compared with the quality control limits fixed according to the ACI 318-71 Code and the ASTM Manual on Quality Control of Materials, 15-C. Any violations or inadequacies are clearly pointed out in the output.

To demonstrate the validity of the program, a sample problem from “Notes on ACI 381-71 Building Code Requirements with Design Applications” (Reference 1) was processed on CONCRETE. The problem determines the average 28-day strength and standard deviation for 46 test cylinders. CONCRETE’s results, shown in Table F.1-1, are identical to the hand calculation solutions in the ACI notes.

F.2 CSEF-III

CSEF-III (Circular Slab on an Elastic Foundation), analyzes any circular slab for arbitrary load conditions. The analysis is based on the solution to the basic differential equation for a plate on an elastic foundation, modified to include a Fourier series representation for the circumferential variation. For each Fourier harmonic, a matrix of linear equations is formulated using a finite difference representation of the plate equation and boundary equations. The system of equations is then solved by means of a Gauss elimination method.

Arbitrary normal pressure loading, edge moment and/or shear loads, and axisymmetric thermal gradient loads may be considered in the analysis. At any radius, the entire circumference may be fixed against rotation or displacement, allowing for a variety of support conditions or boundary conditions.

The program output includes deflection, soil pressure, radial moment, hoop moment, twisting moment, radial shear, tangential shear, and Kirchhoff shear for each Fourier harmonic. Results from individual harmonics are superimposed and output along the radii at specified angles. Moments may be converted to a Cartesian coordinate system for comparison with other programs.

Version III of CSEF was developed by Sargent & Lundy in 1971

Two plate analyses are presented as examples of validation.

The first example is a concrete circular plate resting on an elastic foundation. The plate parameters are as follows:

Radius	$a = 50$ ft,
Thickness	$h = 5$ ft,
Elastic Modulus	$E = 576{,}000$ ksf,
Poisson's Ratio	$\mu = 0.17$, and
Foundation Modulus	$K = 518.4$ kip/ft^3.

The plate is analyzed for a uniform pressure load:

$P = 50$ kip/ft^2

And a uniform edge load:

$M = 1000$ kip/ft/ft.

The results of the CSEF-III analysis are compared with a hand-calculated solution. The hand calculations are based on equations presented in Reference 2. The displacements for the distributed load and edge moments are found independently and superimposed. For the uniform pressure load the displacement is:

$$w = p/k$$

and for the edge moment:

$$w = D_1 Z_1 (\nu r) + D_2 Z_2 (\nu r)$$

where:

$\nu = 4\sqrt{k / D}$

$Z_1 (x) = R_e \ J_o \ (x\sqrt{i})$

$Z_2 (x) = I_m \ J_o \ (x\sqrt{i})$

J_o = Bessel function of the first kind

and

$$D_1 = \frac{M\nu^2}{K} \frac{Z_1' (\nu a)}{Z_1 (\nu a) Z_2' (\nu a) - Z_1' (\nu a) Z_2 (\nu a) + \frac{1-\mu}{\nu a} Z_1'^2 (\nu a) + Z_2'^2 (\nu a)}$$

$$D_2 = \frac{M\nu^2}{K} \frac{Z_2' (\nu a)}{Z_1 (\nu a) Z_2' (\nu a) - Z_1' (\nu a) Z_2 (\nu a) + \frac{1-\mu}{\nu a} Z_1'^2 (\nu a) + Z_2'^2 (\nu a)}$$

$$Z_1' \text{ and } Z_2' = \frac{dZ(x)}{dx} .$$

The results obtained by evaluating these equations for displacement and the appropriate expressions for radial moment are presented along with those obtained from CSEF-III in Figures F.1-1 and F.2-1, respectively. As illustrated in these figures, the independent solutions compare favorably.

For a second example, a simply supported circular plate under a linearly varying pressure load is presented. The solution is obtained for a plate having the following properties and loading:

Radius	a =	100 in.,
Thickness	h =	2 in.,
Elastic Modulus	E =	3.0×10^7 psi,
Poisson's Ratio	ν =	0.3, and
Loading Intensity	q =	$\dfrac{pr \cos \phi}{a}$,

where:

p = 100 psi.

Results from CSEF-III are compared with those obtained for Kalnin's Shell of Revolution (Reference 3). The comparison for radial moment and displacement are shown in Figures F.3-1 and F.4-1. Solutions obtained from both programs favorably compare with a solution presented by Timoshenko and Woinowski-Krieger in Reference 4.

F.3 DAPS-II

DAPS (Dynamic Analysis of Plane Strain or Plane Stress Systems) analyzes the dynamic response of two-dimensional plane stress or plane strain finite element systems to an earthquake. The equations of motion are solved by a stable step-by-step integration procedure or modal analysis. Quadrilateral, triangular, truss, beam, shear spring, and rocking spring elements are included. For step-by-step integration, the program output includes relative nodal displacements, absolute accelerations and stresses at selected time intervals, maximum relative displacements, absolute accelerations and stresses, response spectra frequencies, and modal vectors.

For modal analysis, the output includes maximum displacements, velocities, accelerations, stresses, response spectra, modal frequencies and vectors, participation factors, and an orthogonality check.

DAPS was originally developed by E. L. Wilson of the University of California, Berkeley (Reference 5). It was acquired by Sargent & Lundy.

One of the problems used to validate the program is presented. The ground acceleration is applied at the fixed end of a cantilever beam (Figure F.5-1). Table F.2-1 and Figure F.6-1 compare the results obtained by Wilson et al using the SAP-IV program (Reference 6) with the results obtained from DAPS-II. As illustrated in these figures, the results show good agreement.

F.4 DYNAS

DYNAS (Dynamic Analysis of Structures) is designed to perform dynamic analysis of structures which can be idealized as three-dimensional space frames and/or rigid slabs connected together by translational or torsional springs. The program considers the combined effects of translational, torsional, and rocking motions on the structure. The program uses either the response spectrum or the time-history method of analysis, depending on the type of forcing function available. Each method uses a normal mode approach. In the case of time-history analysis, the decoupled differential equations of motion are numerically integrated using Newmark's β-method (Reference 7).

The DYNAS program is capable of analyzing structures with parts having different associated dampings. An option is also available to analyze a large structural system using a modal synthesis technique. In this option, the system is divided into subsystems whose modal characteristics are computed separately and then synthesized to obtain the response of the complete system. The input base motion can be applied simultaneously in two orthogonal directions. A response spectrum can be generated at specified slabs or joints.

The program output includes modal responses, probable maximum responses, a time-history of structural response, and a response spectrum at specified joints.

The DYNAS program was originally developed by Sargent & Lundy in 1970.

The solutions of two of the problems used for validating DYNAS are presented.

In the first problem, a three-story shear building is analyzed and compared to a solution obtained by Biggs (Reference 8). The structure in conjunction with the applicable masses and stiffness values is represented by the closed-coupled system shown in Figure F.7-1. For this analysis the following response spectrum was used:

Frequency	Displacement
1.00 cps	3.30 in.
2.18 cps	1.40 in.
3.18 cps	0.66 in.

The results obtained by Biggs and from DYNAS are compared in Tables F.3-1 through F.6-1.

In the second example, results of DYNAS are compared to those obtained by Wilson et al. (Reference 5) using the SAP-IV program.

An acceleration is applied at the fixed end of a cantilever beam (Figure F.8-1). The natural periods calculated by both SAP-IV and DYNAS are shown in Table F.7-1. A comparison of the bending moment at the fixed end of the cantilever beam is shown in Figure F.9-1. As demonstrated in both examples, DYNAS performs an accurate analysis.

F.5 DYNAX

DYNAX (Dynamic Analysis of Axisymmetric Structures) is a finite element program capable of performing both static and dynamic analyses of axisymmetric structures. Its formulation is based on the theory of small displacement.

Three types of finite elements are available: quadrilateral, triangular, and shell. The geometry of the structure can be general as long as it is axisymmetric. Both isotropic and orthotropic elastic material properties can be modeled. Discrete and distributed springs can be used to model elastic foundations, etc.

For static analysis, input loads can be structural weight, nodal forces, nodal displacements, distributed loads, or thermal loads. Loads can be axisymmetric or nonaxisymmetric. For solids of revolution, the program outputs nodal displacements and element stresses in the global system (radial, circumferential, and axial) and element and nodal stress resultants in a shell coordinate system. For shells of revolution, the output consists of nodal displacements as well as element and nodal stress resultants in a shell coordinate system (meridional, circumferential, and normal).

For dynamic analysis, three methods are available: direct integration method, modal superposition method, and response spectrum method. Dynamic analysis by direct integration or modal superposition method uses a forcing function input via either 1) nodal force components versus time for any number of nodes, or 2) vertical or horizontal ground acceleration versus time. For nonaxisymmetric loads, the equivalent Fourier expansion is used. Dynamic analysis by the response spectrum method uses a spectral velocity versus natural frequency input with up to four damping constants. The output of dynamic analysis provides nodal displacements, element stresses, and resultant forces and moments at specified time steps. When the modal superposition method is used for earthquake response analysis, the prescribed number of frequencies and mode shapes are computed and printed along with the cumulative response of all specified modes by the root sum square (RSS) method and the absolute sum method.

DYNAX was originally developed under the acronym ASHAD by S. Ghosh and E. L. Wilson of the University of California, Berkeley, in 1969 (Reference 9). It was acquired by Sargent & Lundy.

Validation of the major analytical capabilities of DYNAX is demonstrated by a comparison of the results from six documented problems with DYNAX results.

The first problem is taken from S. Timoshenko's book Theory of Plates and Shells (Reference 10). A clamped shallow spherical shell, shown in Figure F.10-1, is analyzed for displacements and stresses produced by a uniform pressure applied on its outside surface. DYNAX and Timoshenko's solutions are compared in Figures F.11-1 and F.12-1.

The second problem, taken from Theory of Elasticity by Timoshenko and Goodier (Reference 11), is a plane strain analysis of a thick-walled cylinder subjected to external pressure. The finite element idealization and the loading system used for this case are shown in Figure F.13-1. Results of the DYNAX analysis are compared with the exact solution in Figure F.14-1. The agreement for both stresses and displacements is excellent.

The third problem is taken from an article by Budiansky and Radkowski (Reference 12). The structure, illustrated in Figure F.15-1, is a short, wide cylinder with a moderate thickness to radius ratio. The applied loads and the output stresses are pure uncoupled harmonics. For this finite element analysis, the cylinder is divided into 50 elements of equal size. This problem solves for harmonic deflections, element stresses, and forces. Figures F.16-1 and F.17-1 compare DYNAX results with the results given in the article.

The fourth problem is taken from an article by Reismann and Padlog (Reference 13). A ring (line) load of magnitude P (500 pounds) is suddenly applied to the center of a freely supported cylindrical shell. The dimensions of the shell and the time-history of the load are shown in Figure F.18-1. Because of symmetry, only one-half of the cylinder is modeled using 80 elements of equal size. The time-history of radial deflection and meridional moments from DYNAX and from Reismann and Padlog are compared and are shown in Figures F.19-1 and F.20-1, respectively.

For the fifth problem, the method of mode superposition is used to solve a shallow spherical cap with clamped support under the action of a suddenly applied uniformly distributed load. The dimensions of the shell and the load time-history are shown in Figure F.21-1. The first 12 modes were considered to formulate the uncoupled equations of motion. Each of these equations was solved by the step-by-step integration method using a time step of 0.1×10^{-4} seconds. The results are compared graphically with those obtained by S. Klein (Reference 14) in Figures F.22-1 and F.23-1.

The sixth problem is a hyperbolic cooling tower, as shown in Figure F.24-1. The tower is analyzed for horizontal earthquake motion. A response spectrum for 2% damping, as shown in Figure F.25-1, was used for this analysis. The RMS values of the meridional force are compared with those obtained by Abel et al. (Reference 15) in Figure F.26-1.

As shown in these six examples, DYNAX is capable of producing accurate results for both static and dynamic analyses of shells.

F.6 EASE

EASE (Elastic Analysis for Structural Engineering) performs static analysis of two- and three-dimensional trusses and frames, plane elastic bodies, and plate and shell structures. The finite element approach is used with standard linear or beam elements a plane stress triangular element, or a triangular plate bending element. The EASE program accepts thermal loads as well as pressure, gravity, or concentrated loads.

The program output includes joint displacements, beam forces, and triangular element stresses and moments.

EASE was developed by the Engineering Analysis Corporation, Redondo Beach, California, in 1969 and is in the public domain. It was used by Sargent & Lundy.

F.7 EMBANK

EMBANK (Embankment Analysis) is a finite element program used for the evaluation of embankment stresses and deformations. Because construction sequence plays an important part in the deformations developed in earth embankments, an incremental version of the finite element method is employed to account for this effect. In addition, this method has the capability of employing nonlinear, stress-dependent soil stress-strain behavior. The soil constitutive relations are expressed in terms of bulk modulus and deformation modulus, and it is assumed that only the deformation modulus varies with strain.

To take account of incremental construction, the finite element idealization is arranged in horizontal layers corresponding to construction lifts. The analysis then involves the evaluation of stresses and deflections in a succession of structures, corresponding to various stages of proposed construction. In the first analysis, for example, only the lowest layers of elements are considered. The program evaluates the stiffness of these elements and the dead weight forces which they produce and then calculates the stresses and displacements developed in the lowest layer due to its own weight.

After sorting the stresses and displacements resulting from the first lift analysis, the program next considers the structure consisting of the lowest two lifts. The stiffness of all elements in these two lifts is evaluated, but only the dead weight forces of the upper lift are considered. The program calculates stresses and displacements developed throughout the structure as a result of this new increment of load and adds these stress and displacement results to those obtained for the preceding increment. By repeating this procedure for each successive construction lift, the complete history of the development of stresses and displacements can be obtained.

The program output includes the displacements of each node and the resultant direct stresses, shear stresses, and principal stresses with their associated directions for each element.

EMBANK is the Sargent & Lundy version of the LSBUILM program. It was originally developed by F. H. Kulhawy, J. M. Duncan, and H. B. Seed at the University of California, Berkeley, in 1969.

The validity of the original version was checked by Kulhawy, Duncan, and Seed (Reference 17) by comparing the measured movements of the Otter Brook Dam. Otter Brook Dam, a rolled earth dam constructed in New Hampshire, is about 130 feet high and has 2.5:1 slopes upstream and downstream. The calculated displacements from the program were compared to the measured values and are shown in Figure F.27-1. The variations of the horizontal displacement with height above the base of the dam are shown on the left, and the measured and calculated variations of the displacement of a bridge pier located midway on the upstream slope are shown on the right. It can be seen that in both cases the agreement is satisfactory, indicating that the analysis used in the program provides a rational basis for estimating embankment deformations.

To validate Sargent & Lundy's UNIVAC 1106 version of the program, example problems provided by the original authors were rerun. One of these examples is a 30-foot-high embankment (Figure F.28-1) built in three layers. The results for the final shear stresses at the center of various elements of a layer and the x-displacements at a layer junction are shown in Figure F.29-1.

The results obtained from the Sargent & Lundy version were identical with those from the original version of the program, thus indicating a successful implementation on the UNIVAC 1106.

F.8 INDIA

INDIA (Load-Moment Interaction Diagram) is a program used to compute the coordinates and plot the bending moment-axial load interaction diagram for a rectangular reinforced concrete section. The program will plot interaction curves for ultimate strength, yield strength, and working stress methods. Both compression and tension axial loads are considered as well as positive and negative moments for appropriate cross sections.

The procedures used for the working stress and yield stress methods are taken from ACI 318 Code. Equations used for the ultimate stress method are taken from a University of Illinois Civil Engineering Study (Reference 18).

INDIA was originally developed at Sargent & Lundy on the IBM 1130 in 1971.

To demonstrate the validity of the program, a sample problem, shown in Table F.8-1, was executed. Calculations were made by hand, and all results were found to be consistent with the theoretical approach.

F.9 SETTLE

SETTLE is a computer program developed to predict the magnitude of settlements of shallow foundation caused by the foundation load. Janbu's tangent modulus method (Reference 19) is employed to account for the nonlinear stress-strain behavior of soil.

The distribution of contact pressure considering the effect of the foundation structure is taken into account by assuming that the foundation is rigid and, therefore, the settlement distribution is approximately linear. The variations in contact pressure can then be determined from the conditions of force equilibrium and compatibility of the foundation and soil settlements.

The foundation settlement calculation is based on the following fundamental assumptions:

a. The soil profile can be divided into homogeneous horizontal layers with uniform thickness.
b. The stress increment caused by the applied loads can be approximated by the Boussinesq formula.
c. At any point, the stress increment contributed by each of the loading areas can be superimposed to calculate the total stress increment at that point.

Settlements at a point are computed by summing the individual settlement of each soil sublayer of a predetermined thickness. The following calculations are performed for each settlement point:

a. The stress increment caused by each loading area is computed, and the total influence at the center of each soil sublayer caused by all the loading areas is accumulated.
b. After the stress increments have been accumulated, the settlement they produced is computed and accumulated. Settlements are computed by Janbu's tangent modulus concept.
c. The settlement beneath a point is considered as the total of the individual settlement of each soil sublayer.
d. After the settlement at one point has been obtained, the SETTLE computer program proceeds to calculate settlement for the next point.

An iterative procedure is used so as to make the settlement patterns of the foundation and the subsoil compatible. If settlement patterns are not compatible, the distribution of contact pressure underneath the foundation is recalculated to satisfy the deformation prescribed by the subsoil. The new contact pressure distribution is then used to compute the new settlement pattern of the subsoil. The iteration continues until the predetermined convergence criteria are satisfied.

SETTLE was developed by Sargent & Lundy following the SETTLE tangent modulus concept. Results of settlement computation have been validated with those of the ICES-SEPOL program (Reference 20) combined with hand calculations. A typical example consists of a soil layer which is 31 feet thick and has two different soil types. This layer was loaded by three rectangular loading areas with varying intensities. Figure F.30-1 shows the close agreement of the settlement profiles computed by SETTLE and ICES-SEPOL. A 20- x 30-foot rectangular mat foundation loaded by four different loading areas was used to validate the stress distribution calculation. The mat foundation was subdivided into six elements, and a spring was assumed under the center of each element. The results of the SETTLE program and the hand calculation are exactly the same. These satisfactory results indicate a successful development of this program.

F.10 KALSHEL

KALSHEL (Kalnins' Shell of Revolution) is a computer program used to analyze thin axisymmetric shells of revolution for arbitrary load conditions. The solution is obtained by transforming the H. Reissner-Neisser equations to eight first-order ordinary differential equations (Reference 21). An Adams method of numerical integration is used as the basis for the solution of transformed equations. Since the program is based on classical shell theory, it has the same limitations.

The shell wall may vary in thickness along the meridian and consists of up to four layers of different isotropic or orthotropic materials. Branch shells may be connected to the main shell. Surface loads and line loads in the radial, tangential, and/or meridional directions and meridional moments as well as temperature distributions, which are assumed to vary linearly across the thickness, may be considered in the analysis. All loads may be asymmetric.

The program output includes shell displacements in the radial, tangential, and meridional directions, meridional rotations, meridional moment, hoop moment, meridional force, hoop force, transverse shear force, and twist shear force. In addition, outer fiber stresses calculated from the stress resultants may be obtained.

The program was originally developed by A. Kalnins of Lehigh University (Reference 3). It was acquired by Sargent & Lundy in 1969. This version was modified by Sargent & Lundy to sum displacements and stress resultants of the individual Fourier harmonics along meridians at specified angles.

A number of test cases were run to check the program options and validity of solution. One practical example included the analysis of a conical shell subjected to eccentric line load. The shell is made of two parts, cylindrical and conical, and both are of reinforced concrete with different thicknesses, as shown in Figure F.31-1. This problem was analyzed by this program and also by the public domain program SABOR-III (Reference 22).

Results from the two programs are compared in Figures F.32-1 through F.35-1. Figures F.32-1 and F.33-1 show a comparison of shell forces along a meridian at 0° (symmetric with respect to the load). Figures F.34-1 and F.35-1 show a comparison of shell forces around the circumference at an elevation where the load is applied. As shown in these figures, the results indicate comparable values.

F.11 LAFD

LAFD (Analysis of Linear Anchor Forces and Displacements) calculates the maximum force and displacement of anchors resulting from local buckling of thin plate liners anchored to concrete walls. The solution method used in LAFD is described in Reference 23.

First, anchor displacements are found for an assumed postbuckling load by a relaxation technique. Then, using this maximum displacement, the anchor force and the strain in the buckled plate are calculated. The stress-strain relation given in a paper by Young and Tate (Reference 24) is reestablished in the program. Using the calculated strain, first stress is found and then a new load. The new load is then used to find a new set of displacements. The procedure is repeated to find a second new load. This load is then compared to the load used in the previous cycle. The procedure is repeated until the loads obtained in the last two cycles are approximately zero.

The program is capable of analyzing four types of anchors: Nelson studs of 1/2-, 5/8- and 3/4-inch diameter, and 3- x 3-1/4-inch angle continuous rib anchors. The force-deformation relations of these anchors are obtained from the manufacturer's publication (Reference 25).

The program output includes the maximum anchor force, the maximum anchor deformation, and the postbuckling load of the buckled plate.

LAFD was developed by Sargent & Lundy in 1971.

To validate the program, significant calculations were verified with hand calculations. As an example of this validation, a comparison of these calculations is presented for a strip of liner having the following properties:

Strip span a	= 17.5 in.,
Plate thickness t	= 0.375 in.,
Strip width w	= 9 in.,
Modulus of Elasticity E	= 30 x 10^3 ksi, and
Yield Stress σ_o	= 36 ksi.

5/8-inch-diameter Nelson studs are used as anchors.

The anchor displacements, U_n, the force in the anchor adjacent to the buckled panel, f_1, and the postbuckling load P as calculated by the program are shown in Table F-9. Substituting these displacements into the appropriate force-deformation relationship for a 5/8-inch-diameter Nelson stud yields the anchor forces contained in Table F-10.

The validity of the solution is checked using the displacements and anchor forces given in Tables F.9-1 and F.10-1 to verify the equality of the original equations:

$$Fo\text{-}P = \frac{EA}{a}(U_1 - U_2) + f_1 \qquad (1)$$

$$O = \frac{EA}{a}(2U_n - U_{n-1} - U_{n+1}) + f_n \qquad (2)$$

$$N = 1, 2, 3 \ldots. N$$

The postbuckling load, P, as determined by Equation 1, is equal to 21.864K as compared to 21.978K obtained from the program. Substitution into Equation 2 satisfies the equation; equilibrium having been verified, the results obtained from the program are valid.

F.12 MESHG

MESHG (Mesh Generator) checks input data for finite element programs. Using the member incidences and node point coordinates as prepared for a finite element program such as SLSAP, PLFEM, DYNAX, and QUAD4, the program produces a Calcomp plot of the mesh.

Several isometric views of 3-D data may be obtained, axes may be rotated for 2-D data, and scaling may be specified. Element numbers are plotted proportional in size to element areas for ease in detecting errors in element connectivity or nodal coordinates.

The program was developed by Sargent & Lundy in 1970. Validity of the program is verified repeatedly by inspection of each plotted mesh.

F.13 PLFEM-II

PLFEM (Plate Finite Element Method) analyzes plane elastic bodies, plates, and shell structures by the stiffness matrix method. The program uses two finite elements, a rectangular element and a triangular element.

Elastic spring supports and/or an elastic foundation may be considered in the analysis. Orthotropic materials may also be considered in conjunction with the rectangular element. Pressure loads, concentrated forces, nodal displacements, and thermal loads may be considered in the analysis. All loading cases may be factored and/or combined in any manner.

The program output includes deflections and rotations of all joints and membrane stresses (normal, shearing, and principal) at the center of each element, the resultant moments (X, Y, twisting principal), and shears and reaction forces. An equilibrium check is made to determine the accuracy of the results.

PLFEM was developed by Sargent & Lundy.

Three sample problems are presented to demonstrate the validity of PLFEM. Plots of the computer results obtained are compared with theoretical results by other methods.

The first problem is an analysis of a rectangular tank filled with water which was presented by Y. K. Cheung and J. D. Davies (Reference 26). The finite element used was presented by Zienkiewicz and Cheung in August 1964 (Reference 27). Experimental results agreed exactly with the finite element results except at a few isolated points where very small differences were noted. The PLFEM grid and loading for the tank problem are shown in Figure F.36-1. The grid used is the same size as the one used by Cheung and Davies. Moments in three regions of the tank are plotted along with the PLFEM results in Figures F.38-1 and F.39-1.

As a second example, a rectangular plate subjected to a uniform plane stress and having a circular hole in its center is analyzed. The grid used in the PLFEM analysis is shown in Figure F.40-1. Because of double symmetry, only one quarter of the plate is analyzed. Results obtained from the PLFEM analysis are plotted in Figure F.41-1 against the exact values as given by S. Timoshenko and J. Goodier in Reference 11.

As a final example, a square plate having a rectangular hole in its center is analyzed for the effect of a thermal gradient through the plate. The grid used in the PLFEM analysis is shown in Figure F.42-1. Only one quarter of the plate is analyzed because of the double symmetry. Moment values obtained by PLFEM are plotted for two regions of the plate in Figure F.43-1. For comparison, values of the moments obtained by an analysis based on the Hrennekoff framework analogy are also shown.

F.14 QUAD4

QUAD4 is a finite element program which evaluates the seismic response of soil structures using a different damping ratio for each individual element. The base motion can be applied simultaneously in two orthogonal directions. In addition, the procedure allows incorporation of strain-dependent stiffness and damping values for each element.

The program has been written for elements in plane-strain; triangular and quadrilateral elements can be used in representing the continuum. The solution proceeds by assigning modulus and damping values to each element. Because these values are strain-dependent, an iteration procedure is adopted. Thus values of shear moduli and damping are estimated at the outset, and the analysis is performed. Using the computed values of average strain developed in each element, new values of modulus and damping are determined from appropriate data relating these values to strain. Proceeding in this way, a solution is obtained incorporating modulus and damping values for each element which are compatible with the average strain developed.

The program output includes the strain compatible soil properties, response spectrum at specified nodes, probable maximum nodal accelerations, and probable maximum direct stresses, shear stresses, and shear strains for all elements. The shear stress time-history can also be obtained for specific elements.

QUAD4 was originally developed by I. M. Idriss, J. Lysmer, R. Hwang, and H. B. Seed of the University of California, Berkeley (Reference 28). It was acquired by Sargent & Lundy.

To validate Sargent & Lundy's version of QUAD4, a sample problem was taken from the original program documentation (Reference 28).

A 100-foot layer of dense sand as shown in Figure F.44-1 has been analyzed.

The properties of the sand were considered to be as follows:

Total unit weight = 125 pcf,

$(K_2)_{max}$ = 65, and

K_o = 0.5.

The parameter $(K_2)_{max}$ relates the maximum shear modulus, G_{max}, and effective mean pressure at any depth, y, below the surface as follows:

$$G_{max} = 1000\,(K_2)_{max}\,\sigma_m^{1/2}$$

Where:

$$\sigma'_m = \frac{(1 + 2K_o)}{3}\,\sigma'_v$$

and K_o = coefficient of lateral pressure at rest

and σ'_v = effective vertical pressure at depth y.

Damping values and variations of modulus values with strain were based on data given in Reference 28.

The response of the sand layer was evaluated using the time-history of accelerations recorded at Taft during the 1952 Kern County earthquake as base excitation. The ordinates of this time-history were adjusted to provide a maximum acceleration of 0.15g.

The sand layer has been represented by the finite element mesh shown in Figure F-44, which consists of 20 elements and 42 nodal points. To simulate a semi-infinite system, nodal points 1 through 40 have been fixed in the vertical direction and are therefore permitted to move only in the horizontal direction. Nodal points 41 and 42 are fixed to the base. A comparison of the results obtained from this validation run and the published results is shown in Figures F.45-1 and F.47-1. The values of damping and modulus compatible with the strain level computed in each element are presented in Figure F.45-1. The variations of maximum shear stresses and the maximum accelerations with depth are shown in Figure F.46-1. The acceleration spectrum for the computed surface motions is shown in Figure F.47-1. As illustrated by these figures, the comparisons are favorable.

F.15 RSG

RSG (Response Spectrum Generator) generates dynamic response spectra (displacement, velocity, and acceleration) for single-degree-of-freedom elastic systems with various dampings, subjected to a prescribed time-dependent acceleration. The differential equation of motion is solved using Newmark's β-method of numerical integration (Reference 7).

The program may also be used to obtain a response-spectrum-consistent time-history in which the response spectrum of the generated time-history closely envelopes the given spectrum.

The program has the capability of plotting the input time, acceleration function, and the response spectra output on the tripartite and/or acceleration versus period frequency grids.

Depending on the option, the program output includes the response spectra of a given time-history or the response-spectrum-consistent time-history.

RSG was developed by Sargent & Lundy in 1969.

One of the comparisons used for validation is presented.

The response spectrum for a one-degree-of-freedom damped system as presented by Biggs (Reference 8) was determined using RSG. The system was subjected to the sinusoidal ground acceleration shown in Figure F.48-1. A damping factor of 0.2 was used for this example. The response spectra obtained by Biggs and from RSG are also shown in Figure F.48-1. As demonstrated by this comparison, RSG generates an accurate response spectrum.

F.16 SEPOL

SEPOL (Settlement Problem-Oriented Language) is used to aid in the calculation of predicted magnitudes and progress of settlement of shallow foundations. It has four functions: INSITU establishes the soil profile and calculates required soil properties; STRESS DISTRIBUTION is used to calculate stresses due to foundation loading; SETTLEMENT calculates the magnitude of settlement; and RATE calculates the progress of settlement. These computations may be performed individually, or they may be combined into a single integrated computer run.

In STRESS DISTRIBUTION, the stresses and strains in three orthogonal directions in a loaded soil mass are calculated using Boussinesq's linear theory of elasticity, assuming the mass is homogeneous, isotropic, and semi-infinite. The input surface load may be built up by superimposing available basic load segments: rectangular, circular, or strip loading configurations (uniform, ramp, inverse ramp, or any). The required input soil properties are Young's modulus and Poisson's ratio.

In SETTLEMENT, the initial settlement is calculated first, then the consolidation settlement is calculated, and, finally, the initial and consolidation settlements are summed to obtain the total settlement at given surface locations. The calculations are made on a layer-by-layer basis and accumulated only for specified layers or finite depths. The consolidation settlement can be obtained by summing experimentally observed strains (that is, by inputting Young's modulus and Poisson's ratio over the compressible layers); by using the volume compressibility, m_v, the principal stresses, and the pore pressure parameter A; or by using the one-dimensional compressibility (inputting either a_v, C_c, or m_c).

In RATE, generation and dissipation of excess pore pressure using Terzaghi's one-dimensional consolidation theory and an arbitrary load history are calculated. The required input soil properties are the coefficients c_v and m_v. Both single and double drained layers are considered by SEPOL. The output consists of excess pore pressure, total pore pressure, and the degree of consolidation at given points and time, as well as time-settlement values a given surface points.

SEPOL (Reference 29) was developed at the Massachusetts Institute of Technology and is a subsystem of ICES (Integrated Civil Engineering Systems). It has been in the public domain since 1967. It was used by Sargent & Lundy.

F.17 SHAKE

SHAKE (Soil Layer Properties and Response/Earthquake) is a program which computes response in a horizontally layered semi-infinite system subjected to vertically traveling shear waves. Strain-compatible soil properties are computed within the program. Earthquake motion can be specified at any level of the soil profile, and a resulting motion can be computed anywhere else in the profile. The method is based on the continuous solution of the shear wave equation. For soil liquefaction studies, plots of stress time-histories at various levels in a soil profile can also be obtained.

The input for the program includes property data for the soil profile, curves of strain versus shear moduli and damping ratios, and the input earthquake motion.

The output includes the strain-compatible soil properties, response spectra of object and computed motions, printer and CALCOMP plots of time-histories, Fourier spectra, and response spectra. Stress time-history plots are also included.

SHAKE originally was developed by John Lysmer and P. B. Schnabel of the University of California, Berkeley (Reference 31). It was modified by Sargent & Lundy.

To verify Sargent & Lundy's version of SHAKE, results from the program were compared with results from a problem in a paper by Idriss and Seed (Reference 32).

The 100-foot layer of dense sand shown in Figure F.49-1 was analyzed. The properties of the sand were considered to be as follows:

Total unit weight = 125 pcf,

$(K_2)_{max}$ = 65, and

K_o = 0.5.

The parameter $(K_2)_{max}$ relates the maximum shear modulus, G_{max}, and effective mean pressure at any depth, y, below the surface as follows:

$$G_{max} = 1000\,(K_2)_{max}\,{\sigma'_m}^{1/2}$$

where:

$$\sigma'_m = \frac{(1 + 2K_o)}{3}\,\sigma'_v$$

K_o = coefficient of lateral pressure at rest, and

σ'_v = effective vertical pressure at depth y.

Damping values and the variation of modulus values with strain were based on published data for sands (Reference 33).

The response of the sand layer was evaluated using the time-history of accelerations recorded at Taft during the 1952 Kern County earthquake as base excitation. The ordinates of this time-history were adjusted to provide a maximum acceleration of 0.15g.

The results obtained from SHAKE and the published results are compared in Figures F.50-1 and F.51-1. The maximum shear stresses and accelerations from both solutions are compared in Figure F.50-1; the response spectra of the surface motions are compared in Figure F.51-1. As illustrated in these figures, the two solutions compare favorably.

F.18 SLOPE

SLOPE (Slope Stability Analysis) utilizes the theory of equilibrium of forces to determine the factor of safety against sliding of any embankment or slope. It contains the Bishop, Fellenius, and Morgenstern-Price methods of two-dimensional stability analysis. In the Bishop and Fellenius methods, the factor of safety against failure is estimated along a circular surface of failure, whereas any arbitrary failure surface may be chosen for the Morgenstern-Price method.

The input includes the slope geometry, soil profile, soil properties (density, cohesion, and the friction angle) and the piezometric surface(s). The program also has the capability to introduce an earthquake loading assumed as a horizontal gravitational force. Once the problem is input, several options can be used to determine the factor of safety by the various methods. In addition, different stages such as end-of-construction, full-lake, and sudden-drawdown, can be considered in a single run.

The output includes factors of safety for each trial surface and a plot of the slope cross section having slope profile, soil profile, water table conditions, and failure surface for the minimum factor of safety.

SLOPE was developed and put under ICES (Integrated Civil Engineering Systems) by William A. Bailey at the Massachusetts Institute of Technology. It has been in the public domain since 1967. It was used by Sargent & Lundy. (Reference 34).

F.19 SLSAP

SLSAP (Sargent & Lundy Structural Analysis Program) performs static and dynamic analyses of linear structural systems. The program uses the stiffness matrix method to analyze two- and three-dimensional frames, trusses, and grids; three-dimensional elastic solids; axially symmetrical solids, plates, and shells; and straight and curved pipes for arbitrary static loads. Dynamic analyses for frequencies and mode shapes, spectral analysis, and numerical integration analyses are also possible.

The program allows materials with arbitrary elastic constants, combined loadings, rigid members, elastic supports, and a combination of different element types.

The program output includes displacement and rotations of all joints or nodes, forces or stresses in members or elements, frequencies and mode shapes, and dynamic response in terms of displacements and forces.

SAP was developed by E. L. Wilson of the University of California at Berkeley. The original version of the program dates back to 1968.

Sargent & Lundy used the SAP IV version (Reference 6). It is used primarily for static analysis.

Results from the program have been compared with several other static and dynamic computer programs and classical solutions. Two examples of these validation problems are presented.

The first example is a cantilever beam under both uniform and concentrated load. (The beam was modeled for SAP using ten equal length beam elements.) It has a cross-sectional area of 1 x 2 inches, a length of 10 inches, and Young's modulus 30 x 10^3 ksi. A uniform load (q = 2 kips/inch) and a concentrated load (10 kip) are applied at one end of the beam.

The results from the program are compared to analytical results obtained by Timoshenko and Gere (Reference 35). Figure F.52-1 shows excellent agreement for the bending moment obtained in both solutions.

In the second example a simply supported square plate under uniform loading is analyzed. A 10-inch square by 1-inch thick square plate with Poisson's ratio = 0.3 and Young's modulus = 30 x 10^3 ksi was loaded with 1 ksi pressure.

The results obtained were compared to those presented by S. Timoshenko and S. Woinowsky-Krieger (Reference 36). The bending moments M_{xx} and M_{yy} for both the x and y symmetry lines obtained in the two solutions are shown in Figure F.53-1.

The maximum bending moment which occurs at the center of the plate differs by only 1.05%.

F.20 SOR-III

SOR-III (Shell of Revolution) is a computer program used to analyze thin shells of revolution subjected to axisymmetric loading by employing a generalized Adams-Moulton method to integrate numerically the governing differential equations.

Arbitrary distribution of normal, tangential, and moment surface loadings as well as edge forces and deflections may be analyzed in the axisymmetric loadings. Input of boundary conditions allows for the consideration of elastic support conditions. Temperature variations along the meridian or across the thickness may also be considered.

The program output includes shell displacements, outer fiber stresses, and strains and stress resultants.

SOR-III was developed at Knolls Atomic Power Laboratory for the United States Atomic Energy Commission (Reference 37). Version III was acquired by Sargent & Lundy in 1969.

Results from this program have been frequently compared with other available solutions and other computer programs to check the validity of the program. One of these comparisons is the analysis of a circular flat reinforced concrete plate. The details of the problem and the boundary conditions are shown in Figure F.54-1. Results of the SOR-III analysis were compared with the finite element program, SABOR-III (Reference 22). Figure F.55-1 shows the bending moment in the meridional and hoop directions, respectively. Figure F.56-1 shows the comparison of radial shear. As shown in these figures, results compare favorably.

F.21 SSANA

SSANA (Spring-Slab Analysis Program) was written to facilitate the reduction of the reinforced concrete shear building and the equipment in the building to a system of rigid slabs inter-connected by weightless linear springs. The program calculates the centroid, total weight, and the weight moment of inertia about the vertical and two horizontal centroidal axis of each slab. The program also calculates the spring stiffness of concrete walls and its distance from the mass centroid.

Spring constants of shear walls are computed based on the following equation:

$$K = \frac{1}{DT}$$

where:

$$DT = DF + DS$$

and

DF = Flexural deflection/unit load

DS = Shear deflection/unit load.

SSANA was written by Sargent & Lundy.

Hand calculations were used to validate the program. As an example of this validation, stiffness and rotary mass were calculated for elements of the structure shown in Figure F.57-1. The comparison of results from SSANA and hand calculations shown in Tables F.11-1 and F.11-2 demonstrates the accuracy of the program.

F.22 STAND

STAND (Structural Analysis and Design) is an integrated structural code which is programmed to perform analysis and design of structural steel members according to the 1969 AISC Specification. It consists of the following subsystems:

a. beam edit,
b. rolled beam design,
c. composite beam design,
d. plate girder design,
e. column edit,
f. column design, and
g. column base plate design.

The program input consists of member geometry and basic loadings. The design is performed for specified combinations of basic loadings and overstress factors. For floor framing systems, the program is capable of automatic transfer of reactions from tributary beams to supporting members. There are many design control parameters available, such as minimum and maximum depth limitations, shape of the rolled section, location of the lateral support of the compression flange, material grade or yield stress, deflection limitations, flange cutoff criterion, and location of stiffeners.

For columns, the program is capable of accounting for axial loading as well as uniaxial or biaxial bending.

For column base plate design, only axial load and column combinations are considered.

The program output includes the complete final design and provides the designer with sufficient intermediate information to enable him to evaluate the results. For rolled and composite beam designs, complete details of shopwelded and field-bolted end connections are contained in the output. Supplementary information for economic evaluation of the design is also provided.

STAND was developed by Sargent & Lundy. Some of the principal applications include the design of steel floor framing using various types of horizontal structural elements and the design of columns or beam columns.

To validate STAND, results from the program were compared with results from example design problems in the Manual of Steel Construction (Reference 38). Four problems are given.

The first is a rolled beam design problem (Example 1, pp. 2-5). A beam of 36-ksi steel is designed for a 125-kip/ft bending moment, assuming its compression flange is braced at 6-foot intervals. The results, listed in Table F.13-1, show that STAND selects a more efficient section.

The second is a composite beam design problem (Example 1, pp. 2-143 and 2-144). A noncoverplated composite interior floor beam is designed. Limits of 1 1/2 inches for dead load deflection and 1 2/10 inches for live load deflection are imposed. The results, shown in Table F.14-1 are nearly identical.

The third is a column design problem with three examples, (Examples 1, 2 and 5, pp. 3-4, 5, and 9).

The first of these examples is the design of a W12 column of 36-ksi steel that will support a concentric load of 670 kips. The effective length with respect to its minor axis is 16 feet and to its major axis, 31 feet.

The second example is the design of an 11-foot-long W12 interior bay column of 36-ksi steel that will support a concentric load of 540 kips. The column, rigidly framed at the top by 30-foot-long W30 x 116 girders connected to each flange, is braced normal to its web at the top and the base.

The third example is the design of a W14 column of 36-ksi steel for a tier building of 18-foot story height that will support a 600-kip gravity load and a 190-kip/ft maximum wind moment, assuming K = 1 relative to both axes and bending is about the major axis.

The results from all three checks are identical to those in the AISC Manual, and are shown in Table F.15-1.

The fourth problem is a plate girder design problem (Example 1, p. 2-108). A welded plate girder is designed to support a uniform load of 3 kips/ft and two concentrated loads of 70 kips as shown in Figure F.58-1. The compression flange of the girder is supported laterally only at points of concentrated load. The close results are shown in Table F.16-1.

F.23 STRESS

STRESS analyzes plane and space trusses and frames and plane grid by the stiffness matrix method subjected to static loads only.

The structure can be analyzed for arbitrary joint loads, member loads, temperature changes, and joint displacements. Plotting features are available with the program. Output includes joint displacements, equilibrium checks, and reactions and member forces.

STRESS was originally developed at the Massachusetts Institute of Technology and is in the public domain (References 39 and 40). The version used by Sargent & Lundy was adapted by the Chi Corporation, Cleveland, Ohio, (Reference 41).

F.24 STRUDL-II

STRUDL-II (Structural Design Language) is used primarily for static analysis of frame and truss structures. The program is, however, capable of performing linear static or dynamic analyses for finite element representatives of structures using stiffness matrix methods. Nonlinear static problems and stability problems may also be treated.

The program is capable of analyzing plane trusses and frames, grids and elastic bodies, space trusses and frames, or three-dimensional elastic solids subjected to arbitrary loads, temperature changes, or specified displacements. Either earthquake accelerations or time-history force may be used for dynamic analysis. Anisotropic materials may also be used. In addition to analysis, the program is capable of performing structural steel design according to AISC Code and reinforced or prestressed concrete design according to ACI Code.

The program output depends upon the type of finite element used and the analysis that was performed. Included in the output are displacements and member forces and moments or element stresses and moments. Eigenvalues, eigenvectors, and time-history response or nodal response may be obtained for dynamic analyses. Member sizes may be obtained if the design portion is used.

STRUDL-II was developed as part of the Integrated Civil Engineering System at the Massachusetts Institute of Technology (Reference 42).

The program has been in the public domain since 1968. Two versions were used by Sargent & Lundy, one maintained by the McDonnell Douglas Automation Company (Reference 43) and one maintained by UNIVAC (Reference 44).

F.25 TEMCO

TEMCO (Reinforced Concrete Sections Under Eccentric Loads and Thermal Gradients) analyzes reinforced concrete sections subject to separate or combined action of eccentric loads and thermal gradients. The effect of temperature is induced in the section by reactions created by the curvature restraint.

The analysis may be done assuming either a cracked or an uncracked section. Material properties can be assumed to be either linear or nonlinear. The program is capable of handling rectangular as well as nonrectangular sections.

The program input consists of section dimensions, areas and location of each layer of reinforcing steel, loads, load combinations, and material properties.

The curvature and axial strain corresponding to the given eccentric loads (axial load and bending moment) are determined by an iterative procedure. Thermal gradient is applied on the section by inducing reactions created by the curvature restraint, i.e., there is no curvature change due to a thermal gradient on the section. The axial expansion is assumed to be free after thermal gradient is applied. An iterative procedure is employed again for finding the final strain distribution such that equilibrium of internal and external loads is satisfied.

The program output consists of the echo of input, combined loads, final location of neutral axis, final stresses in steel and concrete, and final internal forces. Similar intermediate results (before thermal gradient is applied) can also be output if desired.

The program has applications to a wide variety of reinforced concrete beams and columns, slabs, and containment structures subject to various combinations of external loads and thermal gradients.

This program was developed by Sargent & Lundy.

To demonstrate the validity of TEMCO, program results are compared with hand calculated results. Three example problems are considered. The section and material properties for each problem are given in Table F.17-1, along with the applied external forces and thermal gradients.

The first problem considered involves a section with two layers of steel under the action of a compressive force applied at the centerline of the section, a bending moment, and a thermal gradient. A cracked analysis of the section is required assuming nonlinear material properties.

The second problem considered involves a section with two layers of steel under the action of a tensile force applied at the centerline of the section, a bending moment, and a thermal gradient. A cracked analysis of the section is required assuming nonlinear material properties.

The third problem considered involves a section with two layers of steel under the action of a tensile force applied at the centerline of the section, a bending moment, and a thermal gradient. A cracked analysis of the section is required assuming linear material properties.

The hand-calculated solution was obtained according to the following procedure:

a. Assume the location of neutral axis and the stress distribution to be the same as those given by the program under the given mechanical loading.
b. Compute the strain distribution under the given mechanical loading.
c. Compute the stress resultants by integration and using the proper stress-strain relationships.
d. Check for equilibrium with external mechanical loads.
e. If equilibrium is satisfied, compute the curvature imposed on the section by the given thermal gradient.
f. Compute the final curvature by subtracting the thermal curvature from the mechanical curvature.
g. Compute the new axial strain such that equilibrium is satisfied keeping the curvature constant.
h. Compute the final stress resultants by integration and using the proper stress-strain relationships.
i. Compute the thermal moment.
j. Check for equilibrium and compare program results with hand calculated results.

Results obtained using this procedure together with those computed by TEMCO are presented in Table F.18-1.

It is concluded that results given by the program agree very well with results obtained by hand calculations and that equilibrium between internal and external forces is satisfied for all three problems.

F.26 WARLOC

The basic FLASH-4 program, which uses a generalized multicell, multiflowpath thermal hydraulic transient calculational method, was improved by S&L to include an expanded network with air as a flowing fluid component in addition to a water/steam component and a number of other new features. The S&L modified version of the FLASH-4 program, designated WARLOC, thus has the major physical modeling capability to describe system transients in both boiling water and pressurized water reactors (BWR and PWR), with the associated balance-of-plant compartments. As an option, the blowdown fluid generation model (reactor vessel simulation) can be coupled with or decoupled from the compartment transient analytical model. When this subcompartment transient analysis is performed, the decoupled option is utilized (i.e., the reactor vessel simulation model is not used), and the blowdown data is read in or input into the program.

The basis for the computer program is a network of fluid control volumes (fluid nodes) and flow paths (interconnecting control volumes), for which the conservation equations of mass, momentum, and energy for air-steam-water mixtures are solved in space and time. Superimposed on the network are computer subroutines which permit physical modeling of the reactor system, the containment, plant subcompartments, safeguard fluid systems, and the pipe rupture flow. Flow types ranging from homogeneous to stratified can be specified for each node, with transition from one flow type to another. Nonthermal equilibrium treatment of the nodes enables the nodal atmosphere to have low initial relative humidities and allows steam to coexist with subcooled liquid in the same node, if desired.

Leak flows from ruptured lines are calculated using the Moody two-phase critical flow model with a multiplier of 1.0 or orifice flow model. The vent flow process at a subcritical flow condition is described by the one-dimensional momentum equation for a homogeneous two-phase, two-component flow (air-steam-water mixture) through a flow path connecting two pressure nodes. Time-dependent blowdown mass flow rates with their associated enthalpies can be read in as input to model pipe breaks if this information is known from other blowdown codes.

For the normal flow path, the choked mass flow rate can be calculated either by using Moody's correlation with a multiplier or by using the isentropic expansion method. The actual mass flow rate can be determined by taking the minimum flow rate predicted from any one of the three following comparisons:

a. the inertia flow rate and choked flow rate from Moody's correlation;
b. the inertia flow rate and the choked flow rate from the isentropic expansion method; or
c. the inertia flow rate, the choked flow rate from Moody's correlation, and the choked flow rate from the isentropic expansion method.

The choked mass flow rate using Moody's correlation with a multiplier of .6 is usually more conservative than that obtained using the isentropic expansion method. In the calculation, allowance is made for the presence of liquid droplets in the vent flow. For the air-steam-water mixture system with large pressure differences, isentropic flow of a homogeneous compressible fluid through an orifice or nozzle can be used as one of the choked flow calculations. First the exponent of isentropic expansion, γ is calculated for the mixture of air and water.

$$\gamma = (1\text{-}y_{air})\,\gamma_w + \gamma_{air}\, y_{air}$$

where:

γ_{air} = isentropic exponent for air = 1.4,

γ_w = isentropic exponent,

= 1.035 + 0.1 X for wet steam,

= 1.135 for saturated steam and 1.3 for superheated steam,

x = steam quality, and

y_{air} = air mole fraction in the flowing fluid.

Using this exponent, the critical pressure ratio is evaluated from

$$\frac{P_{sink}}{P_{source}} = \left(\frac{2}{\gamma + 1}\right)^{\gamma / (\gamma-1)}$$

where:

P_{sink} = pressure of downstream node, and

P_{source} = pressure of upstream node.

The actual pressure ratio is compared to this value. If the actual pressure ratio is supercritical, the choked mass flow rate follows

$$w_{choked} = 144 \; C_D \, A \left(\frac{2}{\gamma + 1}\right)^{(\gamma+1)/(2\gamma-2)} \left(g_c \gamma \, P_{sink} \, P_{source}\right)^{1/2}$$

where:

C_D = discharge coefficient (normally set to 1.0 for pipes),

A = cross-sectional area of path, and

P_{source} = density of fluid in upstream node.

For determining the choked flow rate from Moody's correlation for an air-steam-water system, the total nodal pressure and the enthalpy of the mixture corresponding to the mixture density from the steam table were used with an input Moody multiplier.

The program has the option of using the instantaneous phase separation assumption or the homogeneous fine mist assumption for the air-steam-water system.

F.27 PIPSYS

PIPSYS (Piping Analysis Information System) is the primary program used in the analysis of stresses on piping systems. It is a composite of several lesser programs. It combines thermal, weight, and seismic analysis into one program. It may also be broken down into a static analysis portion, a dynamic analysis portion, and a stress combination portion, which is in accordance with ASME Boiler and Pressure Vessel Code, Section III, Articles NB-3000 and NC-3000.

NUPIPE subroutine is the nuclear code part which combines and computes the stresses in piping components according to ASME Boiler and Pressure Vessel Code, Section III, Articles NB-3000 and NC-3000. It was validated by intercomparison with the results of a Teledyne Corporation program for the same sample problem. The final results were essentially identical.

Like NUPIPE, TEE is also an independent program as well as a subroutine of PIPSYS. It also was validated by hand calculation for a problem identical to that processed by the computer.

TEE determines the blowdown force at the discharge pipe due to the sudden opening of a safety/relief valve. It calculates the stress levels at the branch intersection points and valve outlets according to Power Piping Code ANSI B31.10. Finally, it checks the proper discharge pipe sizing at the valve outlet in order to sustain the dynamic force due to the valve blowout.

DYNAPIPE analyzes the dynamic model and upper bound response of a structure (modeled as a space frame) subjected to prescribed dynamic forces at discrete points or subjected to earthquake type motion prescribed as acceleration response spectra. Elbows and tees are treated per piping code.

F.28 PENAN

PENAN handles the analysis of axially symmetric solids of revolution, which are composed of orthotropic materials (with temperature-dependent properties), and which are subjected to asymmetric and time-dependent heating and loading.

Mainly, the structure of this program is a combination of two, suitably modified, axisymmetric finite element programs. The two programs are:

a. NOHEAT (Nonlinear Heat Transfer Analysis Program), by I. Farhoomand and E. Wilson (Structural Engineering Laboratory, University of California, Berkeley, California. Report Number UC SESEM 71-6, April 1971), and
b. ASAL (Finite Element Analysis-Axisymmetric Solids with Arbitrary Loads), by R. S. Dunham and R. E. Nickell (Structural Engineering Laboratory, University of California, Berkeley, California, Report Number 67-6, June 1967).

Outlined below is a brief description of the program's most significant features.

a. PENAN is designed to handle automatic finite element mesh generation and plotting for various penetration assembly configurations.
b. It has a built-in material property bank covering temperature-dependent mechanical and thermal properties (including fatigue design curves) for all Section III materials.
c. It forms optimal loads and load-range combinations for the various Code-specified loading categories and generates Fourier series coefficients for all asymmetrically applied loads.
d. Through repeated use of the same stiffness matrix and unit-load stresses, the program carries out all required multiple stress evaluations.

e. PENAN calculates the allowable stresses for all stress and loading categories, makes the necessary stress comparisons, and generates the entire Penetration Assembly Stress Analysis Report.

F.29 FAST

FAST is used primarily for the dynamic analysis of linear axisymmetric structures using transfer functions (modeled as a multi-degree-of-freedom system). The structural model is given as input either in the form of its eigenvalues, eigenvectors, participation factors and modal damping ratios, or in the form of its responses for a typical dynamic load. This program is used to obtain the time history, transfer function and response spectra for various response components of the model. The results are given in print and plot forms.

FAST was developed at Sargent & Lundy in 1975.

F.30 STAAD PRO

STAAD PRO V8i (2007 - Structural Analysis and Design Software) performs static and dynamic analysis of two and three-dimensional trusses and frames, plane elastic bodies, and plate and shell structures. The finite element approach is used with standard linear or beam elements, a plane stress triangular element, or a triangular plate bending element. The STAAD PRO V8i (2007) program accepts dynamic, thermal, pressure, gravity, as well as concentrated loads.

The program output includes joint displacements, beam forces, and shell plate element stresses and moments.

The program also has an available AISC structural steel input member database and can perform AISC code compliance checks.

STAAD PRO was developed by Bentley Systems, Incorporated, 685 Stockton Drive Exton, PA 19341, United States. It was used by URS Corporation.

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TABLE F.1-1

(Sheet 1 of 1)

RESULTS FOR 28-DAY STRENGTH

RESULTS FOR 28-DAY STRENGTH

Number of Samples Collected - - -	46				
Mean Observed Strength - - - - - -	3456.1	Allow. Design Str. - - - - - - -	2955.1	Spec. Design Str. - -	2500.0
Observed Standard Deviation - - -	373.0	C.O.V. % - - - - -	10.8	Expected C.O.V. % -	15.0
Within Test Std. Deviation - - - - -	.0	C.O.V. % - - - - -	.0	Expected C.O.V. % -	5.0
Mean Observed Range - - - - - - - -	.0				

Total Number of Bad Samples – Average Strength	0
Number of Times Inefficient Testing Observed	0

CONTROL ACCORDING TO ACI MANUAL

Number of Samples Falling Below FC - - - - - - - - - - - -	0	Percent of Samples Collected - - - -	.00
Number of Samples Falling Below FC-500 - - - - - - - -	0	Percent of Samples Collected - - - -	.00
Number of Times Moving Avg Fell Below FC - - - - - -	0	Percent of Samples Collected - - - -	.00

TABLE F.2-1

NATURAL PERIODS FOR THE EIGHT LOWEST FLEXURAL MODES

MODE NUMBER	PERIODS (seconds) SAP-IV	DAPS-II
1	525.79	526.54
2	85.368	86.213
3	30.965	31.658
4	16.059	16.676
5	9.9006	10.439
6	6.8276	7.2634
7	5.1865	5.4680
8	4.3777	4.4676

TABLE F.3-1

STRUCTURAL FREQUENCIES

MODE NUMBER	STRUCTURAL FREQUENCY (cps) BIGGS	DYNAS
1	1.00	1.00
2	2.18	2.18
3	3.18	3.18

TABLE F.4-1

PROBABLE MAXIMUM STORY DISPLACEMENTS

MODE NUMBER	PROBABLE MAXIMUM STORY DISPLACEMENT (inches)	
	BIGGS	DYNAS
1	1.50	1.51
2	3.22	3.20
3	4.86	4.68

TABLE F.5-1

ABSOLUTE MAXIMUM STORY SHEARS

MODE NUMBER	ABSOLUTE MAXIMUM STORY SHEAR (kips) BIGGS	DYNAS
1	3020	3010
2	2080	2068
3	1345	1353

TABLE F.6-1

PROBABLE MAXIMUM STORY SHEARS

MODE NUMBER	PROBABLE MAXIMUM STORY SHEAR (kips)	
	BIGGS	DYNAS
1	2250	2262
2	1740	1757
3	895	902

TABLE F.7-1

NATURAL PERIODS FOR THE EIGHT LOWEST

FLEXURAL MODES

MODE NUMBER	PERIODS (seconds) SAPIV	DYNAS
1	525.79	525.69
2	85.368	85.369
3	30.965	30.964
4	16.059	16.060
5	9.9006	9.9010
6	6.8276	6.8279
7	5.1865	5.1866
8	4.3777	4.3778

TABLE F.8-1

(Sheet 1 of 5)

INTERACTION DIAGRAM AXIAL LOAD VS. BENDING MOMENT REFERRED TO THE PLASTIC CENTROID OF THE SECTION.

LIST OF SYMBOLS

B = WIDTH OF SECTION, IN.
T = HEIGHT OF SECTION, IN.
D = DEPTH OF TENSILE STEEL, IN.
AS = AREA OF TENSILE STEEL, SQ IN.
DC = DEPTH OF COMPRESSION STEEL, IN.
ASC = AREA OF COMPRESSION STEEL, SQ IN.
ES = MODULUS OF ELASTICITY OF REINFORCING STEEL, KSI
SSY = YIELD STRESS OF REINFORCING STEEL, KSI.
SSU = ULTIMATE STRESS OF REINFORCING STEEL, KSI.
PRESTR= INITIAL PRESTRAIN OF REINFORCING STEEL
ULTSTR= ULTIMATE STRAIN OF REINFORCING STEEL
CUS = 28 DAY STRENGTH OF CONCRETE CYLINDER, KSI.
EPSZ = CONCRETE STRAIN FOR MAXIMUM STRESS
EPSU = CONCRETE STRAIN AT CRUSHING
NSIESS = NUMBER OF POINTS IN INTERACTION DIAGRAM
EPSL = MAXIMUM TOP STRAIN FOR WHICH ID. IS COMPUTED

INPUT DATA

B= 12.000 T= 48.000 D= 45.000 AS= 2.7500 DC= 10.000 ASC= 1.2500 ES= 29000.0000 SSY= 80.0000 SSU= 90.0000
ULTSTR= .020000 PRESTR= .000000 CUS= 4.5000 EPSZ= .002000 EPSU= .004000 NSIESS= 20 EPSL= .003000

RESULTS GIVEN IN THE FOLLOWING ORDER

COUNTER	CURVATURE	TOP STRAIN	AXIAL LOAD (KIPS)	BENDING MOMENT (FT.-KIP)	AXIAL LOAD DIMENSIONLESS	BENDING MOMENT DIMENSIONLESS	C.R.FACTOR PHI	REDUCED AXIAL LOAD (KIPS)	REDUCED BENDING MOMENT (FT.-KIP)
INITIAL POINT UNDER UNIFORM COMPRESSION STRAIN – EPSZ									
1	.00000000	.00200000	2419.8999	6.0875	1.0984	.0007	.8981	2173.3915	5.4674
2	.00001562	.00225000	2349.4036	96.0163	1.0654	.0109	.8982	2110.2042	86.2406
3	.00003125	.00250000	2216.4314	258.0601	1.0060	.0293	.8983	1990.9977	231.8128
4	.00004687	.00275000	2023.3391	495.1479	.9184	.0562	.8984	1817.8463	444.8601
5	.00006250	.00300000	1770.1265	807.2801	.8034	.0916	.8986	1590.6962	725.4495
PRECEDING POINT HAD BOTTOM FIBER STRAIN = ZERO									
6	.00007316	.00300000	1377.5877	1191.7730	.6253	.1352	.8989	1238.3646	1071.3289
7	.00008966	.00300000	1163.2063	1325.1266	.5280	.1504	.8991	1045.8417	1191.4246
8	.00010115	.00300000	985.3649	1400.5915	.4472	.1589	.8992	886.0792	1259.4674
9	.00011264	.00300000	833.6598	1447.7928	.3784	.1643	.8994	749.7575	1302.0822
BALANCED POINT. TENSILE STEEL STRAIN = - EPSY									
10	.00016245	.00300000	518.9094	1288.7116	.2355	.1462	.8996	466.8107	1159.3245
11	.00021226	.00300000	339.3652	1137.9047	.1540	.1291	.8997	305.3398	1023.8163
12	.00026207	.00300000	218.2740	1021.0622	.0991	.1159	.8998	196.4099	918.7840
13	.00031188	.00300000	127.3296	930.6383	.0573	.1056	.8999	114.5842	837.4830
14	.00036169	.00300000	52.4308	857.1947	.0238	.0973	.9000	47.1856	771.4405
15	.00041149	.00300000	-11.2036	797.4642	-.0051	.0905	.9000	-10.0834	717.7247
16	.00046130	.00300000	-67.2224	747.8149	-.0305	.0849	.9001	-60.5036	673.0722
PRESENT POINT HAD TENSILE STEEL STRAIN = - ULTSTR									
17	.00044047	.00200001	-174.2335	571.9126	-.0791	.0649	.9001	-156.8336	514.7982
18	.00041964	.00100000	-269.5452	392.8982	-.1223	.0446	.9002	-242.6467	353.6901
19	.00039880	.00000000	-317.0636	299.7994	-.1439	1.6329	.9002	-285.4348	269.8928
20	.00000000	-.02000000	-360.0000	273.9367	-.1634	.0311	.9003	-380.0000	273.9367

DISTANCE OF PLASTIC CENTROID TO BOTTOM FIBER.IN = 24.9313

TABLE F.8-1

(Sheet 2 of 5)

AS AND ASC CHANGED VALUES, READ MOMENT WITH OPPOSITE SIGN

INPUT DATA

B= 12.000	T= 48.000	D= 45.0000	AS= 1.2500	DC= 10.0000	ASC= 2.7500	ES= 29000.0000	SSY= 60.0000	SSU= 90.0000
ULTSTR= .020000	PRESTR= .000000	CUS= 4.5000	EPSZ= .002000	EPSU= .004000	NSIESS= 20	EPSL= .003000		

RESULTS GIVEN IN THE FOLLOWING ORDER

COUNTER	CURVATURE	TOP STRAIN	AXIAL LOAD (KIPS)	BENDING MOMENT (FT.-KIP)	AXIAL LOAD DIMENSIONLESS	BENDING MOMENT DIMENSIONLESS	C.R.FACTOR PHI	REDUCED AXIAL LOAD (KIPS)	REDUCED BENDING MOMENT (FT.-KIP)
INITIAL POINT UNDER UNIFORM COMPRESSION STRAIN + EPSZ									
1	.00000000	.00200000	2419.8999	-1.8527	1.0984	-.0002	.8981	2173.3915	-1.6640
2	.00001562	.00225000	2371.9226	84.3211	1.0766	.0073	.8982	2130.3893	57.7713
3	.00003125	.00250000	2258.0534	206.7475	1.0249	.0235	.8983	2029.3136	185.7126
4	.00004687	.00275000	2083.4750	431.3483	.9457	.0489	.8984	1871.7781	387.5200
5	.00006250	.00300000	1843.1875	738.1239	.8389	.0838	.8986	1660.7331	663.2583
PRECEDING POINT HAD BOTTOM FIBER STRAIN = ZERO									
6	.00007816	.00300000	1484.8191	1109.2716	.6739	.1259	.8989	1334.6301	997.0735
7	.00008966	.00300000	1292.5999	1224.0333	.5887	.1389	.8990	1162.0507	1100.4092
8	.00010115	.00300000	1133.6276	1273.4436	.5145	.1445	.8991	1019.2732	1144.9854
9	.00011264	.00300000	999.4452	1290.4057	.4536	.1464	.8992	898.7299	1160.3700
BALANCED POINT. TENSILE STEEL STRAIN = - EPSY									
10	.00016245	.00300000	669.1896	1129.0560	.3037	.1281	.8995	601.9351	1015.5674
11	.00021226	.00300000	474.8520	963.1077	.2155	.1093	.8996	427.1928	866.4440
12	.00026207	.00300000	339.6791	826.0203	.1542	.0937	.8997	305.6222	743.2018
13	.00031188	.00300000	234.6630	712.3114	.1065	.0808	.8998	211.1542	640.9513
14	.00036169	.00300000	143.7225	611.7068	.0652	.0694	.8999	129.3743	550.4683
15	.00041149	.00300000	64.0464	523.6752	.0291	.0594	.9000	57.8386	471.2818
16	.00046130	.00300000	-8.0140	444.9537	-.0036	.0505	.9000	-7.2127	400.4611
PRECEDING POINT HAD TENSILE STEEL STRAIN = - ULTSTR									
17	.00044047	.00200001	-135.5442	255.9720	-.0615	.0280	.9001	-122.0039	230.4015
18	.00041964	.00100000	-232.6858	84.0563	-.1056	.0095	.9002	-209.4590	75.6658
19	.00039980	.00000000	-282.0341	-6.7816	-.1280	-.0369	.9002	-253.8921	-6.1049
20	.00000000	-.02000000	-360.0000	-83.3721	-.1634	.0095	.9003	-360.0000	-83.3721

DISTANCE OF PLASTIC CENTROID TO BOTTOM FIBER.IN = 23.7166

TABLE F.8-1

(Sheet 3 of 5)

INTEPACTION DIAGRAM --- P VS. M ABOUT C. G. OF UNCRACKED TRNSFORMED SECTION

YIELD-STRENGTH THEORY

WIDTH OF SECTION (IN.)	=	12.000	AREA OF TENSILE STEEL (IN.)	=	2.750
HEIGHT OF SECTION (IN.)	=	48.000	DEPTH OF TENSILE STEEL (IN.)	=	45.000
ELASTIC MODULUS, STEEL (KSI)	=	29000.	AREA OF COMPRESSIVE STEEL (IN.)	=	1.250
ELASSTIC MODULUS, CONCRETE (KSI)	=	3865.	DEPTH OF COMPRESSIVE STEEL (IN.)	=	10.000

28-DAY STRENGTH OF CONCRETE CYLINDER (KSI)	=	4.500
YIELD STRESS FOR REINFORCING STEEL (KSI)	=	54.000
DEPTH OF C.G. OF UNCRACKED TRANSFORMED SECTION (IN.)	=	24.435
MODULAR RATIO	=	8.000
MAXIMUM STRESS OF CONCRETE (KSI)	=	3.825
MAXIMUM STRESS OF REINFORCING STEEL (KSI)	=	48.600
UNIT WEIGHT OF CONCRETE (LS./CU. FT.)	=	145.000

PHI1 IS THE CAPACITY REDUCTION FACTOR

POSITION NO.	AXIAL LOAD (KIPS)	BENDING MOMENT (FT.-KIP)	PHI1	REDUCED AXIAL LOAD (KIPS)	REDUCED BENDING MOMENT (FT.-KIP)
1	2302.69	.00	.8992	2068.33	.00
2	1130.43	796.74	.8991	1016.45	716.37
3	1056.93	841.20	.8992	950.38	756.39
4	262.26	846.88	.8998	235.98	752.02
5	-147.15	212.81	.9001	-132.45	191.55
6	-194.40	155.97	.9001	-174.99	140.40
7	-71.30	-54.99	.9001	-64.18	-49.50
8	-51.38	-90.70	.9000	-46.24	-81.63
9	316.89	-674.33	.8998	285.12	-606.73
10	940.50	-909.37	.8993	845.77	-817.77
11	1172.20	-796.61	.8991	1053.92	-716.23
12	2302.68	.00	.8982	2068.33	.00

TABLE F.8-1

(Sheet 4 of 5)

INTERACTION DIAGRAM --- P VS. M ABOUT C.G. OF UNCRACKED TRANSFORMED SECTION

WORKING-STRESS DESIGN METHOD

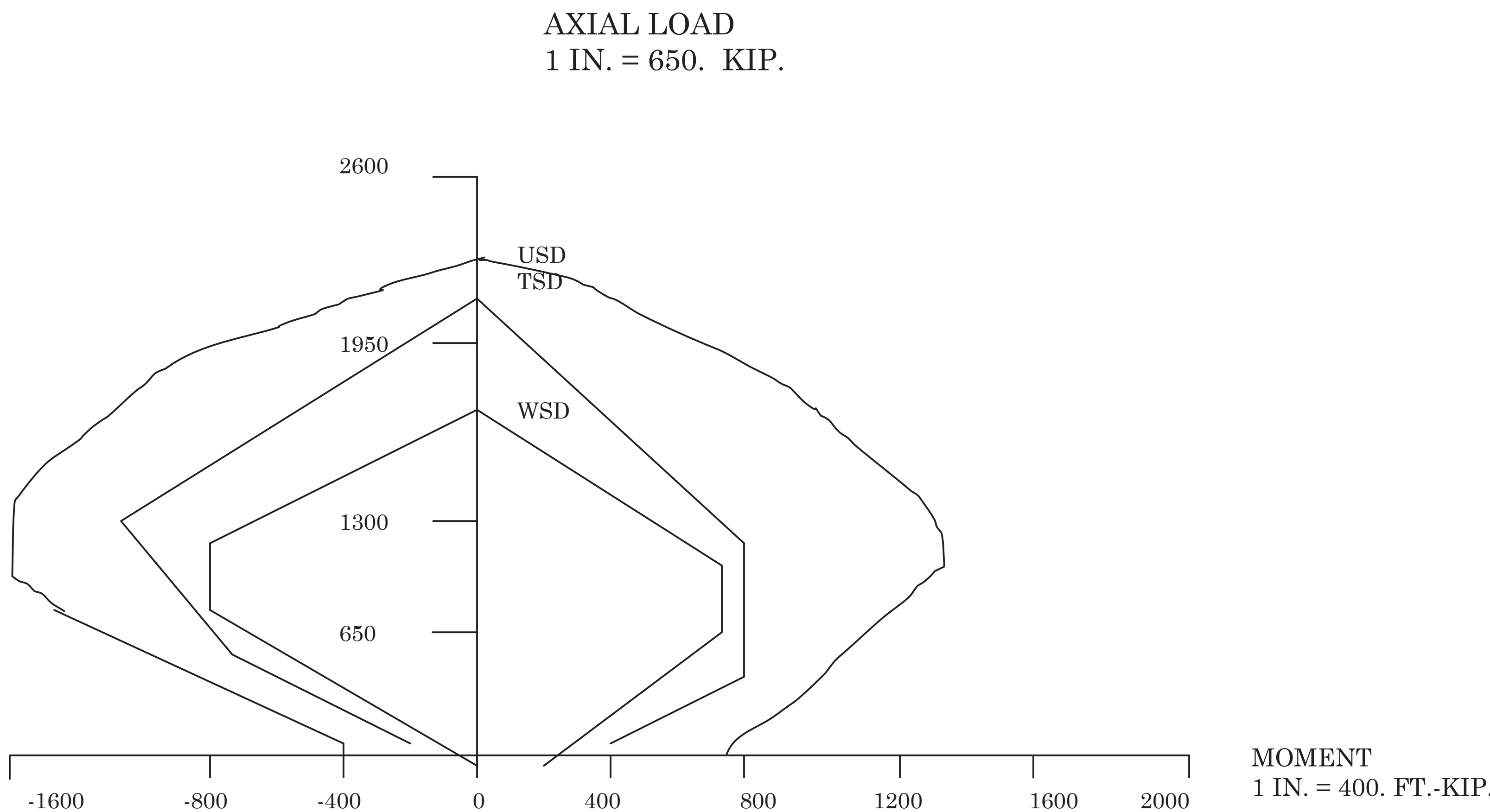
WIDTH OF SECTION (IN.)	=	12.000	AREA OF TENSILE STEEL (IN.)	=	2.750
HEIGHT OF SECTION (IN.)	=	48.000	DEPTH OF TENSILE STEEL (IN.)	=	45.000
ELASTIC MODULUS STEEL (SKI)	=	29000.	AREA OF COMPRESSIVE STEEL (IN.)	=	1.250
ELASTIC MODULUS CONCRETE (KSI)	=	3865.	DEPTH OF COMPRESSIVE STEEL (IN.)	=	10.000

ALLOWABLE STRESS OF CONCRETE IN BENDING (KSI)	=	2.700
ALLOWABLE STRESS IN REINFORCING STEEL (KSI)	=	20.000

POSITION NO.	AXIAL LOAD (KIPS)	BENDING MOMENT (FT.-KIP)
1	1717.20	.00
2	824.64	618.32
3	768.37	653.56
4	340.10	659.09
5	-60.56	94.98
6	-80.00	60.75
7	-29.34	-23.89
8	-21.14	-38.23
9	389.08	-627.77
10	718.18	-704.17
11	892.56	-618.00
12	1717.20	.00

TABLE F.8-1

(Sheet 5 of 5)



INTERACTION DIAGRAM – AXIAL LOAD VS. BENDING MOMENT.

TEST 4 48 X 12 STEEL T=2.75/40. C=1.25/10. FCP=4.5 FP=60.

TABLE F.9-1

COMPUTER OUTPUT

DISPLACEMENTS

LOCATION	VALUE (inches)
U_1	.059492
U_2	.045083
U_3	.033292
U_4	.023913
U_5	.016642
U_6	.011246
U_7	.007491
U_8	.004830
U_9	.002874
U_{10}	.001338
f_1	16.293 kips

Post-Buckling Load

P = 21.978 kips

TABLE F.10-1

COMPUTER OUTPUT

ANCHOR FORCES

LOCATION	VALUE (kips)
f_1	16.270
f_2	15.430
f_3	14.149
f_4	12.338
f_5	10.935
f_6	9.531
f_7	6.348
f_8	4.093
f_9	2.436
f_{10}	1.134

TABLE F.11-1

STIFFNESS

ELEMENT	STIFFNESS PROGRAM	HAND CALCULATIONS (kip/ft.)
1	398821	398880
2	398821	398880
3	398821	398880
4	398821	398880

TABLE F.12-1

WEIGHT MOMENT OF INERTIA ABOUT X-AXIS (Ip)

ELEMENT	WEIGHT INERTIA PROGRAM	HAND CALCULATIONS (ip)
1	2005	2005
2	531	531.25
3	531	531.25
4	32	31.5

TABLE F.13-1

ROLLED BEAM DESIGN PROBLEM

	MAXIMUM MOMENTS (kip-ft.)	SECTION SELECTED	SECTION MODULUS (in^3)
AISC	125	W16x40	64.6
STAND	125.58	W18x40	68.4

TABLE F.14-1

COMPOSITE BEAM DESIGN PROBLEM

	BENDING MOMENTS (kip-ft.)				
	CONSTRUCTION LOAD	DESIGN LOAD	MAXIMUM SHEAR (kips)	STEEL SECTION	NUMBER OF SHEAR CONNECTORS
AISC.	71.3	237.2	26.4	W21x44	42
STAND	71.3	236.5	26.3	W21x44	42

TABLE F.15-1

COLUMN DESIGN PROBLEM

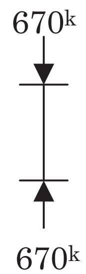
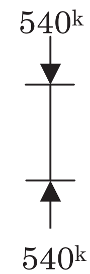
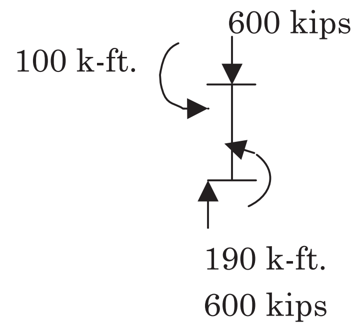
ITEMS	AISC EXAMPLE 1	AISC EXAMPLE 2	AISC EXAMPLE 5
Column Design Parameters			
AISC SOLUTION	W12x161	W12x99	W14x142
STAND SOLUTION	W12x161	W12x99	W14x142

TABLE F.16-1

PLATE GIRDER DESIGN PROBLEM

RESULTS	AISC	STAND
Maximum Bending Moment (kip-ft.)	2054	2045
Maximum Vertical Shear (kips)	142	141.3
Web Section	1 plate, 70x5/16	1 plate, 70x5/16
Flange Section	2 plates, 18x3/4	2 plates, 18x3/4
Stiffener End Spacing (ft.)	3.5	3.56
Stiffener Intermediate Spacing (ft.)	6.75	6.72
Area of * Stiffeners Furnished (in^2)	2.0	1.88

*required area is 1.78 in^2

TABLE F.17-1

SECTION AND MATERIAL PROPERTIES FOR PROBLEMS

SECTION AND MATERIAL PROPERTIES	PROBLEM NUMBER 1	2	3
Thickenss (in.)	42.0	30.0	42.0
Width (in.)	12.0	12.0	12.0
Area of 1st steel layer (in^2)	6.25	2.25	3.12
Distance of 1st steel layer (in.)	3.0	3.0	3.0
Area of 2^{nd} steel layer (in^2)	6.25	4.0	3.12
Distance of 2^{nd} steel layer (in.)	37.0	25.0	37.0
Concrete unit weight (lb/ft^3)	150.0	150.0	150.0
Concrete compressive strength (lb/in^2)	4000.0	4000.0	4000.0
Concrete coef. of thermal expansion (in/°F)	5.56×10^{-6}	5.56×10^{-6}	5.56×10^{-6}
Steel yield strength ($kips/in^2$)	45.0	45.0	45.0
Steel modulus of elasticity ($kips/in^2$)	29000.0	29000.0	29000.0
Material properties	Nonlinear	Nonlinear	Linear
Applied axial force (kips)	-38.25	76.53	34.65
Applied bending moment (ft-kips)	129.75	-9.49	206.25
Inside temperature (°F)	82.50	67.50	247.50
Outside temperature (°F)	52.50	0.0	115.50

TABLE F.18-1

RESULTS OF PROBLEMS

RESULTS	PROBLEM NUMBER 1	2	3
Equilibrating axial force given by TEMCO program (kips)	-38.25	76.53	34.65
Equilibrating axial force computed by hand (kips)	-38.253	76.53	34.65
Equilibrating bending moment given by TEMCO program (ft-kips)	129.75	-9.49	206.25
Equilibrating bending moment computed by hand (ft-kips)	129.752	-9.493	206.25
Thermal moment given by TEMCO program (ft-kips)	-54.58	-21.07	-137.75
Thermal moment computed by hand (ft-kips)	-54.585	-21.071	-137.757

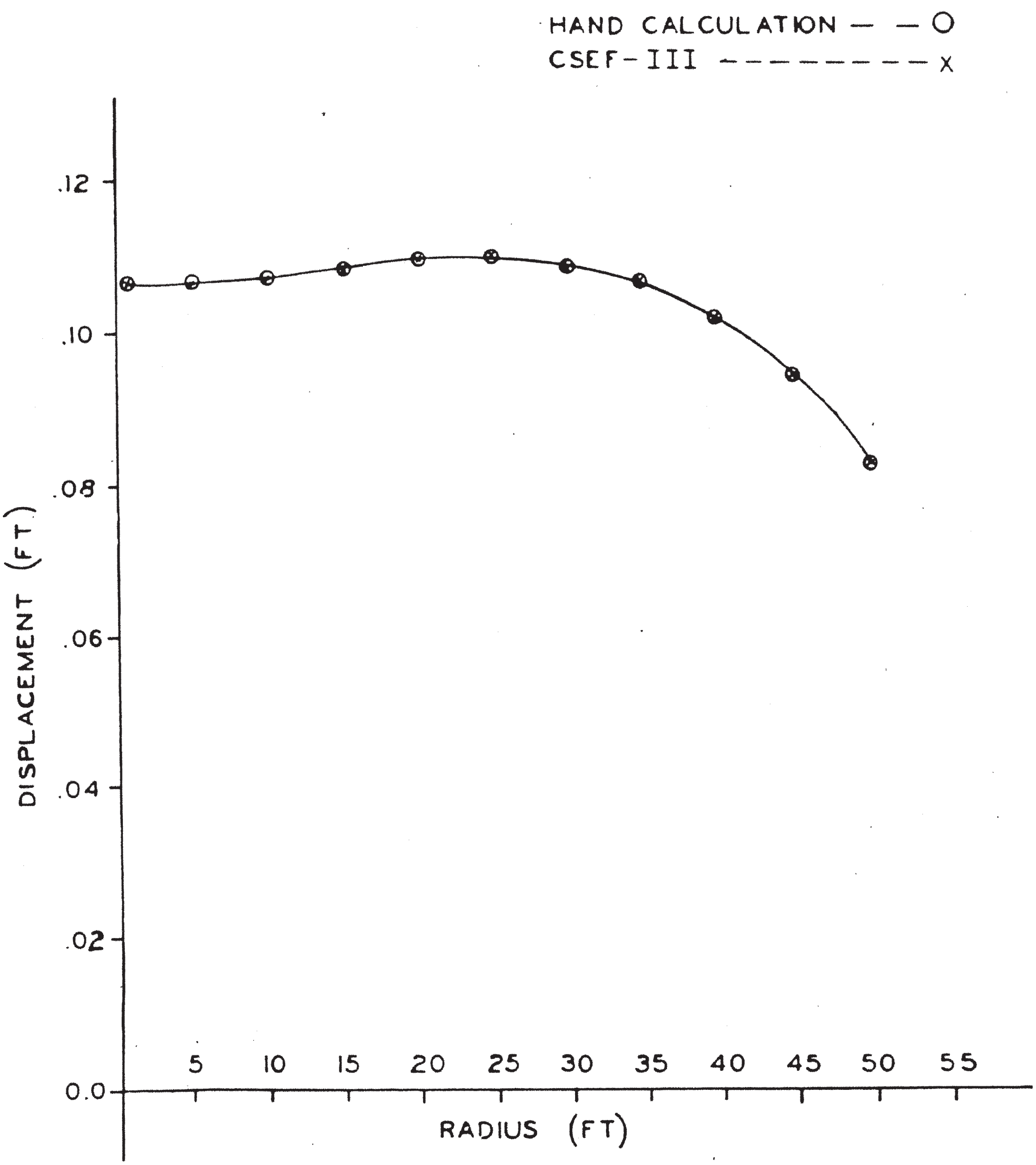


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FIGURE F.1-1

UNITS 1 & 2

DEFLECTION OF A CIRCULAR PLATE DUE TO UNIFORM PRESSURE AND AXISYMMETRIC EDGE MOMENT

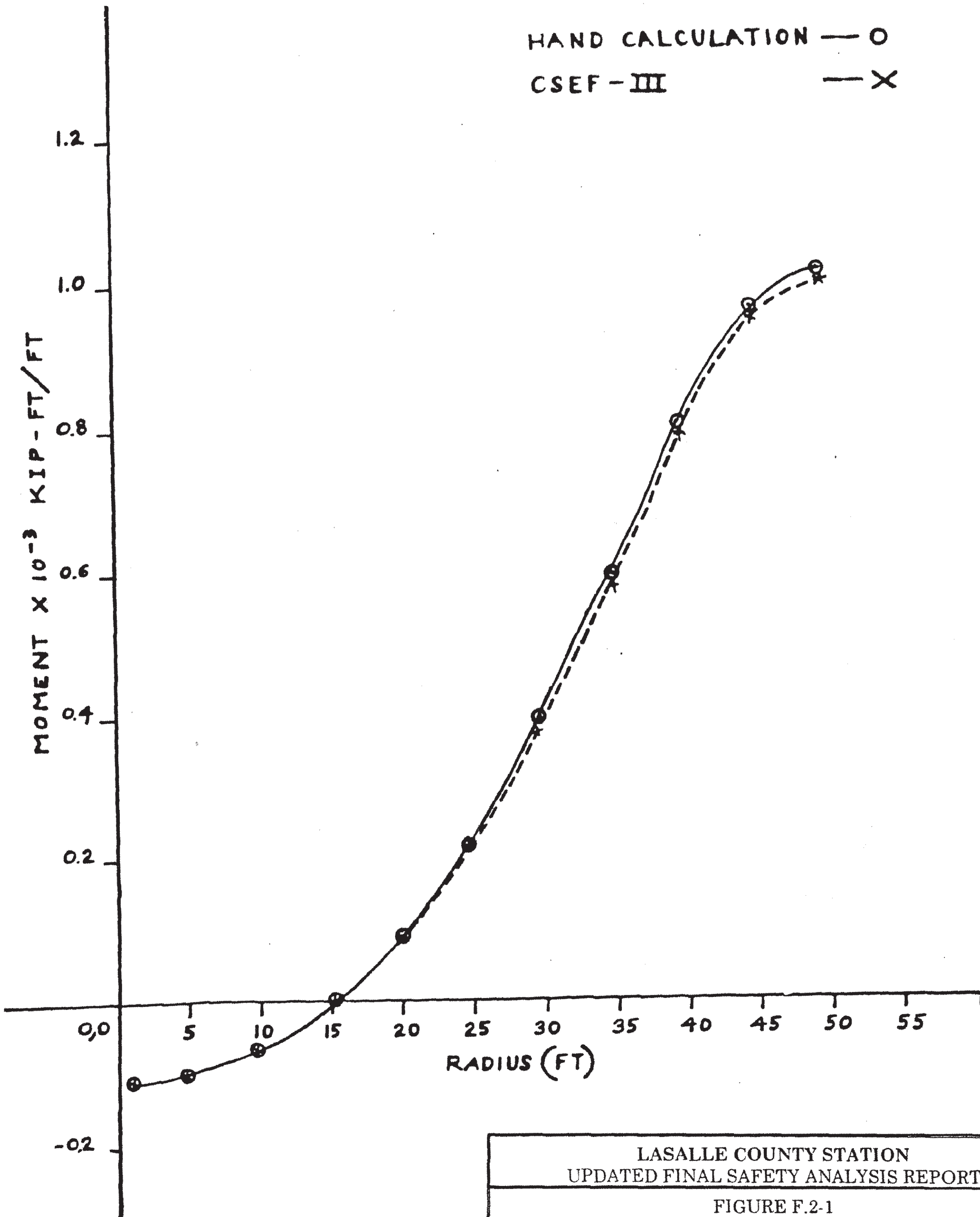

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FIGURE F.2-1

UNITS 1 & 2

RADIAL MOMENT DUE TO UNIFORM PRESSURE AND AXISYMMETRIC EDGE MOMENT

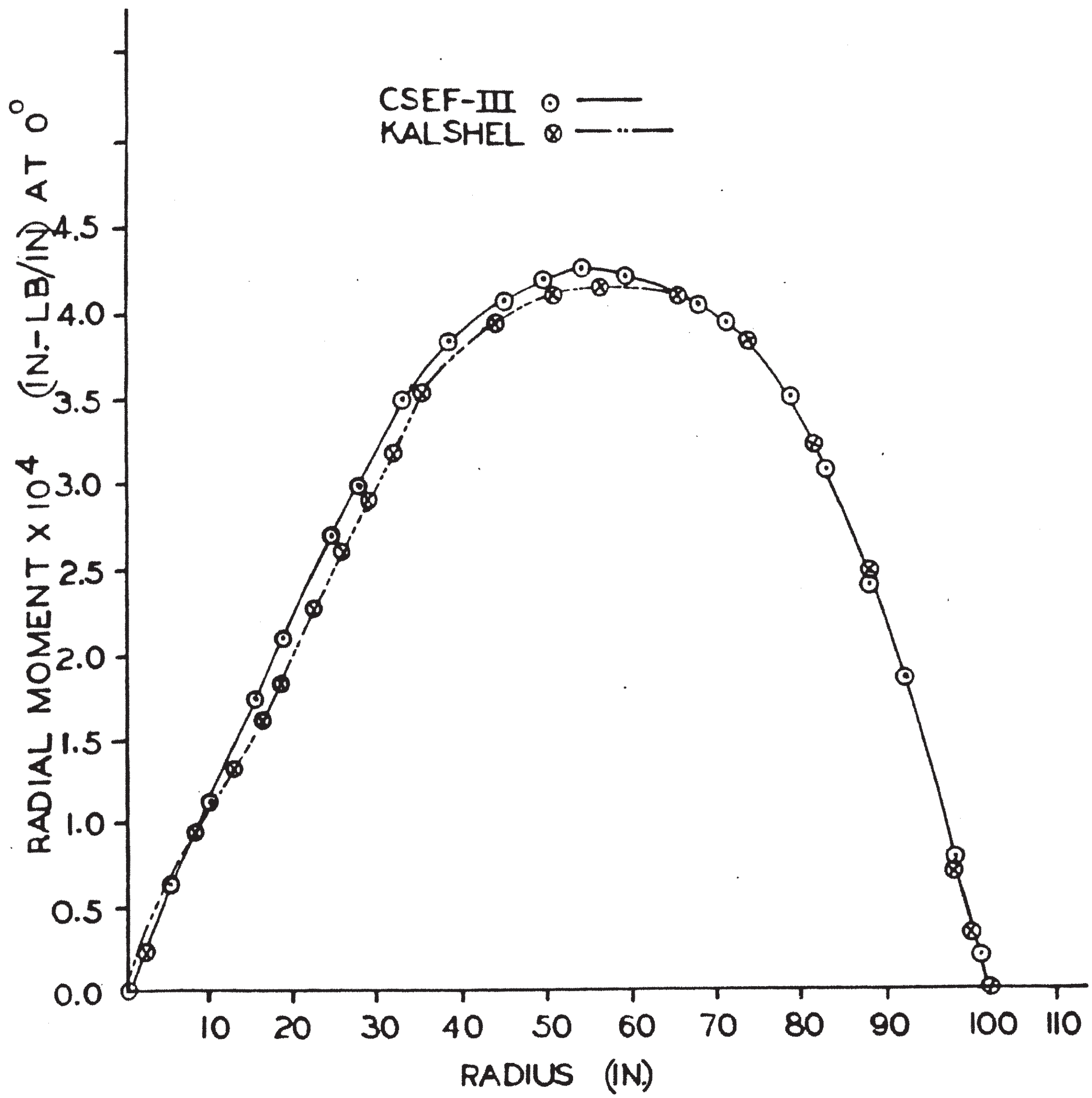

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FIGURE F.3-1

UNITS 1 & 2

RADIAL MOMENT – SIMPLY SUPPORTED CIRCULAR PLATE, LINEARLY VARYING PRESSURE LOAD

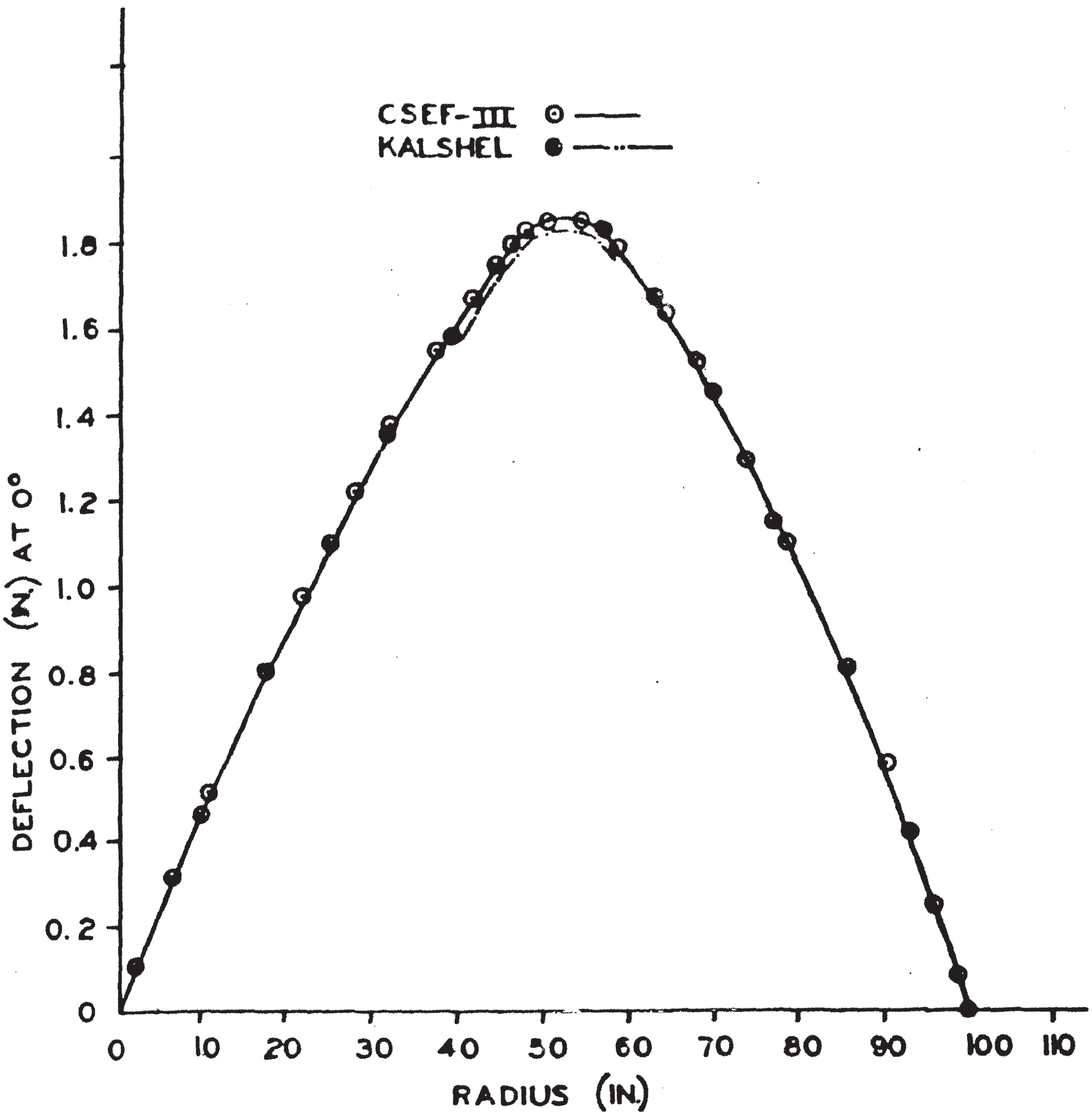

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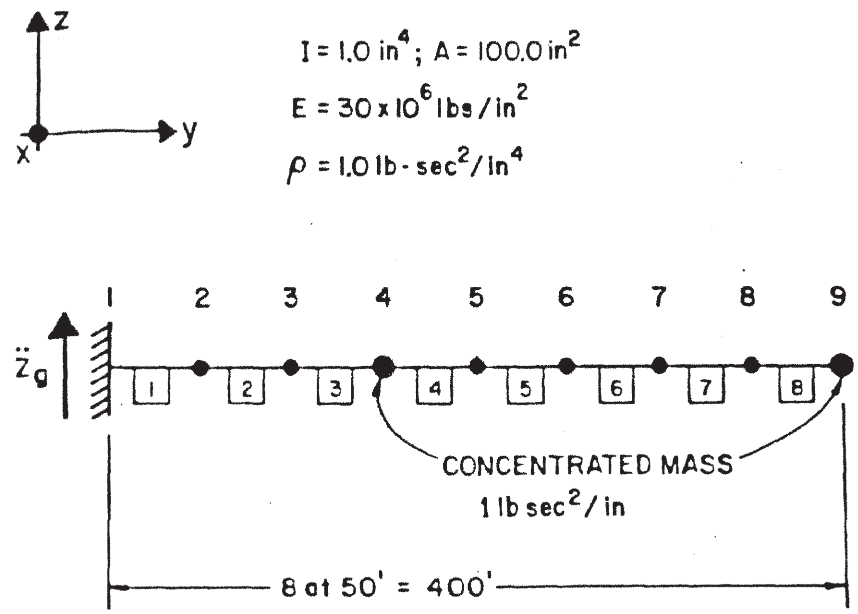
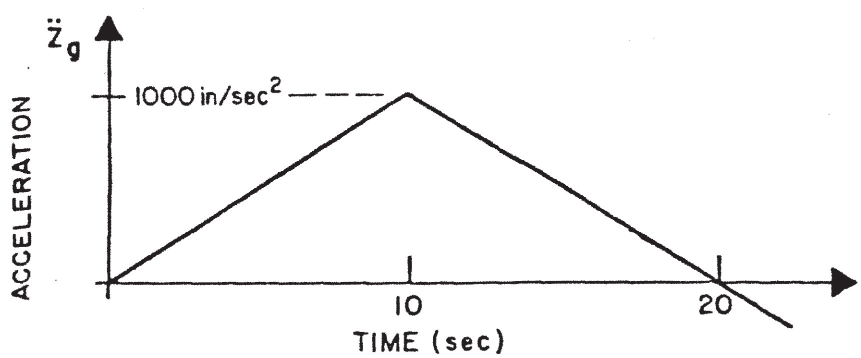
FIGURE F.4-1 UNITS 1 & 2 DEFLECTION – SIMPLY SUPPORTED CIRCULAR PLATE, LINEARLY VARYING PRESSURE LOAD



(a) NODE AND BEAM NUMBER ASSIGNMENTS FOR THE CANTILEVER MODEL



(b) GROUND ACCELERATION APPLIED AT NODE 1

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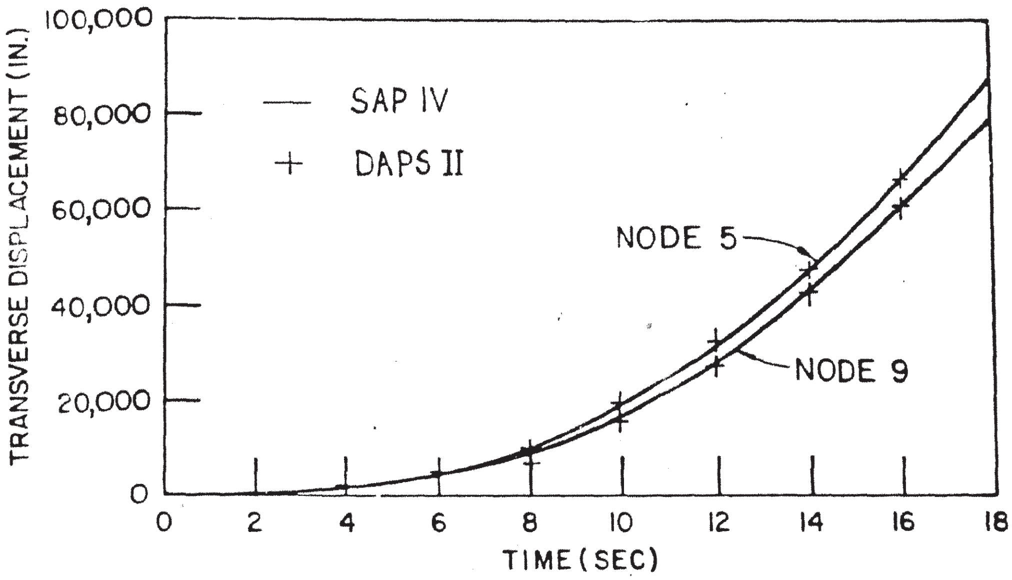
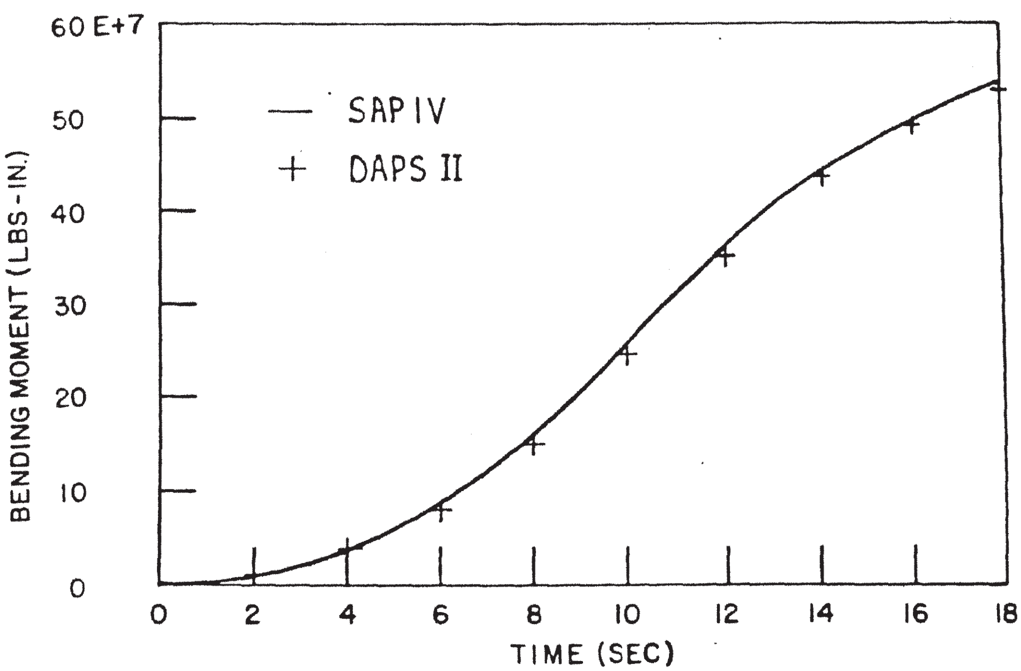
FIGURE F.5-1

UNITS 1 & 2

RESPONSE HISTORY ANALYSIS OF CANTILEVER BEAM FOR DAPS-II



(a) TRANSVERSE DEFLECTIONS



(b) MOMENT AT NODE 1
(FIXED END OF CANTILEVER)

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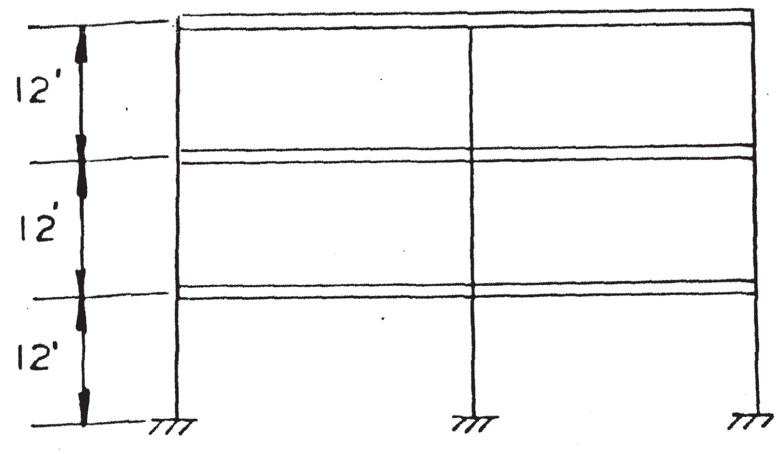
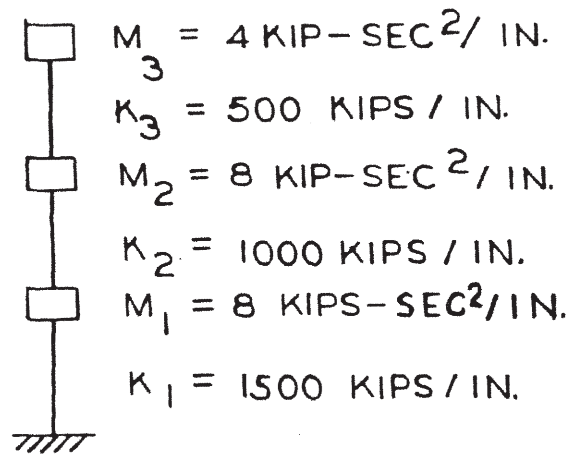
FIGURE F.6-1 UNITS 1 & 2 CANTILEVER RESPONSE



(A)



(B)

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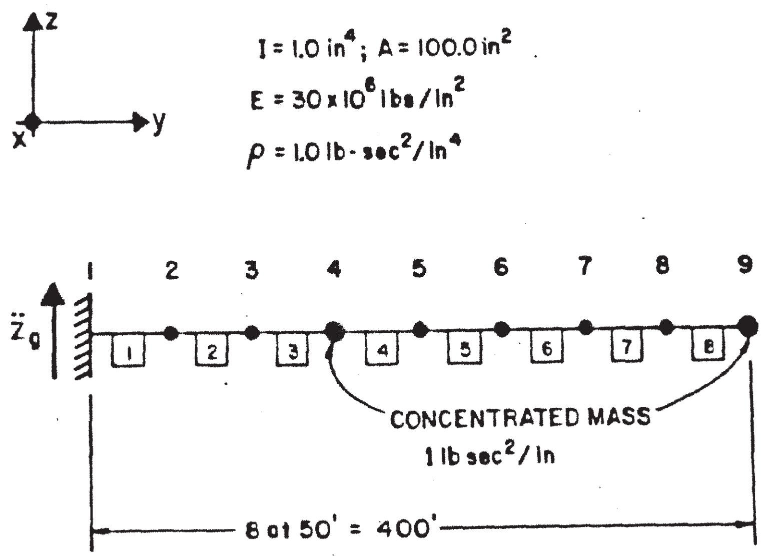
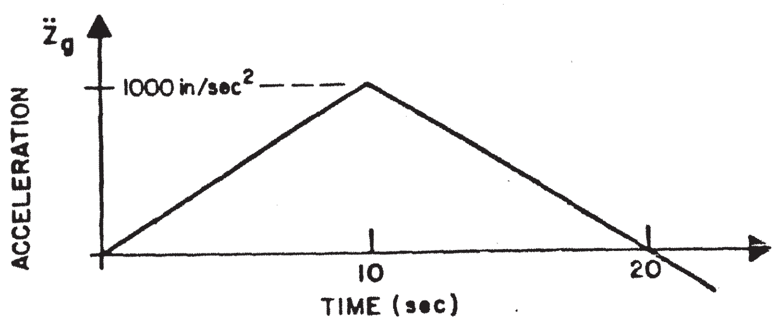
FIGURE F.7-1 UNITS 1 & 2 THREE-STORY SHEAR BUILDING



(a) NODE AND BEAM NUMBER ASSIGNMENTS FOR THE CANTILEVER MODEL



(b) GROUND ACCELERATION APPLIED AT NODE 1

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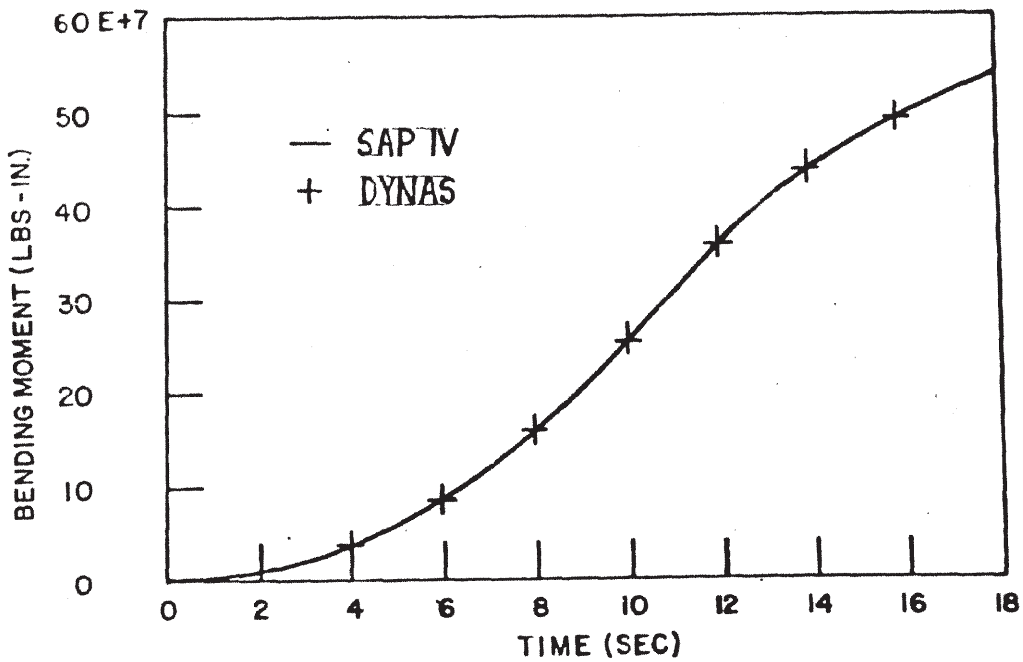
FIGURE F.8-1 UNITS 1 & 2 RESPONSE HISTORY ANALYSIS OF CANTILEVER BEAM FOR DYNAS PROGRAM



MOMENT AT NODE 1
(FIXED END OF CANTILEVER)

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FIGURE F.9-1 UNITS 1 & 2 CANTILEVER RESPONSE

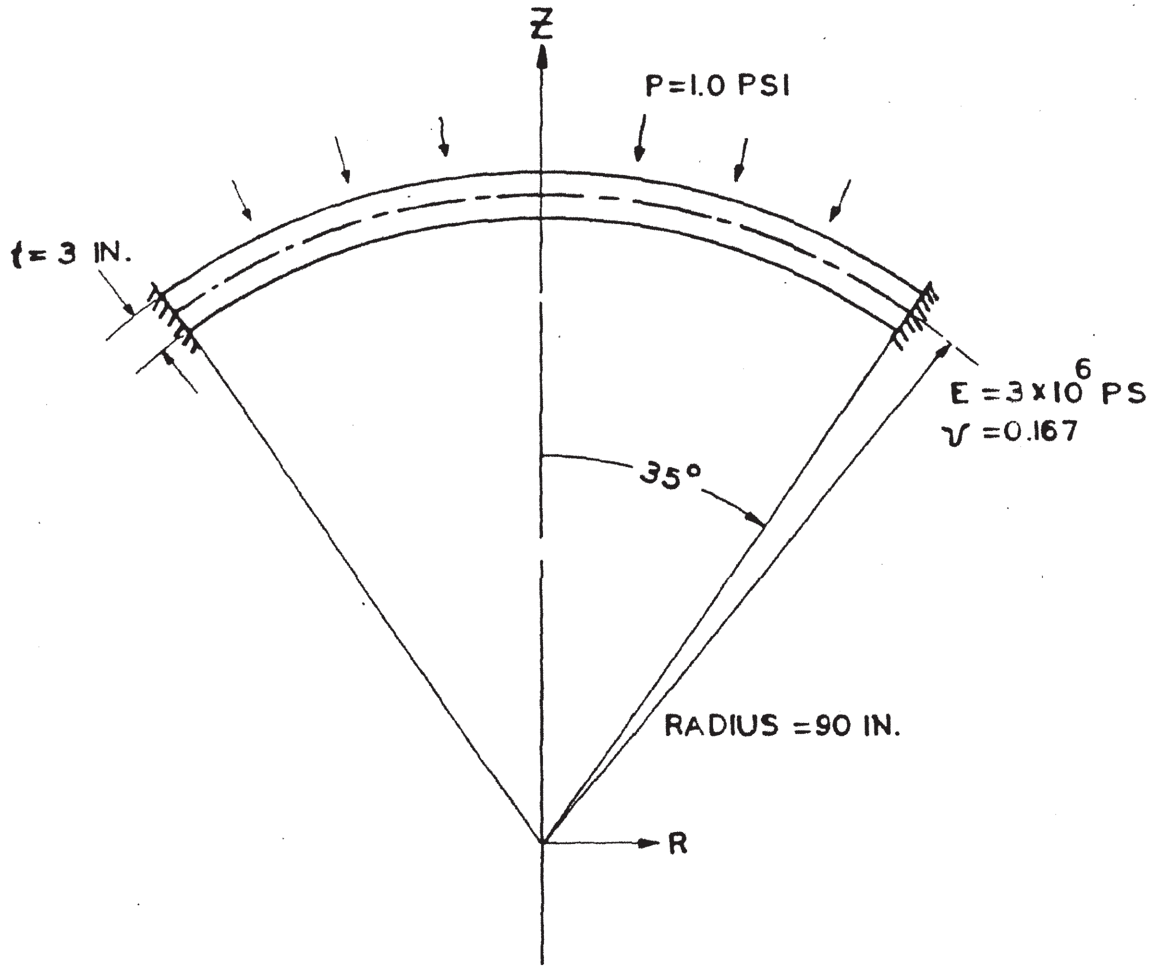

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FIGURE F.10-1

UNITS 1 & 2

SHALLOW SPHERICAL SHELL

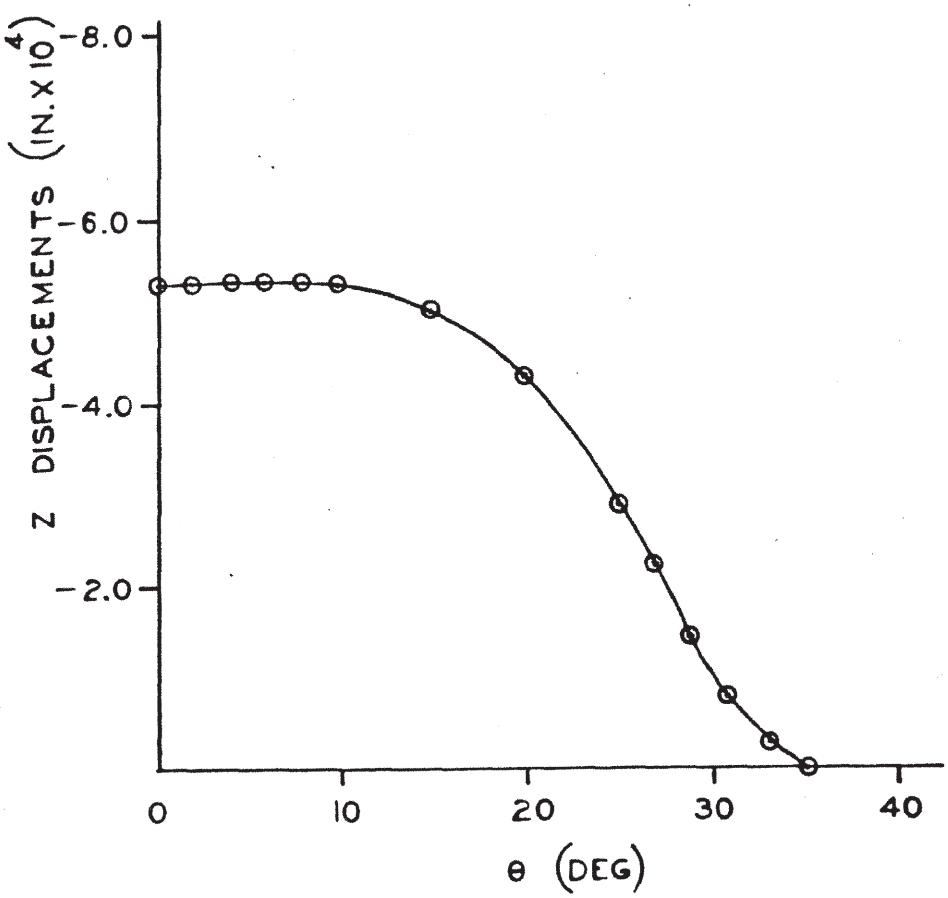

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FIGURE F.11-1

UNITS 1 & 2

AXIAL DISPLACEMENT SHALLOW SPHERICAL SHELL

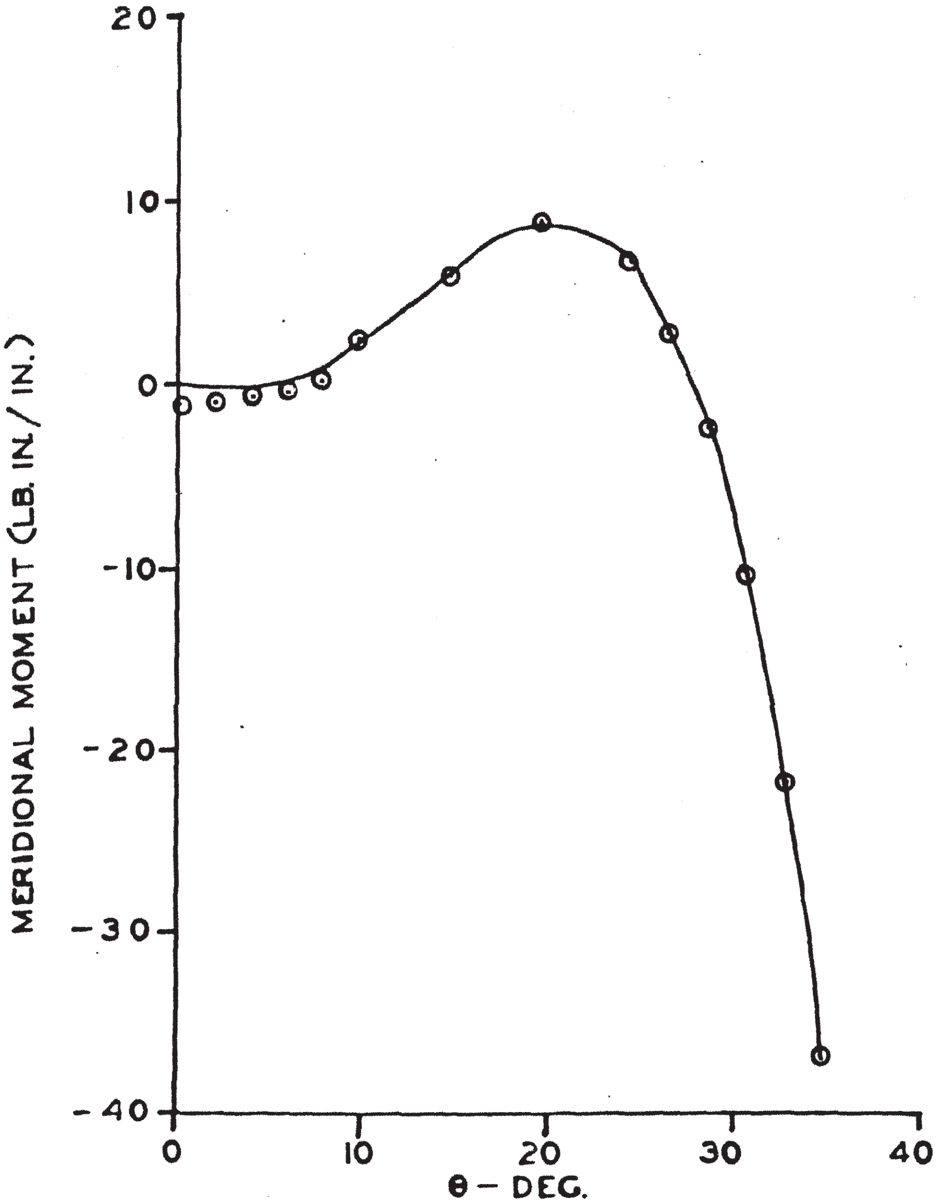

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FIGURE F.12-1 UNITS 1 & 2 MERIDIONAL MOMENT SHALLOW SPHERICAL SHELL

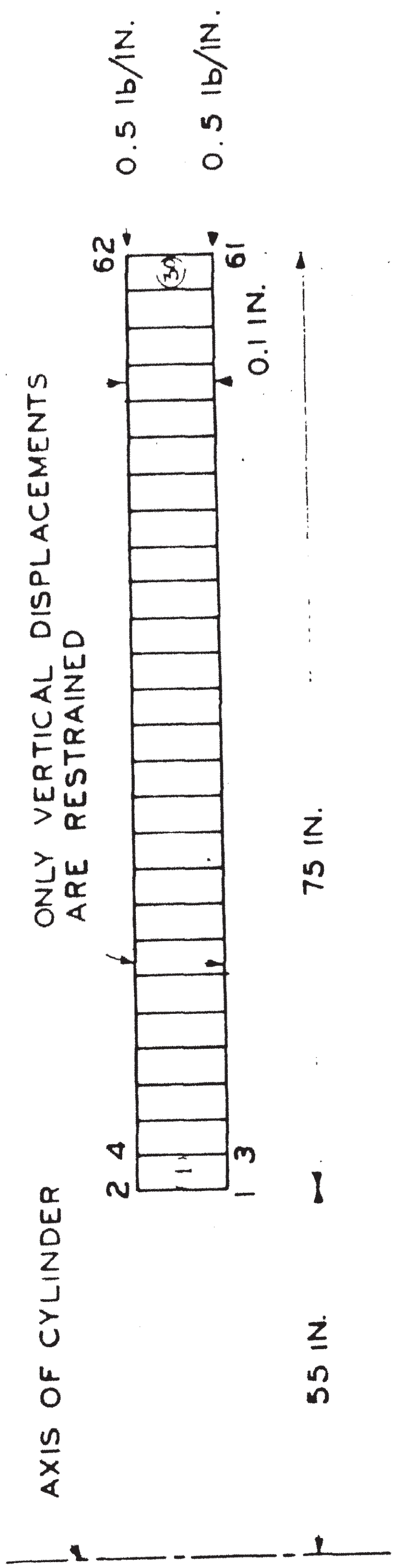

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FIGURE F.13-1

UNITS 1 & 2

FINITE ELEMENT IDEALIZATION OF THICK-WALLED CYLINDER

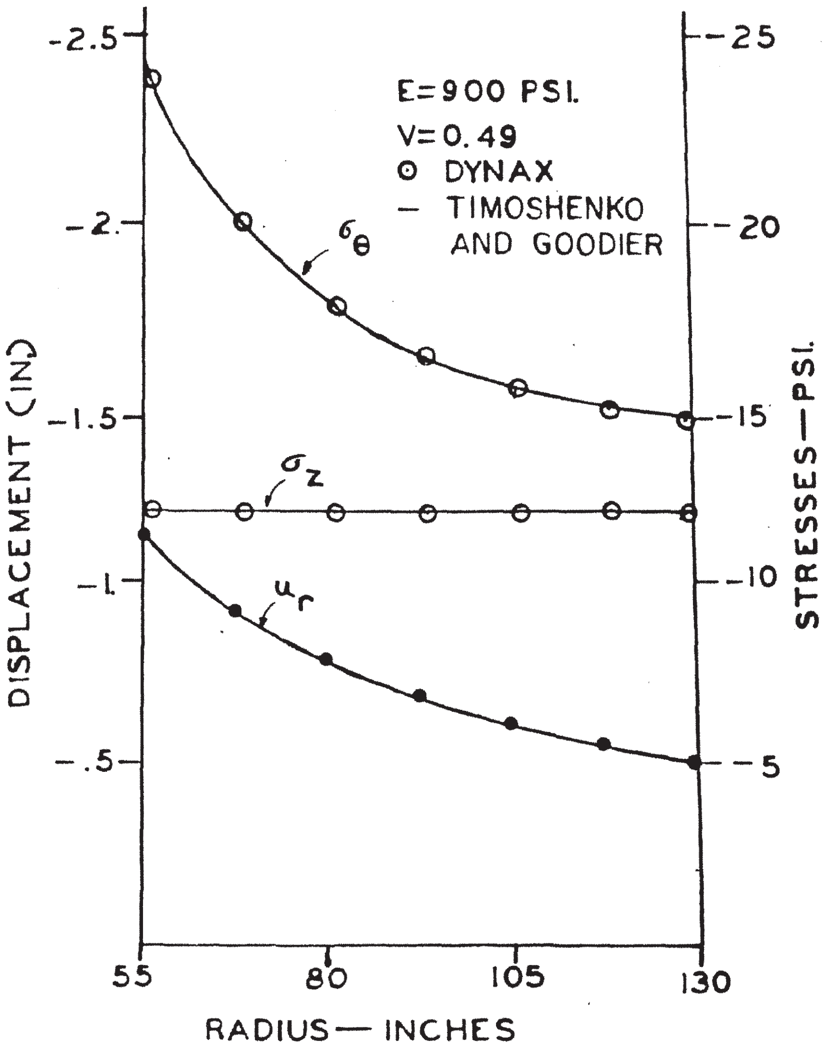

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FIGURE F.14-1

UNITS 1 & 2

STRESSES AND DISPLACEMENTS IN THICK-WALLED CYLINDERS

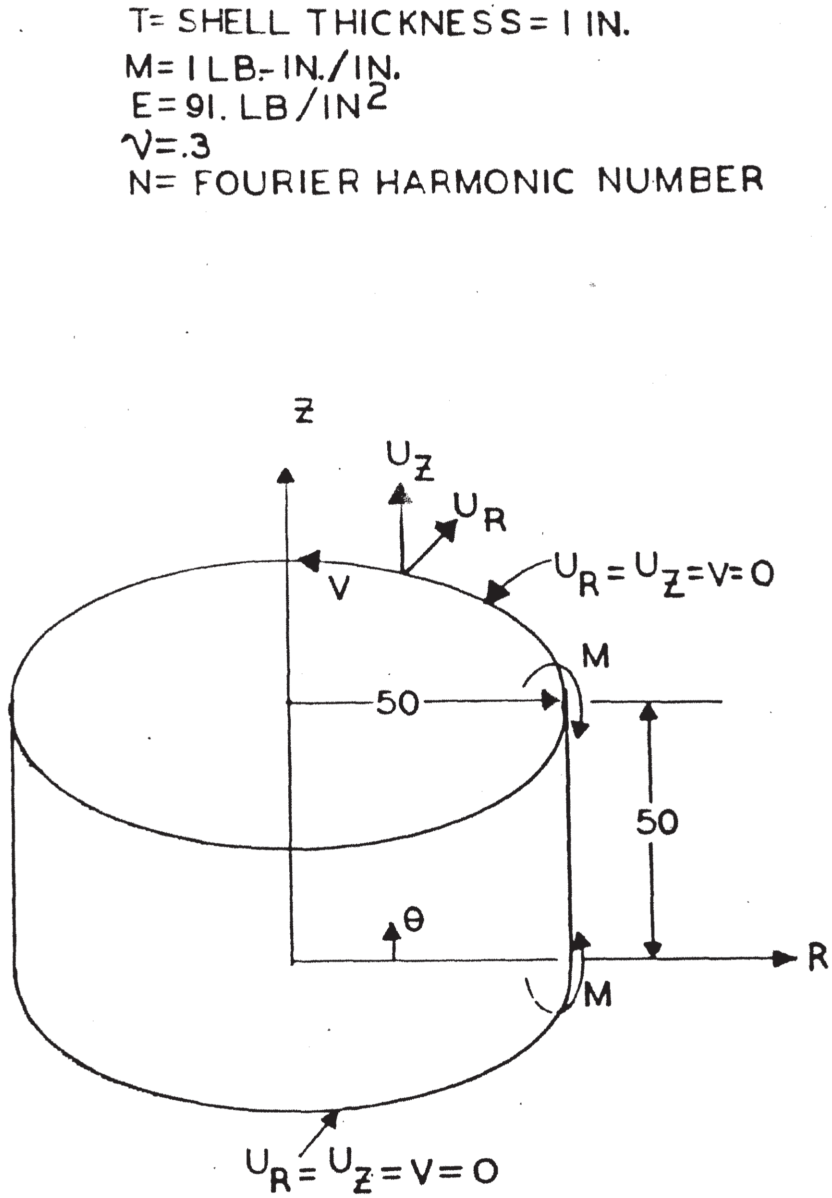

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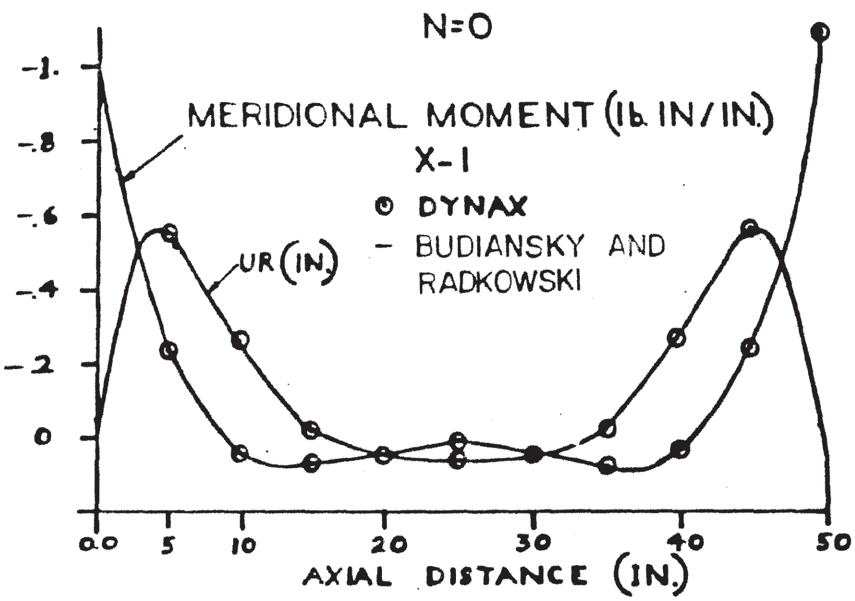
FIGURE F.15-1

UNITS 1 & 2

CYLINDER UNDER HARMONIC LOADS



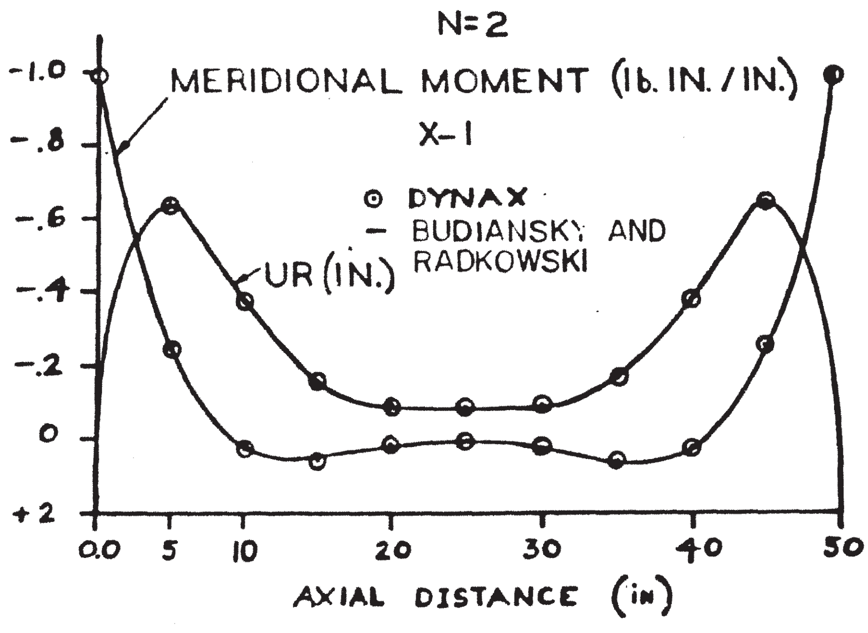


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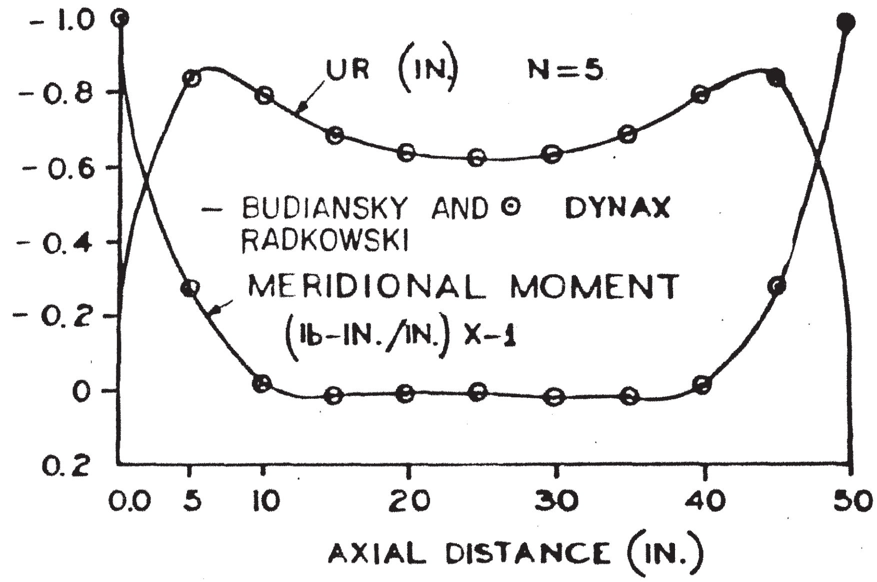
FIGURE F.16-1 UNITS 1 & 2 MERIDIONAL MOMENTS AND DEFLECTIONS OF CYLINDER – N=0 AND N=2



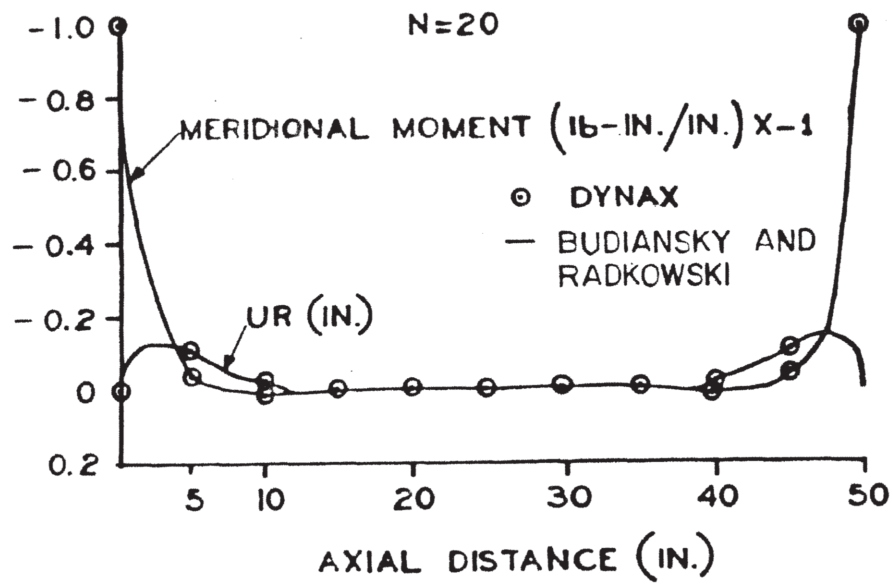


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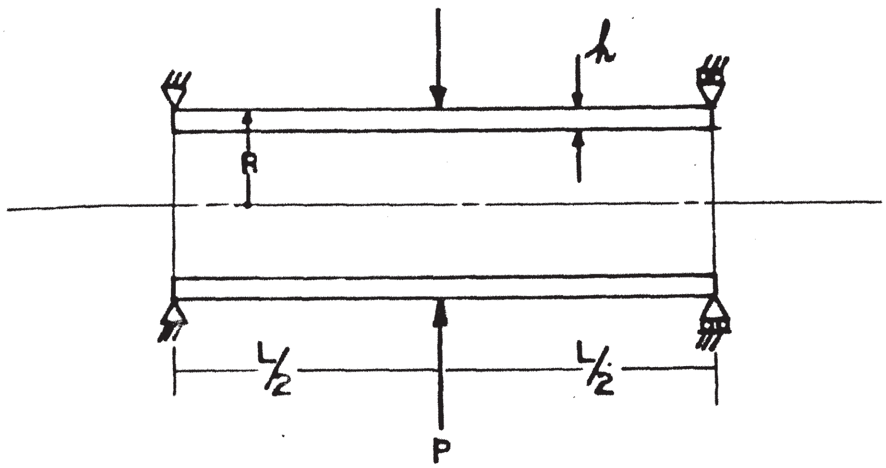
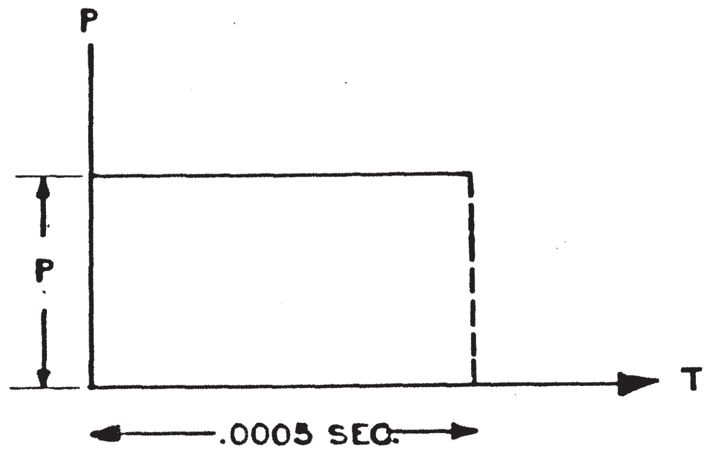
FIGURE F.17-1 UNITS 1 & 2 MERIDIONAL MOMENTS AND DEFLECTIONS OF CYLINDER – N=5 AND N=20





TIME HISTORY LOADING

$L = 18$ IN. MASS DENSITY $(\rho) = 0.0187 \dfrac{\# \, SEC.^2}{IN^4}$

$P = 500$ lb.

$\nu = 0.3$

$R = 3$ IN.

$h = 0.3$ IN. TIME STEP = .000005 SEC.

$E = 30 \times 10^6$ lb./IN.^{2}

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FIGURE F.18-1

UNITS 1 & 2
SUDDENLY APPLIED RING LINE LOAD

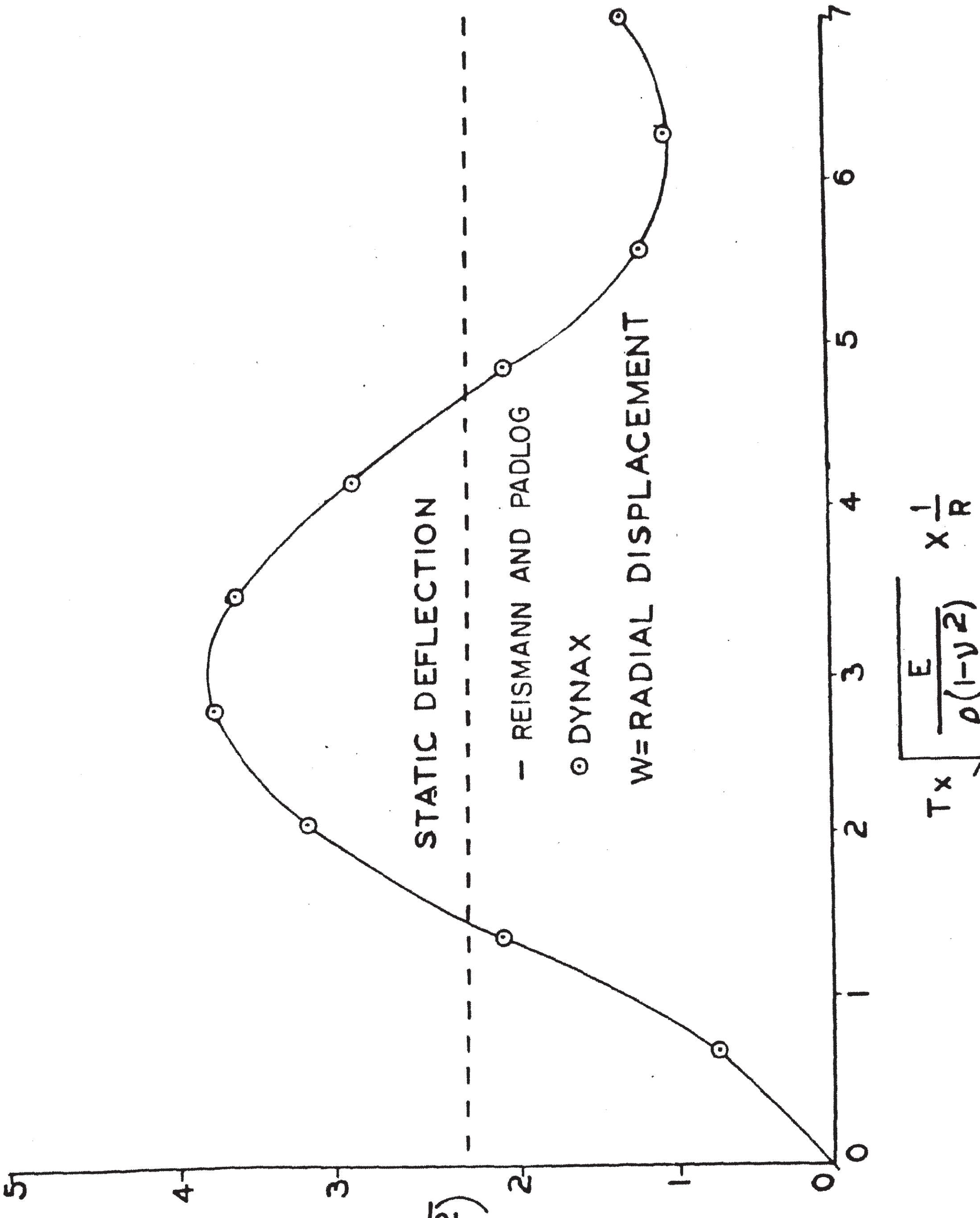

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FIGURE F.19-1

UNITS 1 & 2
RADIAL DISPLACEMENT VS. TIME

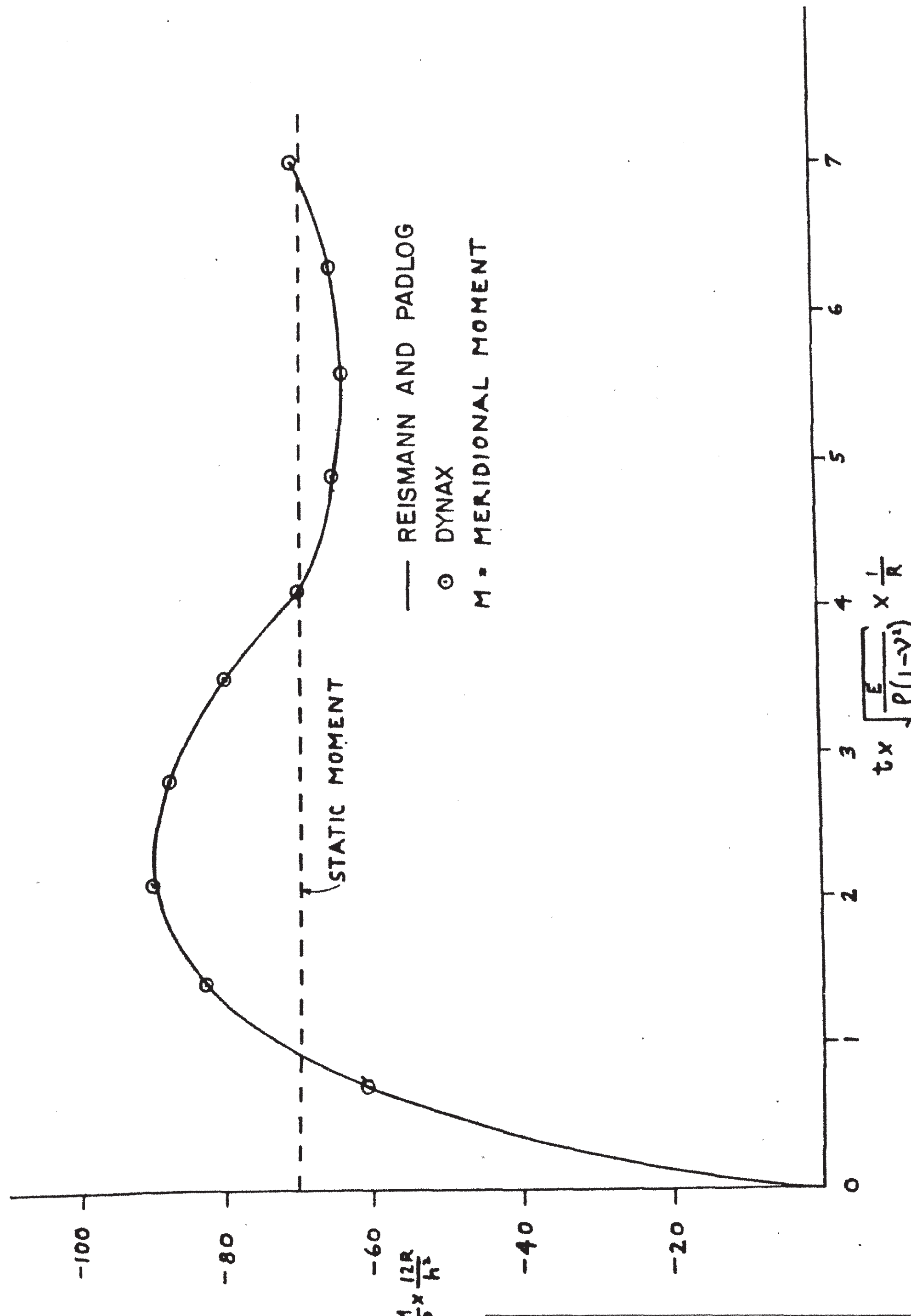

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FIGURE F.20-1

UNITS 1 & 2

BENDING MOMENT VS. TIME – SUDDENLY APPLIED RING (LINE) LOAD

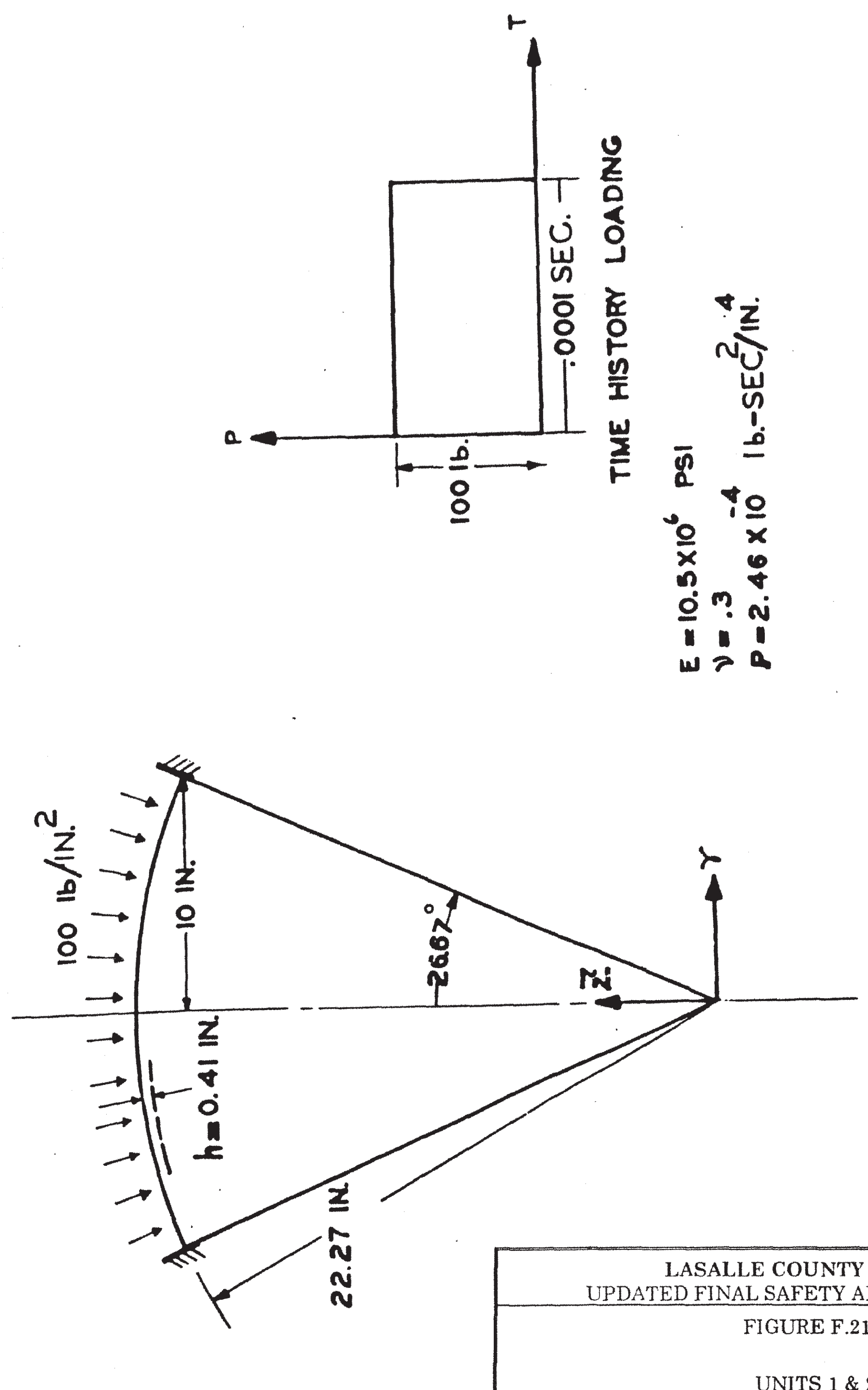

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FIGURE F.21-1

UNITS 1 & 2
SPHERICAL CAP

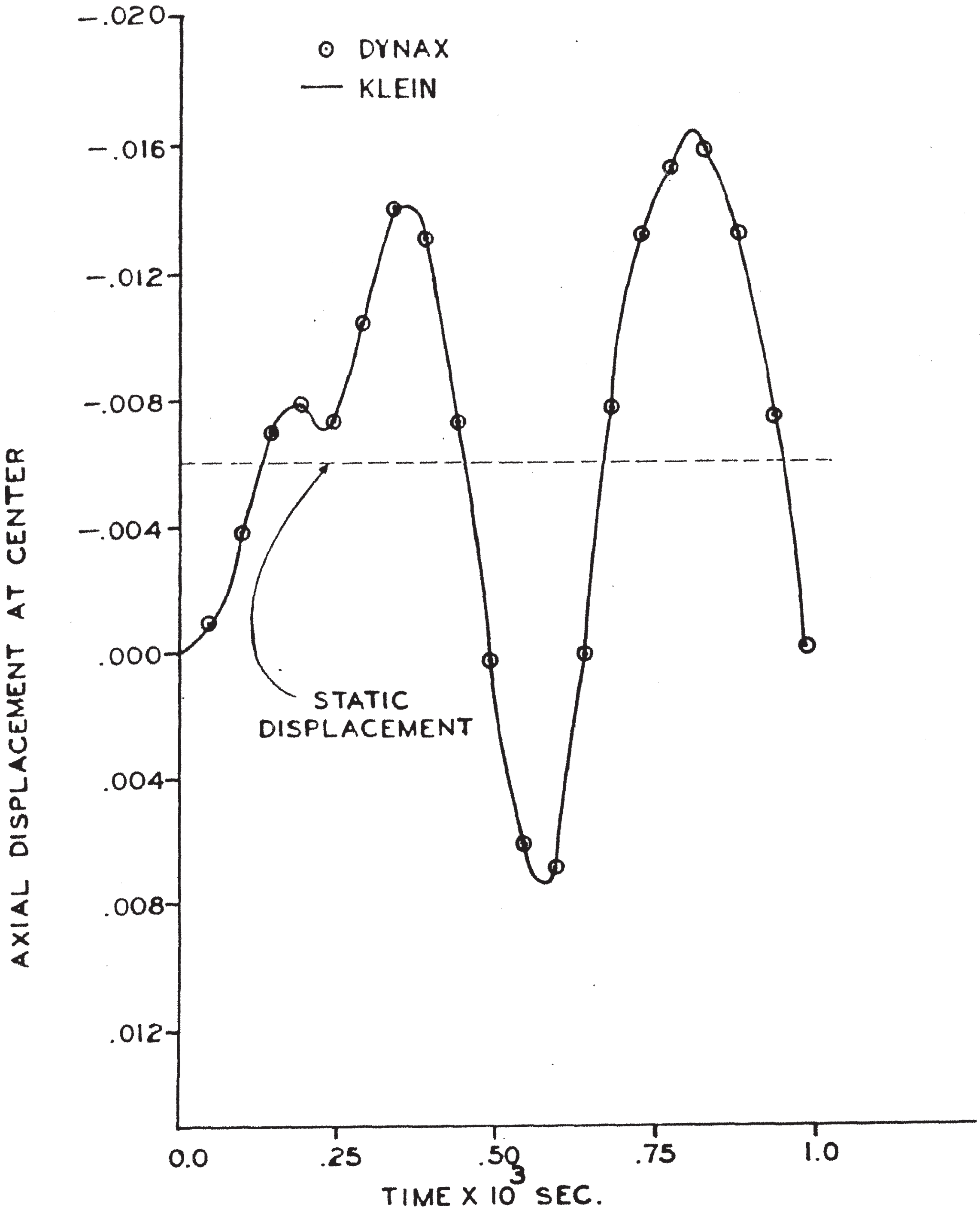

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FIGURE F.22-1 UNITS 1 & 2 AXIAL DISPLACEMENT OF SPHERICAL CAP UNDER DYNAMIC LOAD

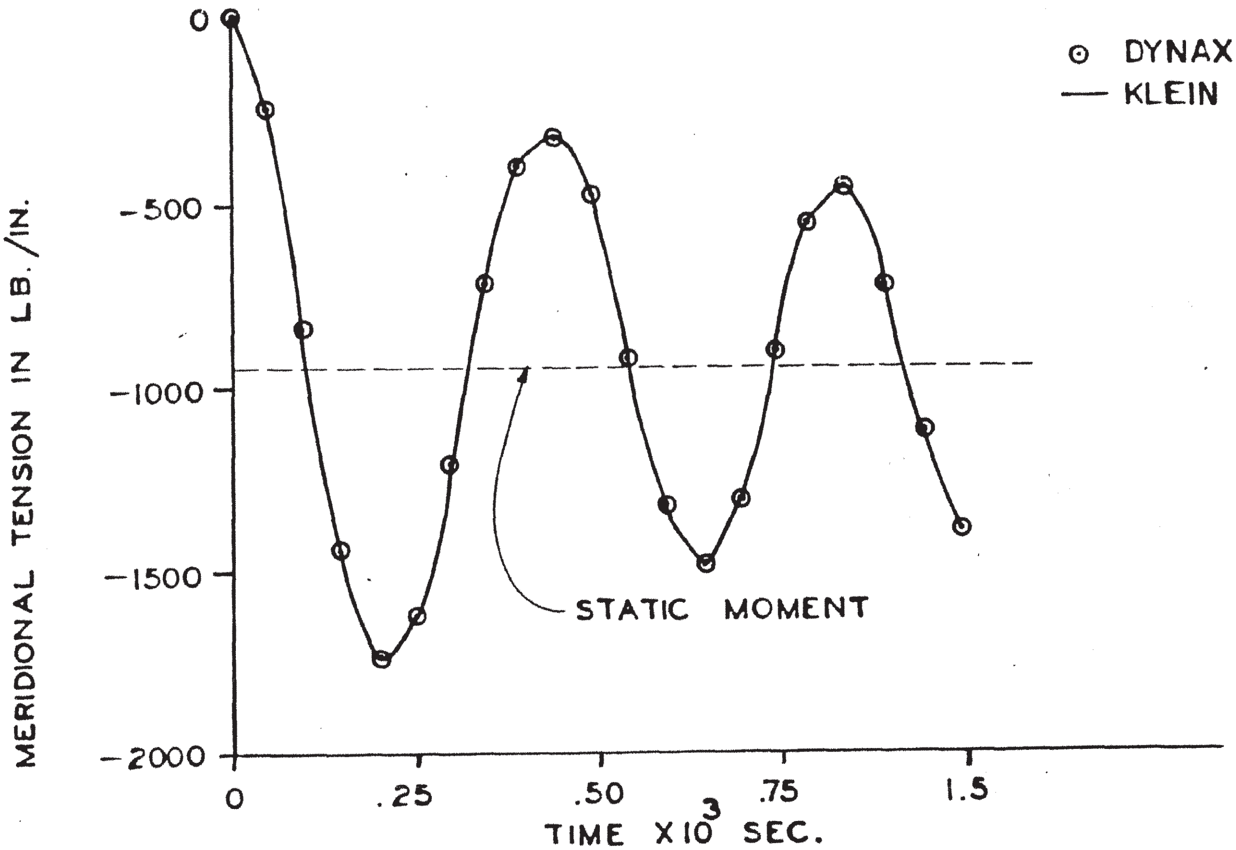

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FIGURE F.23-1 UNITS 1 & 2 MERIDIONAL TENSION OF SPHERICAL CAP UNDER DYNAMIC LOAD

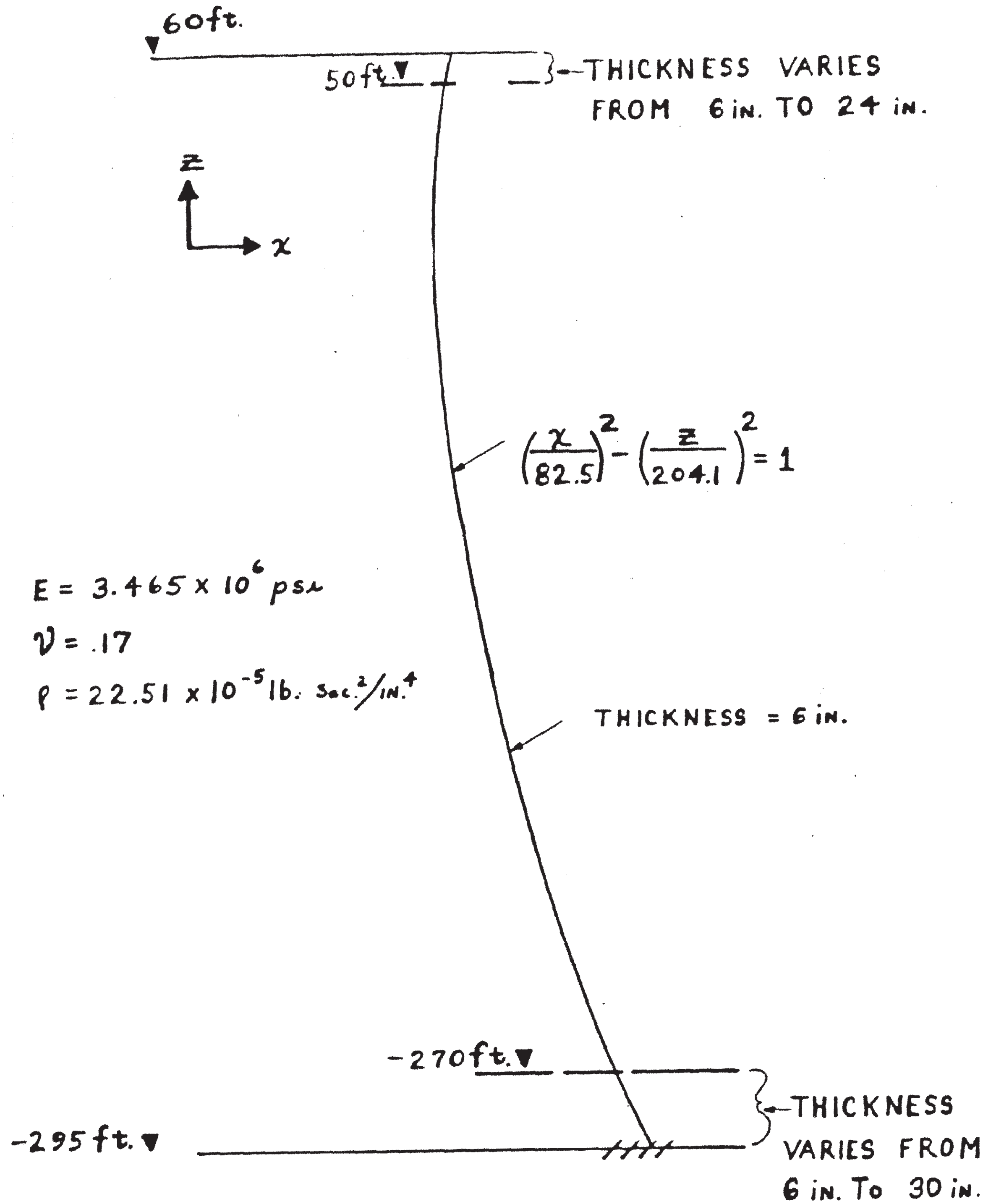

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FIGURE F.24-1

UNITS 1 & 2
HYPERBOLIC COOLING TOWER

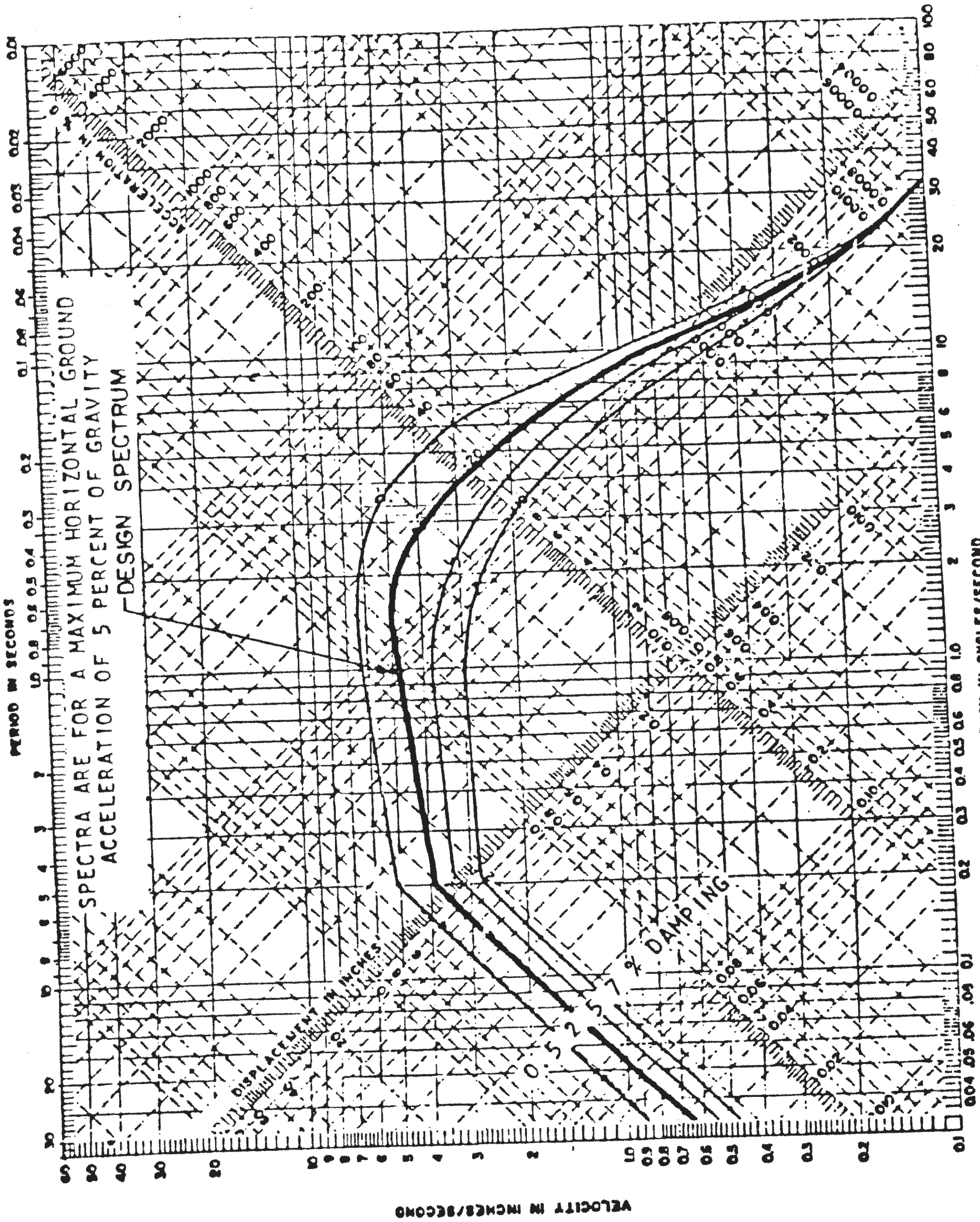

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FIGURE F.25-1 UNITS 1 & 2 SPECTRUM OF DESIGN EARTHQUAKE

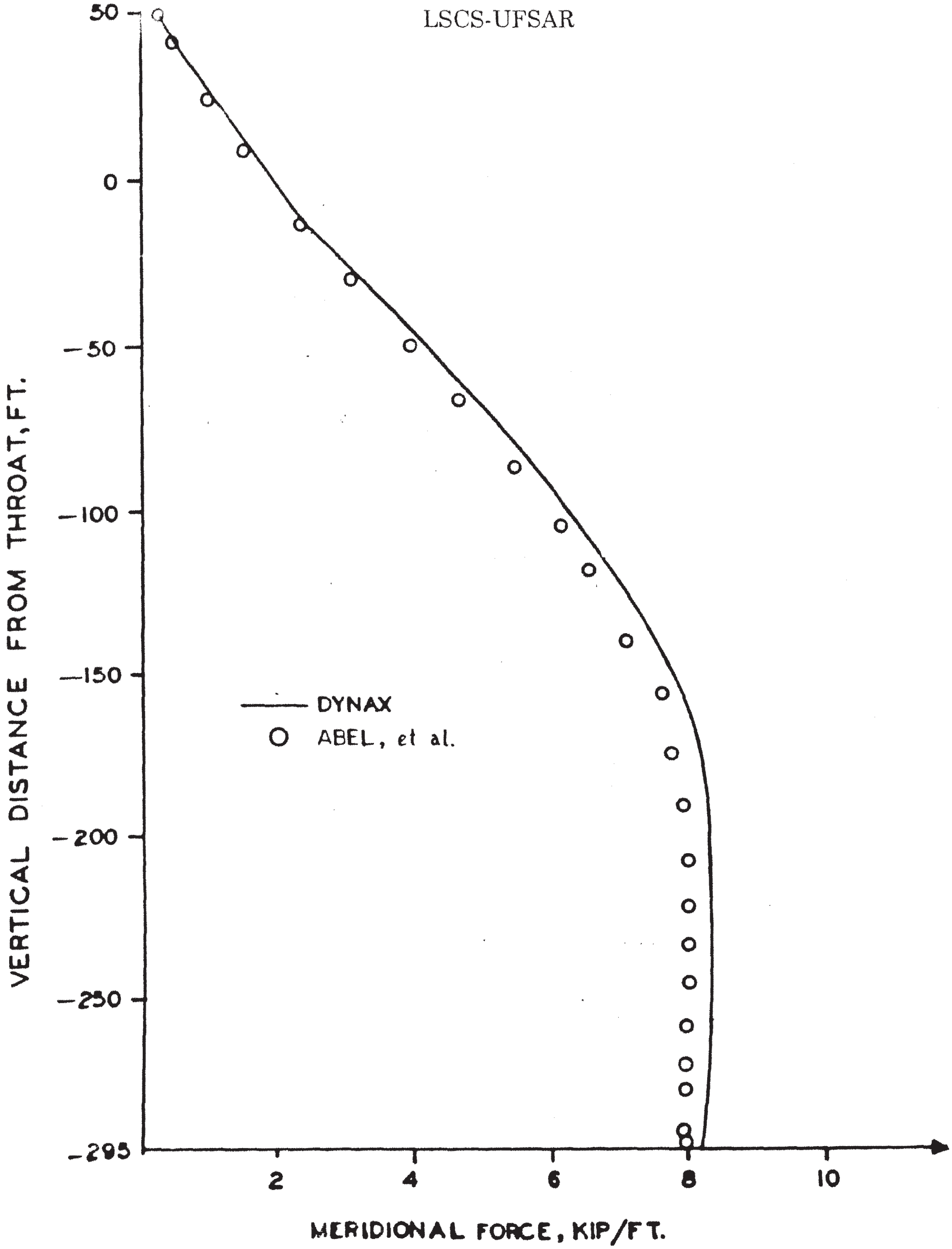

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FIGURE F.26-1

UNITS 1 & 2
COOLING TOWER MERIDIONAL FORCE

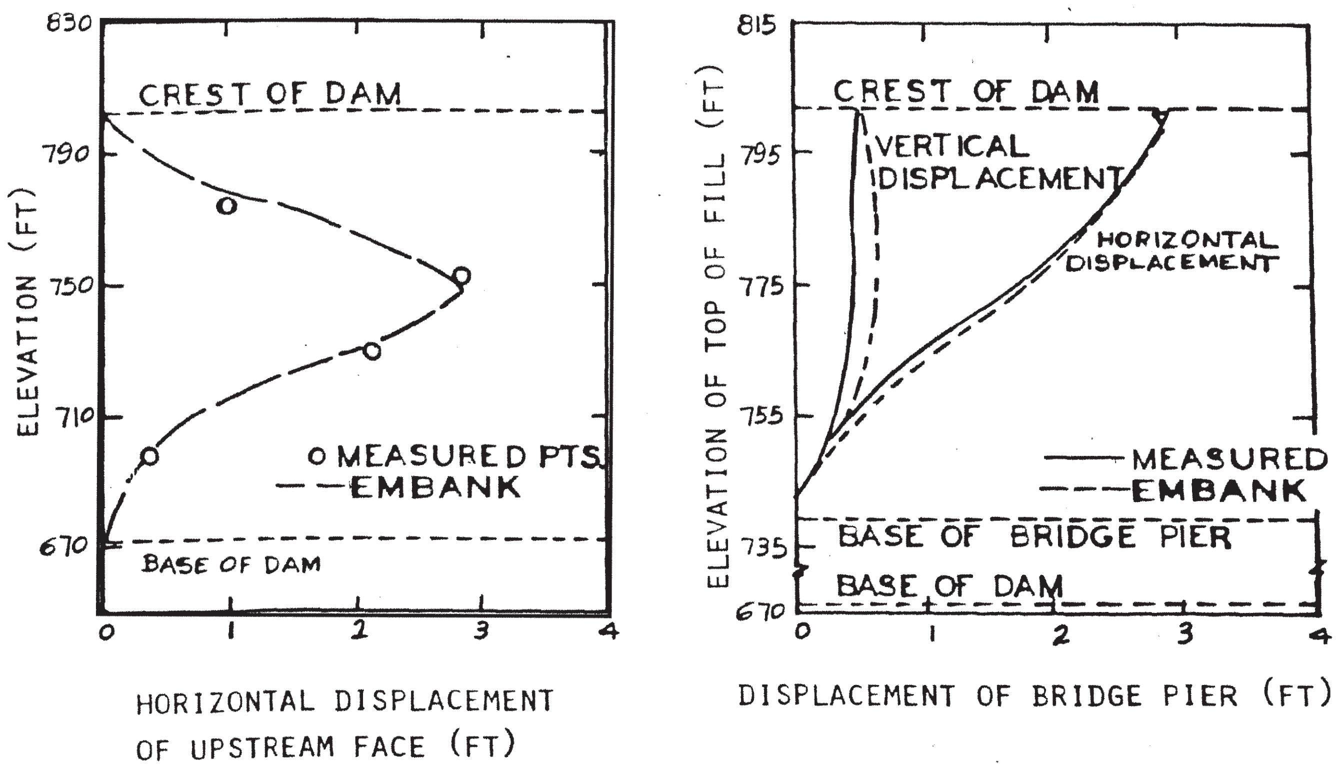

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FIGURE F.27-1

UNITS 1 & 2

COMPARISON OF MEASURED DISPLACEMENTS IN OTTER BROOK DAM WITH DISPLACEMENTS CALCULATED USING EMBANK

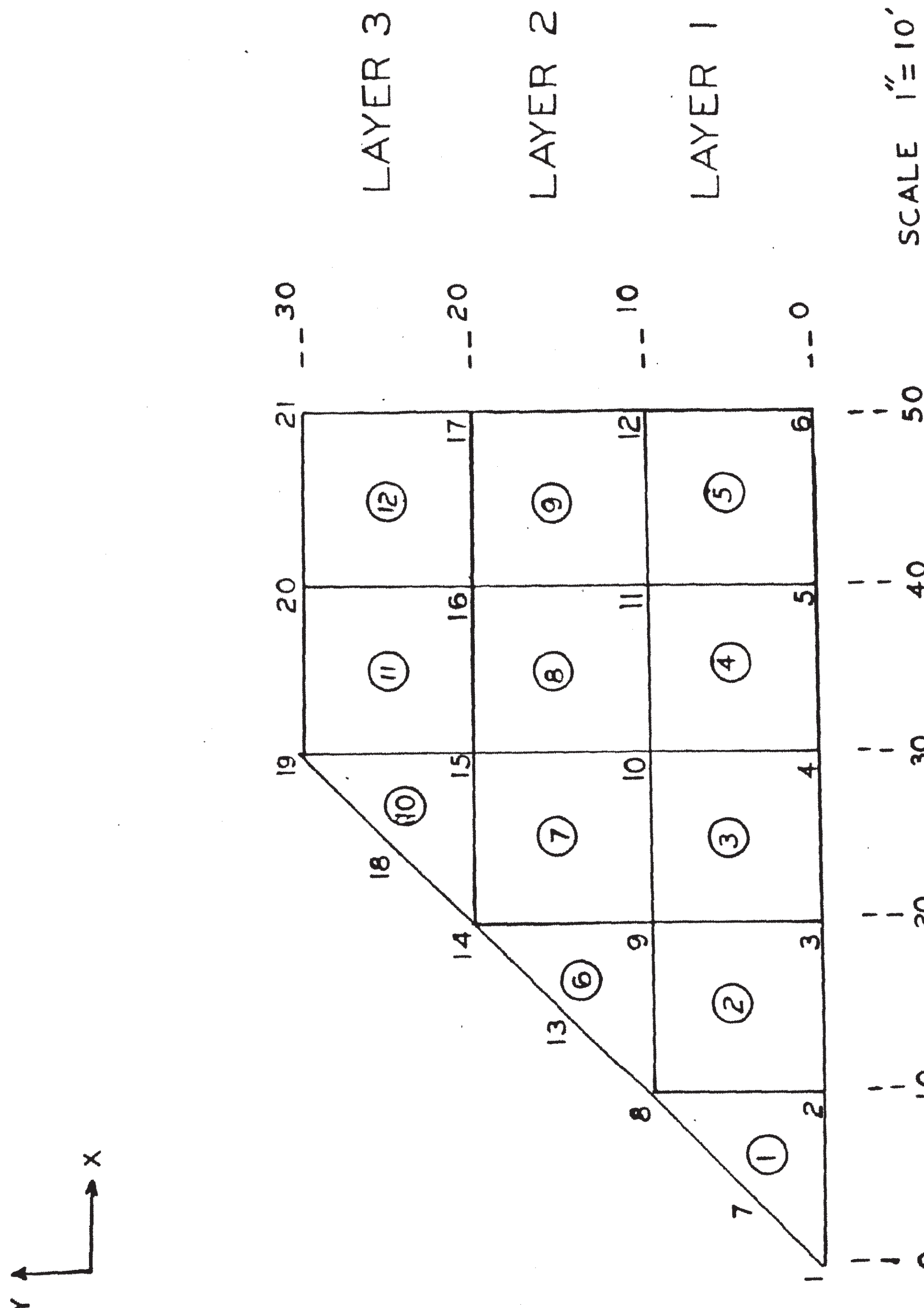

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FIGURE F.28-1

UNITS 1 & 2
MESH FOR SAMPLE PROBLEM (EMBANK)

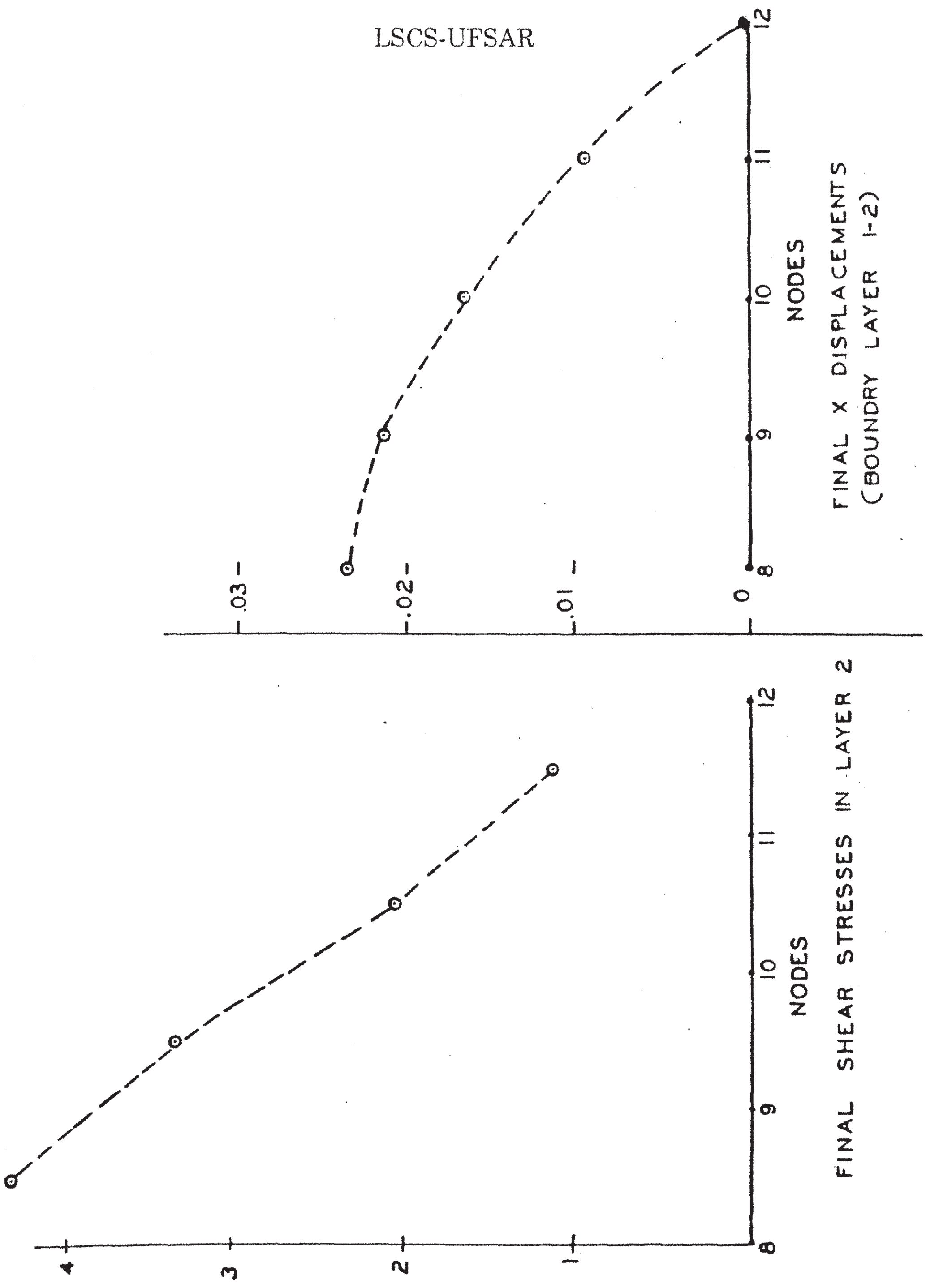

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FIGURE F.29-1

UNITS 1 & 2

COMPARISON OF RESULTS FROM ORIGINAL VERSION OF EMBANK AND SARGENT & LUNDY VERSION

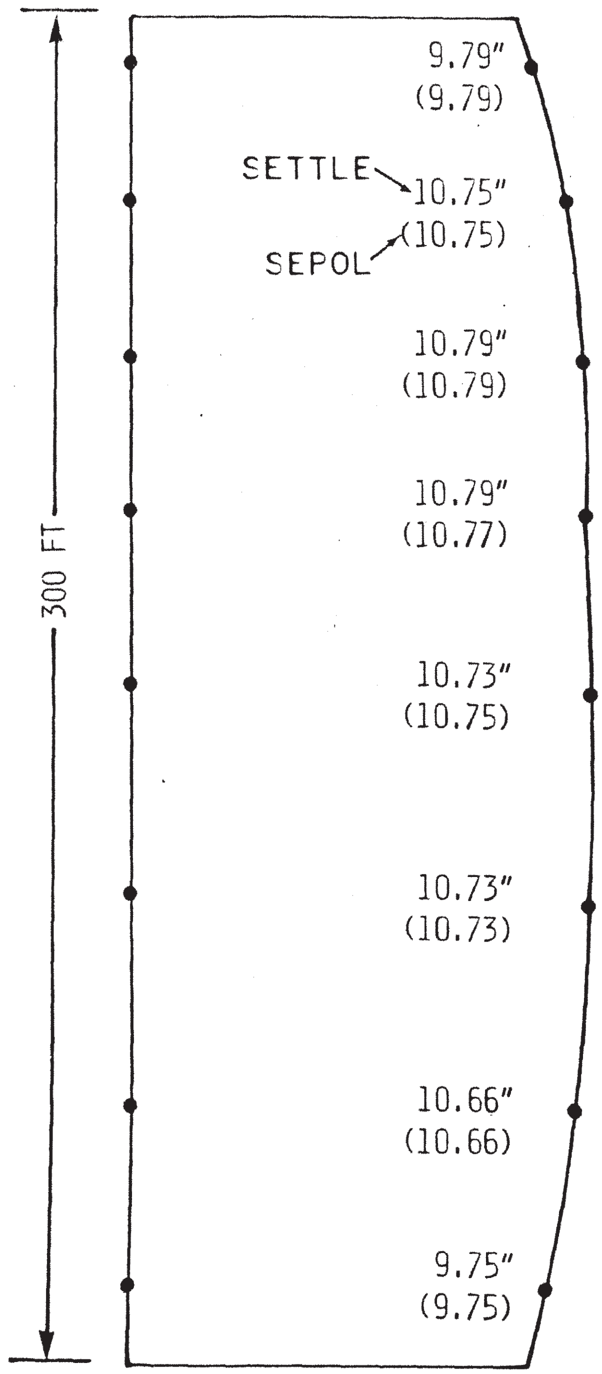

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FIGURE F.30-1

UNITS 1 & 2
SETTLEMENT PROFILE COMPUTED BY
SETTLE AND SEPOL

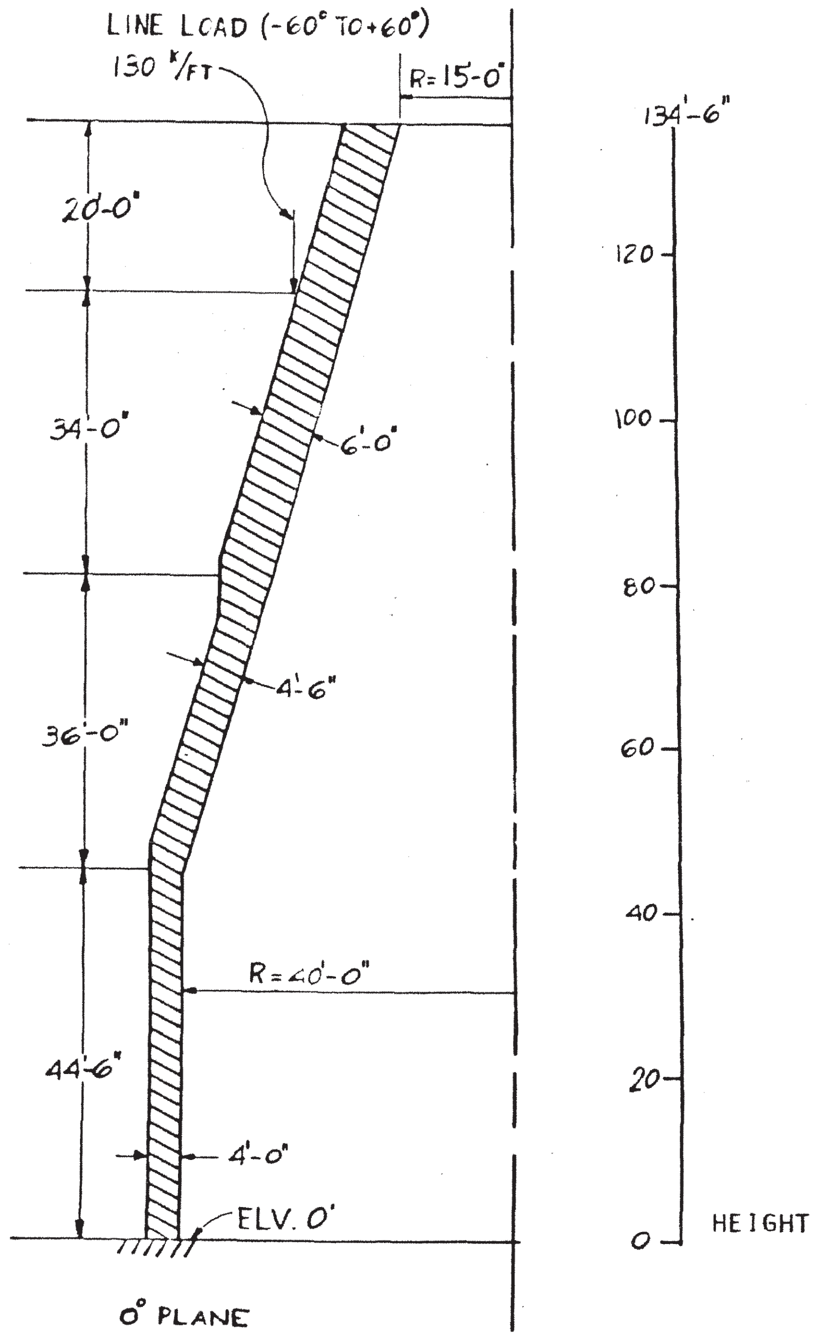

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FIGURE F.31-1

UNITS 1 & 2

KALSHEL VALIDATION EXAMPLE – ECCENTRIC LINE LOAD ON CONICAL/CYLINDRICAL SHELL

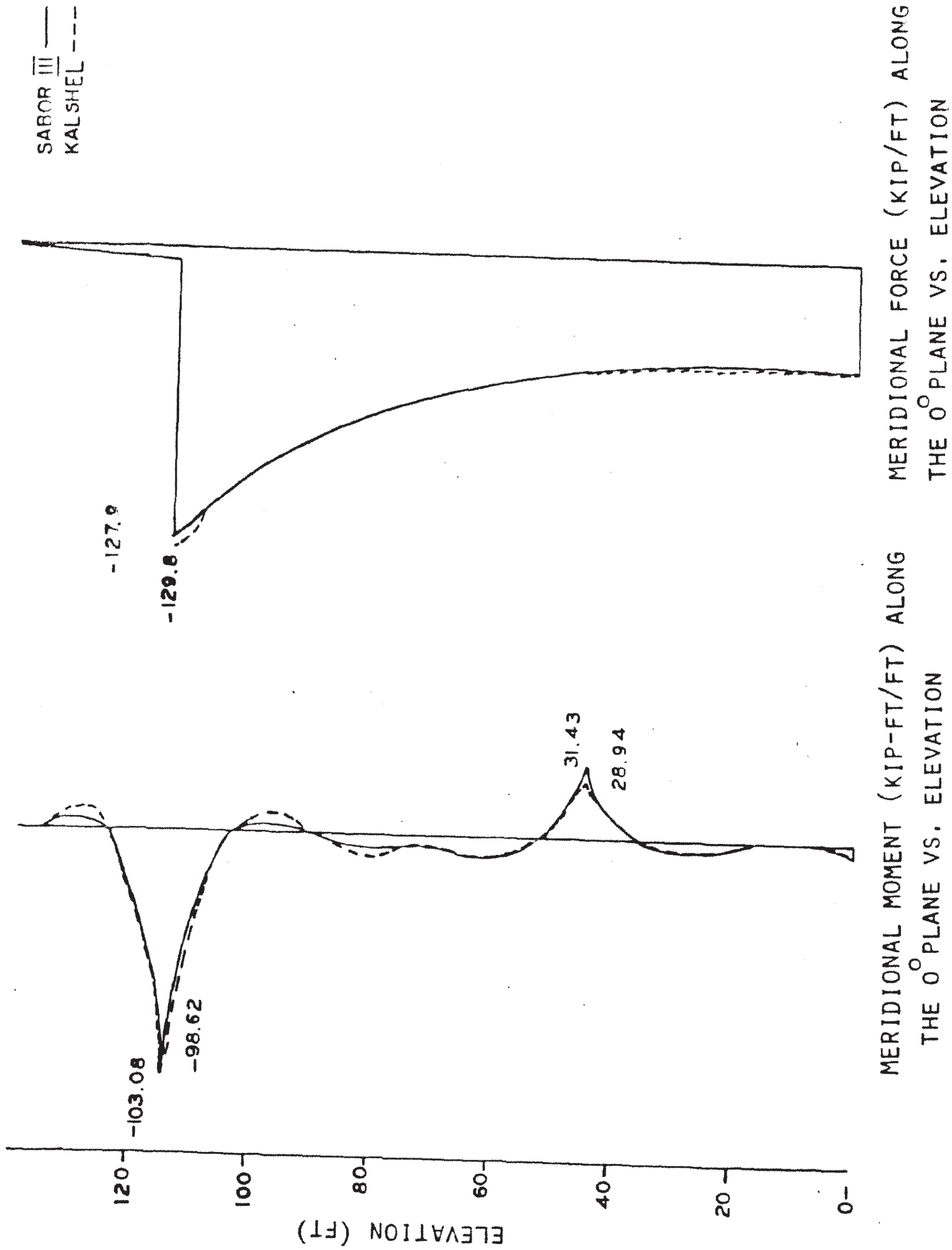

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FIGURE F.32-1

UNITS 1 & 2
MERIDIONAL MOMENT AND MERIDIONAL FORCE ALONG THE 0° PLANE

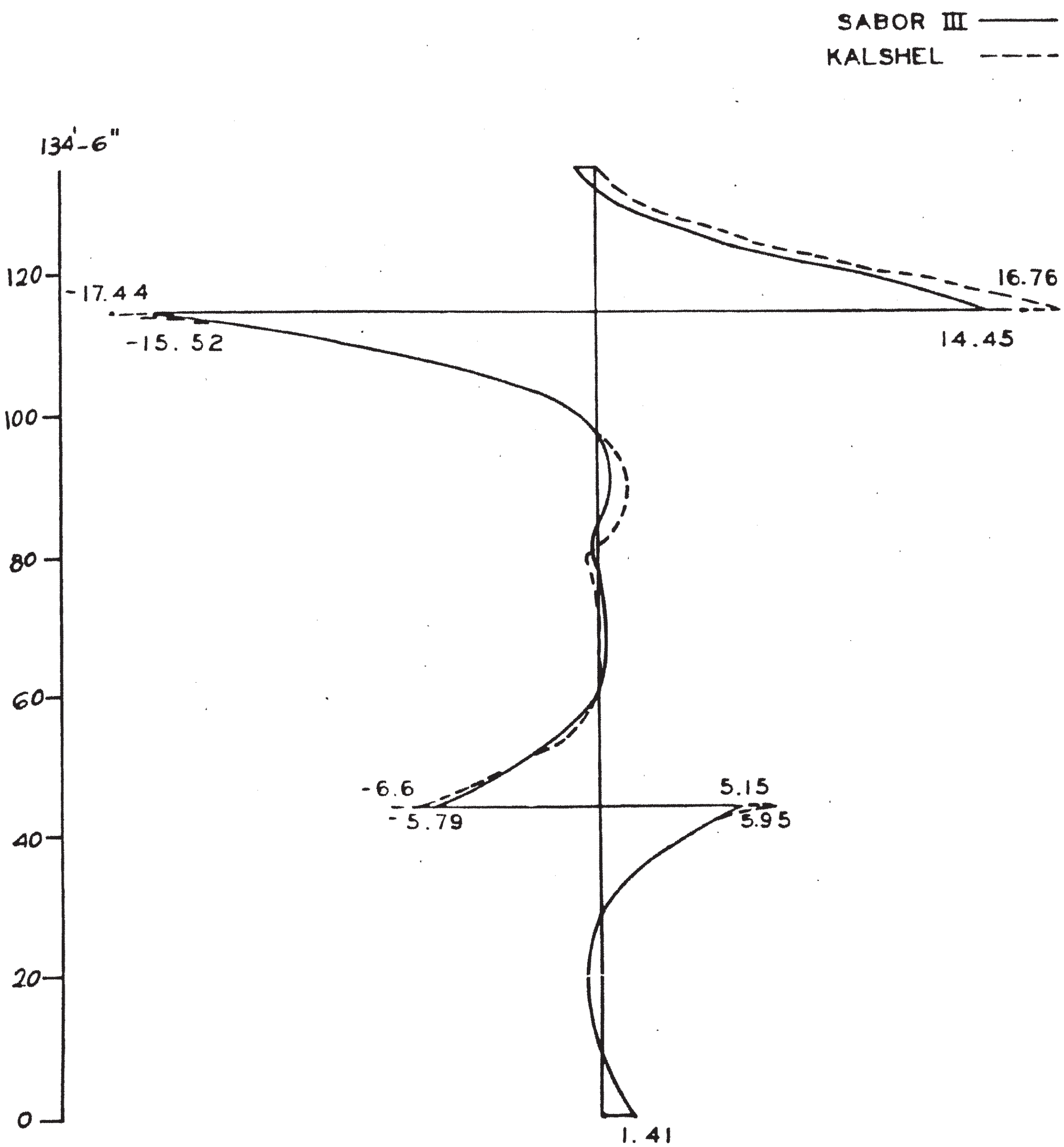

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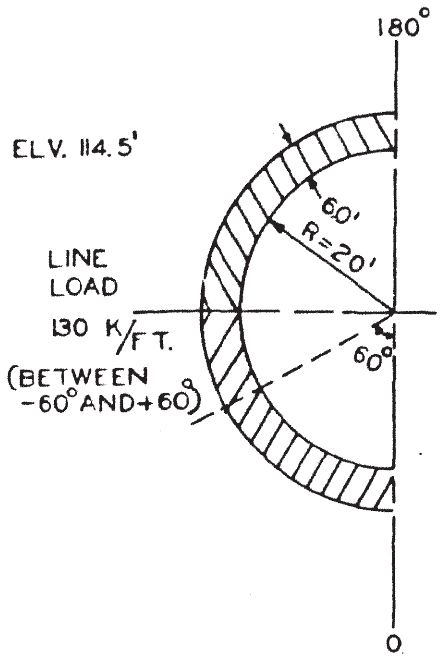
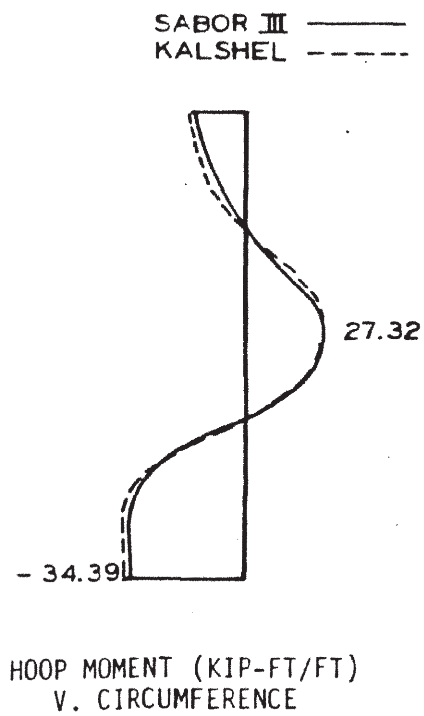
FIGURE F.33-1 UNITS 1 & 2 RADIAL SHEAR ALONG THE 0° PLANE





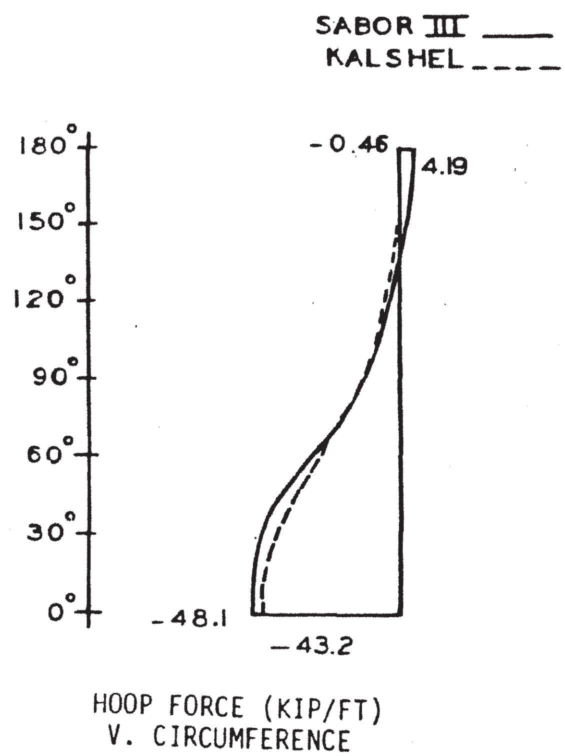


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FIGURE F.34-1

UNITS 1 & 2

HOOP MOMENT AND HOOP FORCE

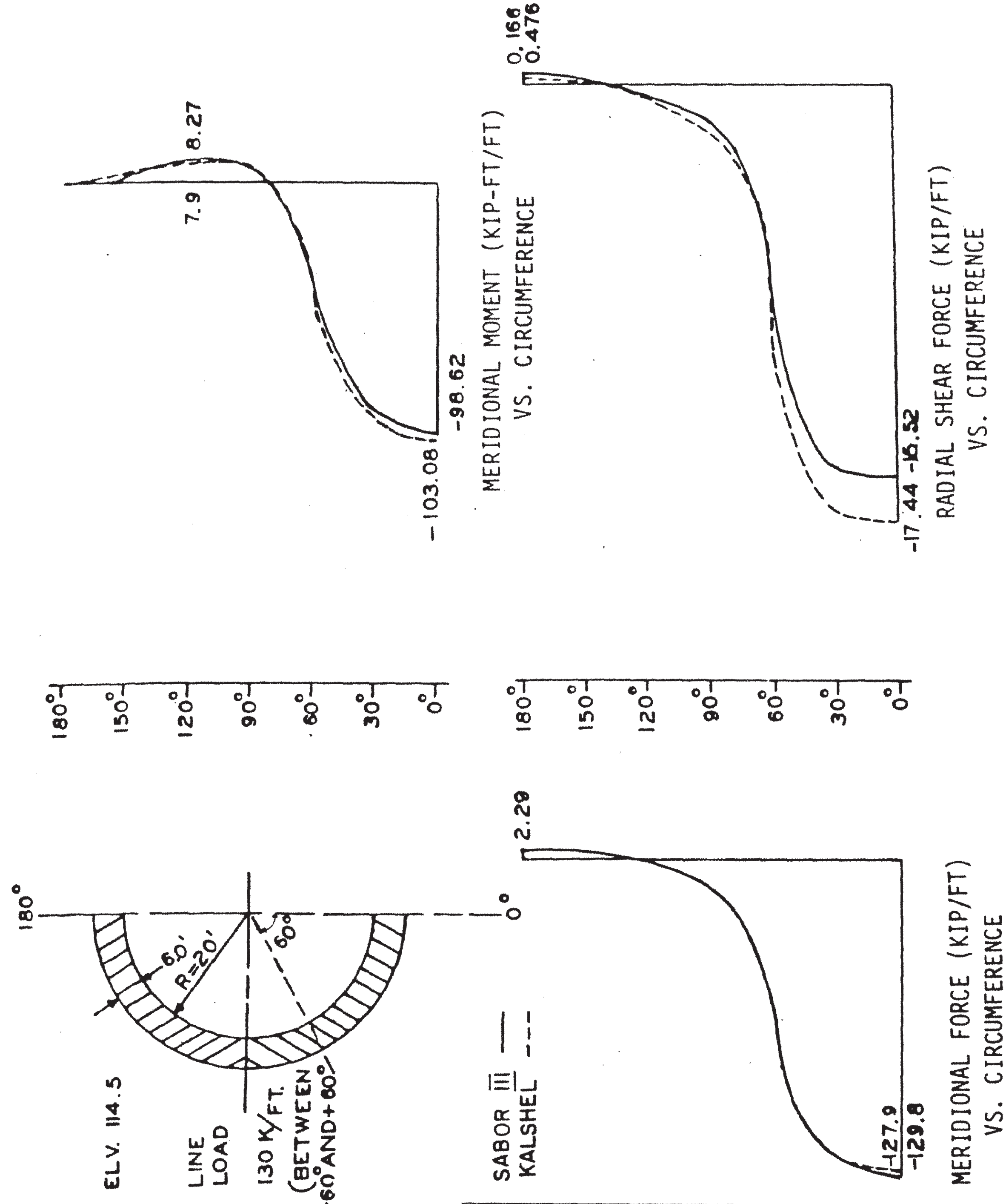

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FIGURE F.35-1

UNITS 1 & 2

MERIDIONAL FORCE, MERIDIONAL MOMENT, AND RADIAL SHEAR FORCE

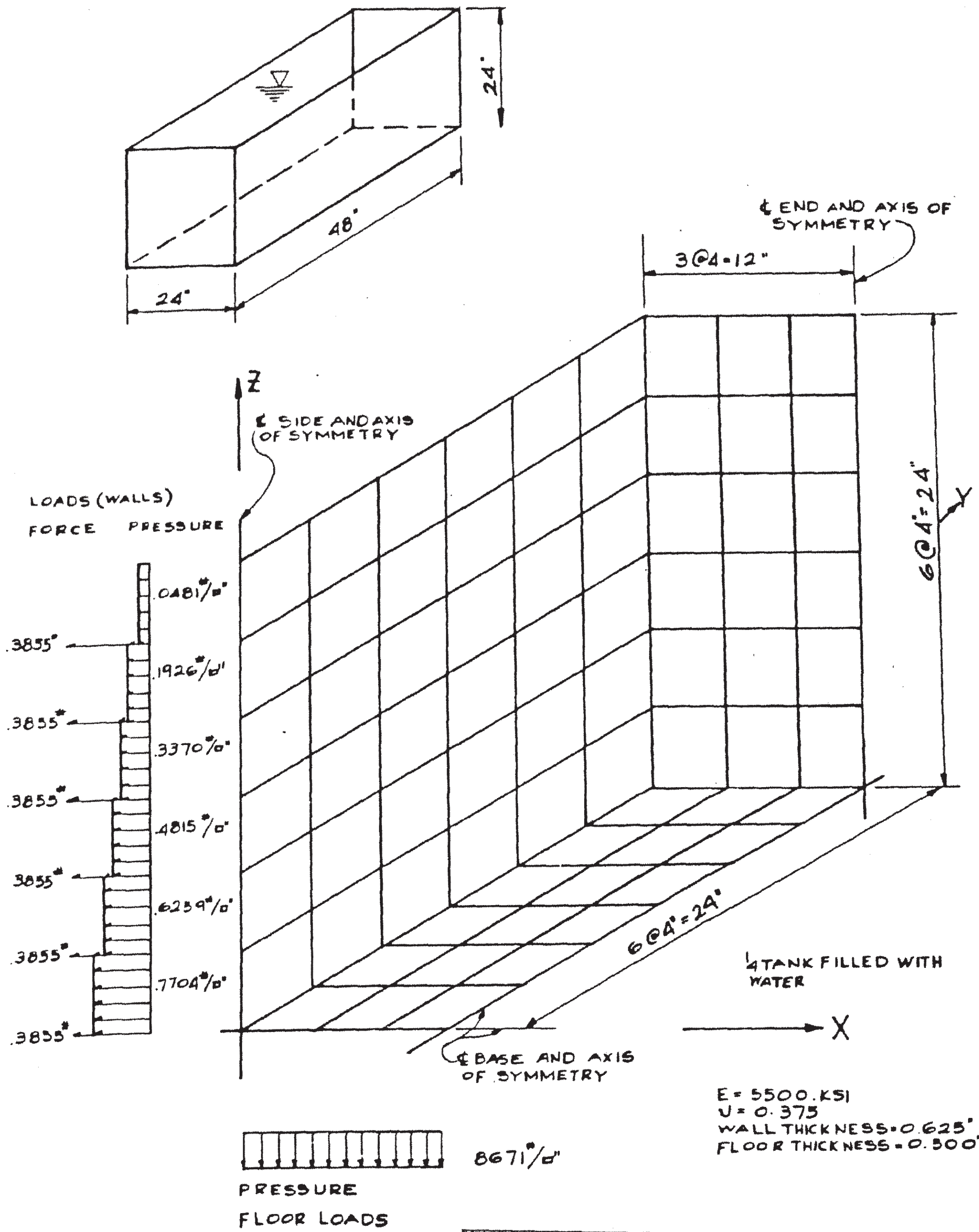

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FIGURE F.36-1

UNITS 1 & 2
RECTANGULAR TANK FILLED WITH WATER

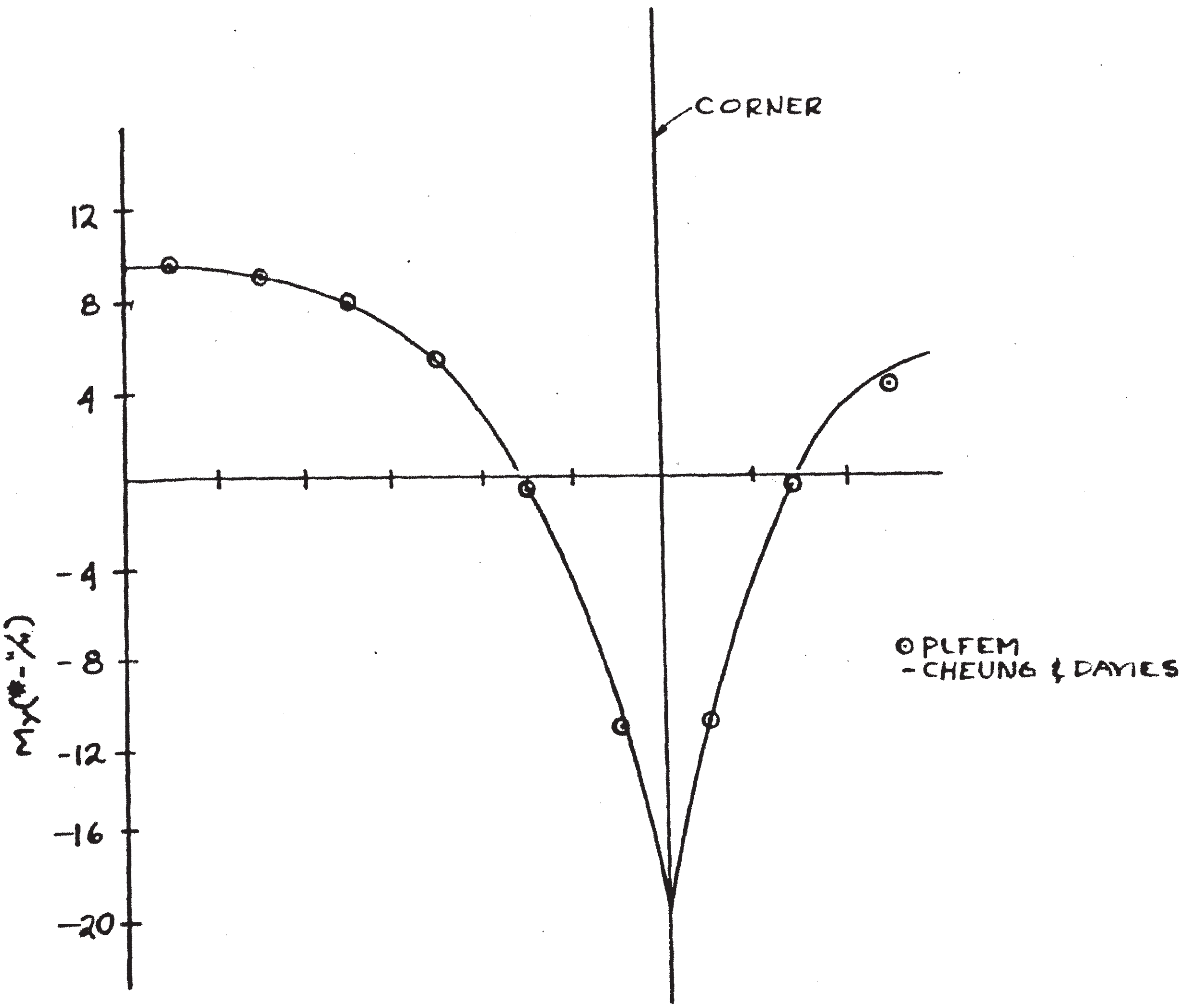

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FIGURE F.37-1

UNITS 1 & 2
MOMENT OF M_y AT HORIZONTAL
CROSS SECTION OF WALLS

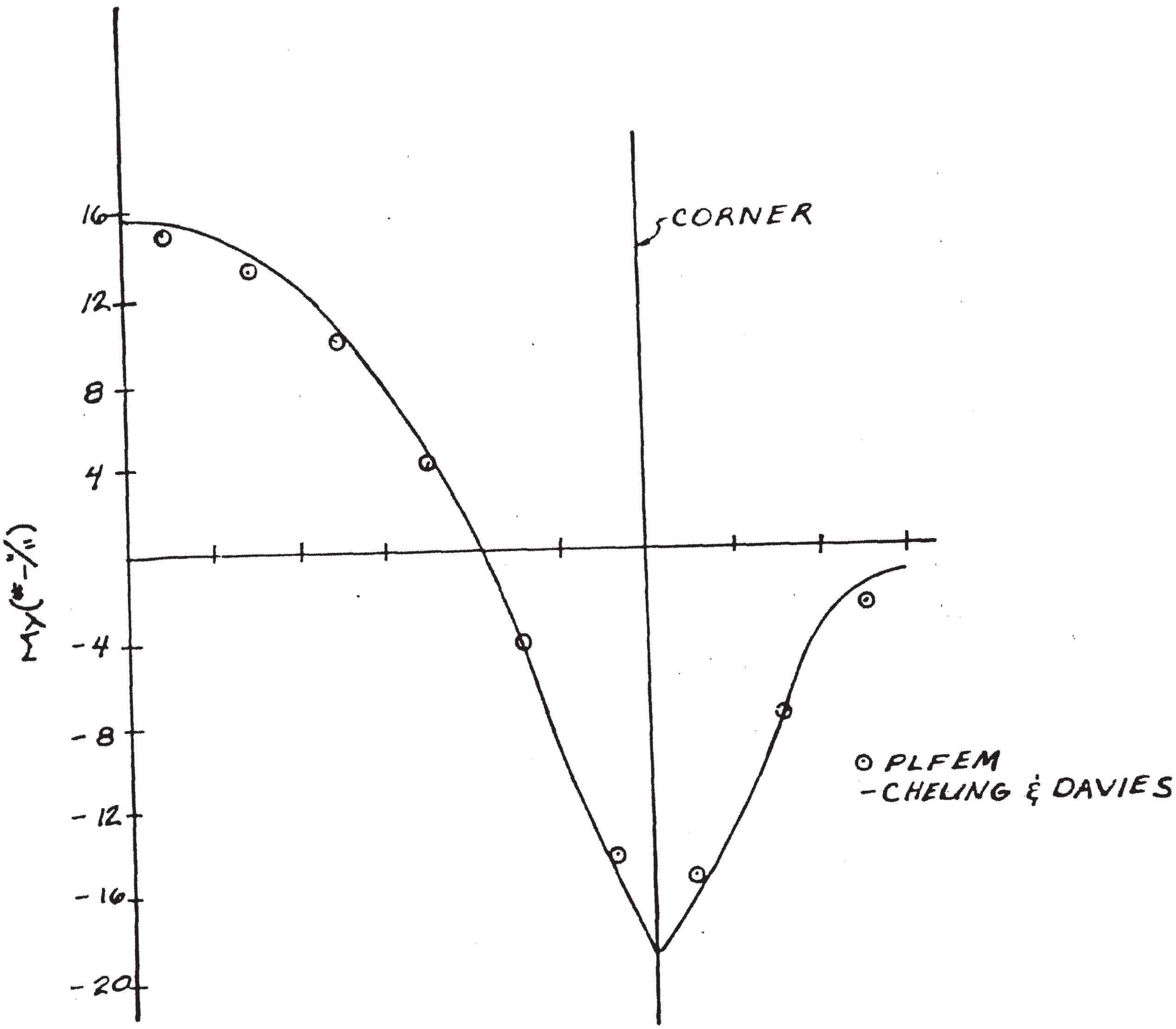

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FIGURE F.38-1

UNITS 1 & 2

MOMENT M_y AT TOP OF WALL

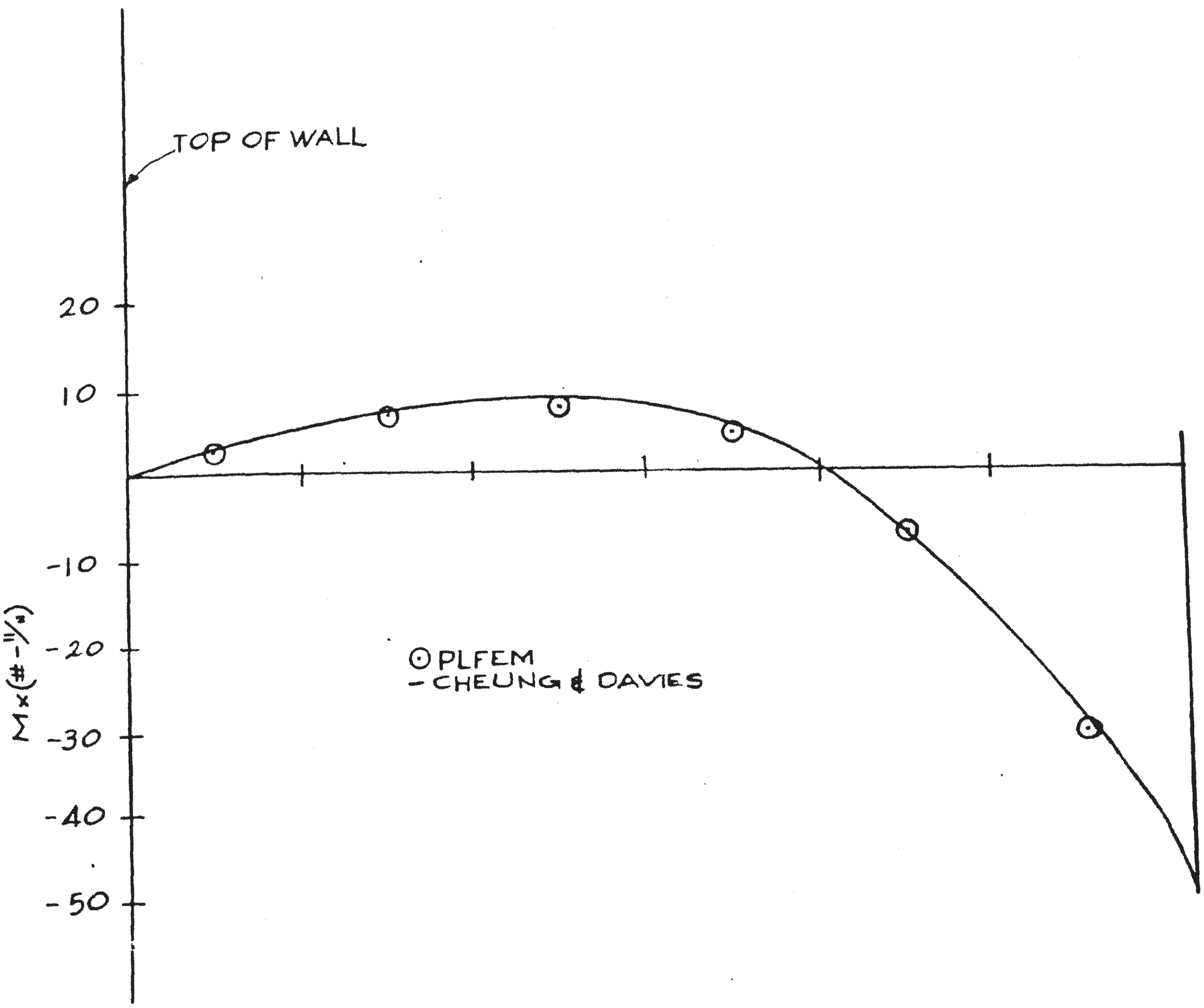

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FIGURE F.39-1 UNITS 1 & 2 MOMENT M_x ALONG CROSS SECTION OF LONG WALL

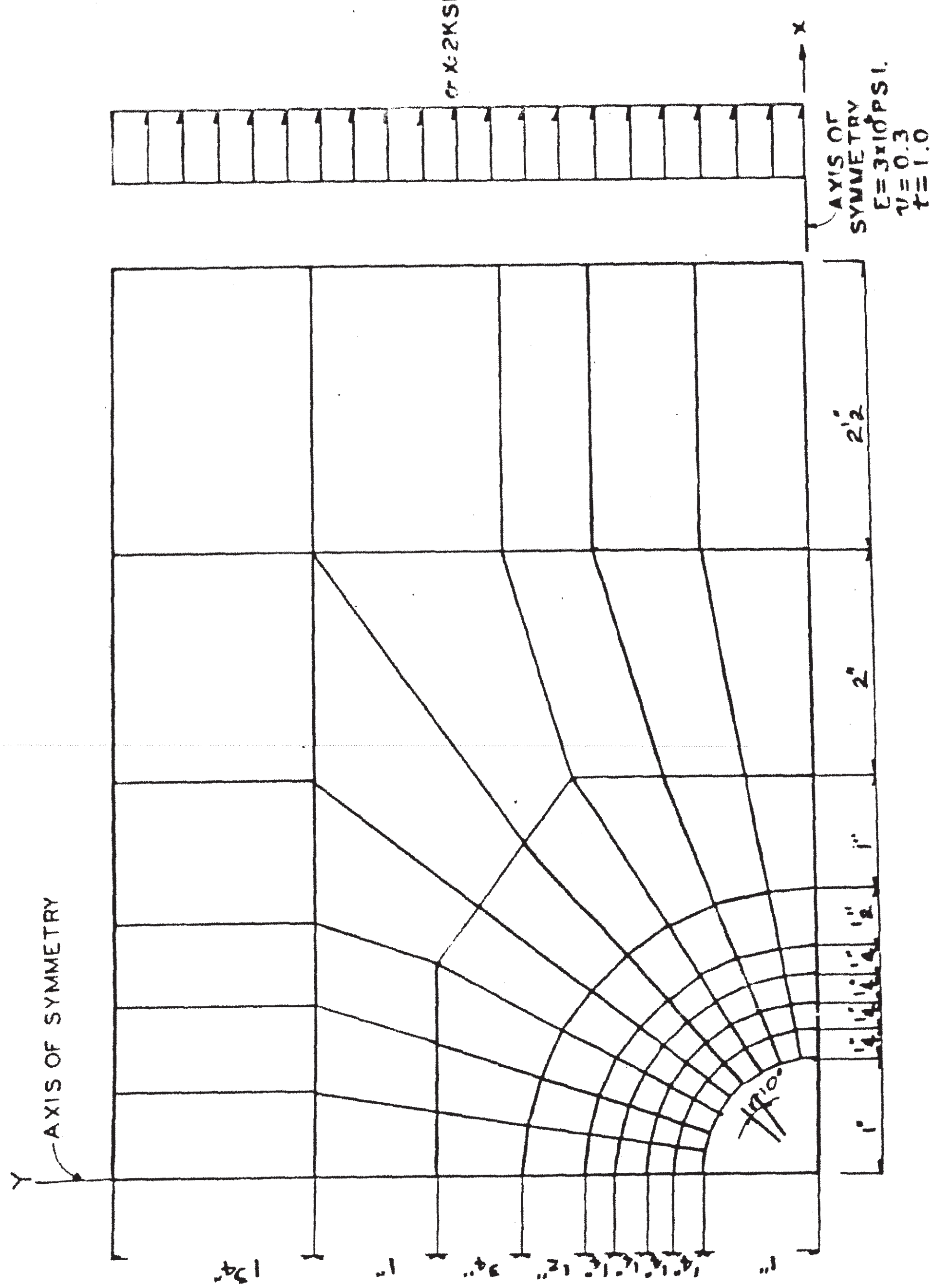

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FIGURE F.40-1 UNITS 1 & 2 PLATE WITH CIRCULAR HOLE UNDER UNIFORM TENSION

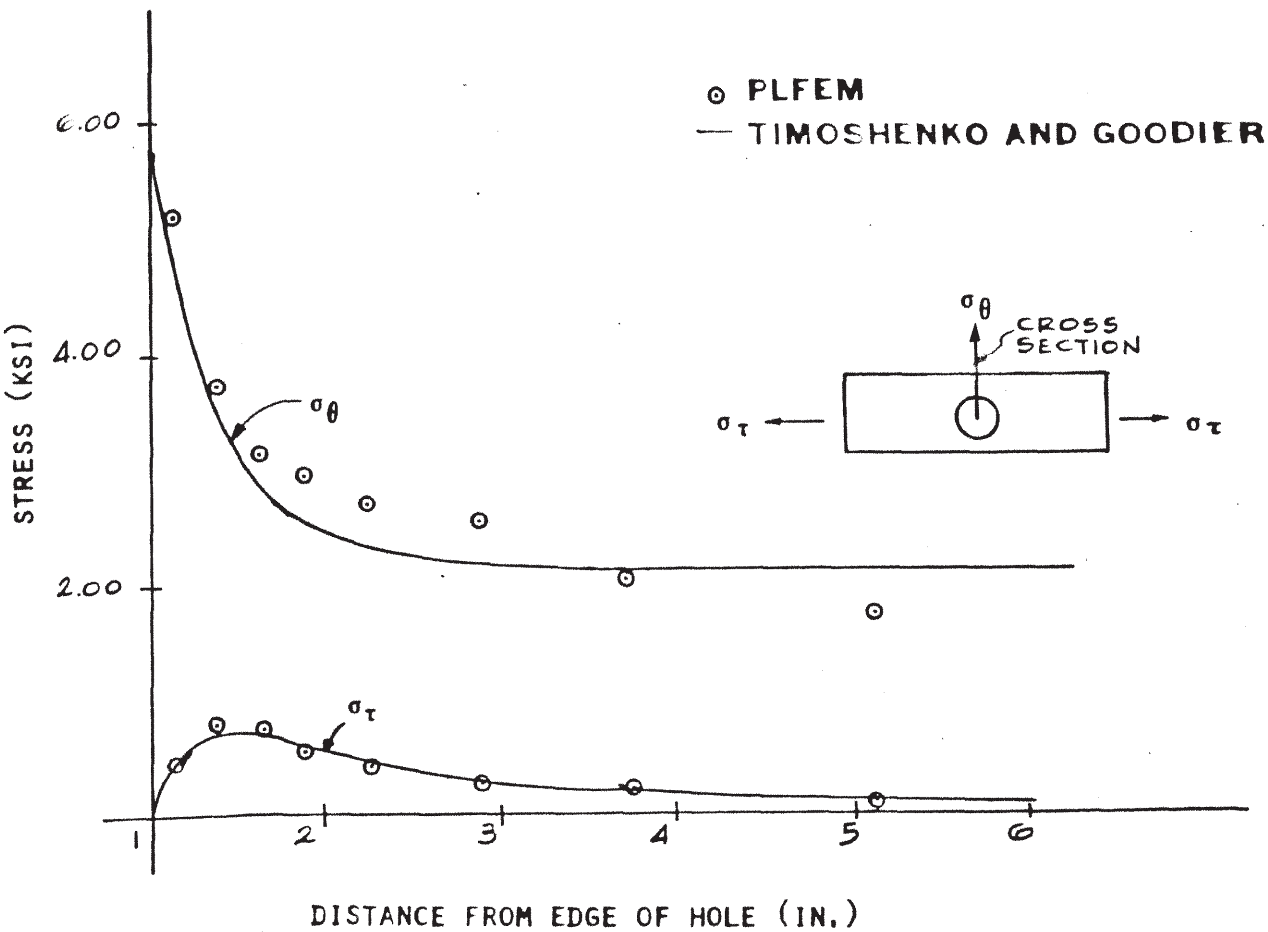

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FIGURE F.41-1 UNITS 1 & 2 STRESSES IN PLATE WITH CIRCULAR HOLE UNDER UNIFORM TENSION

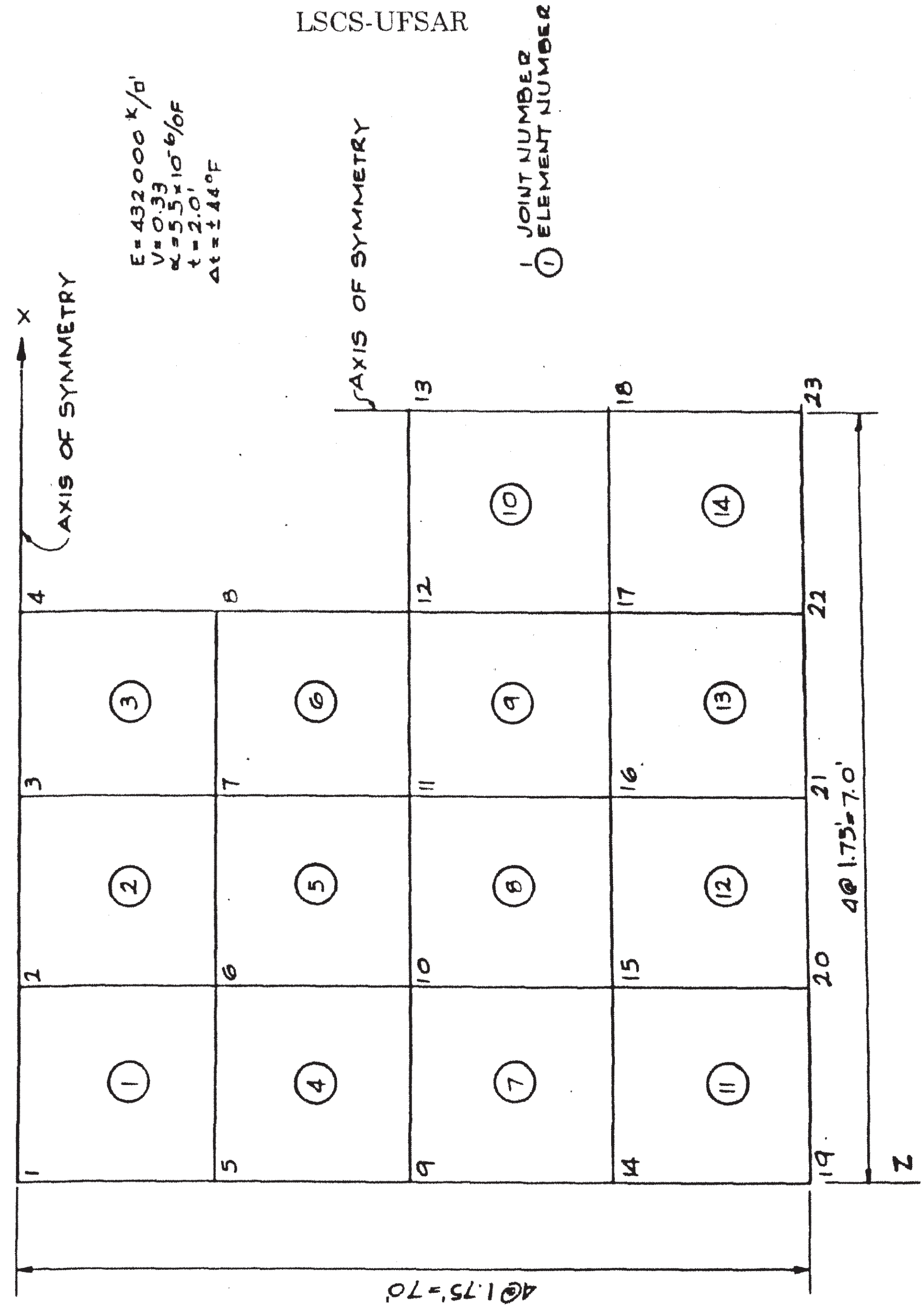

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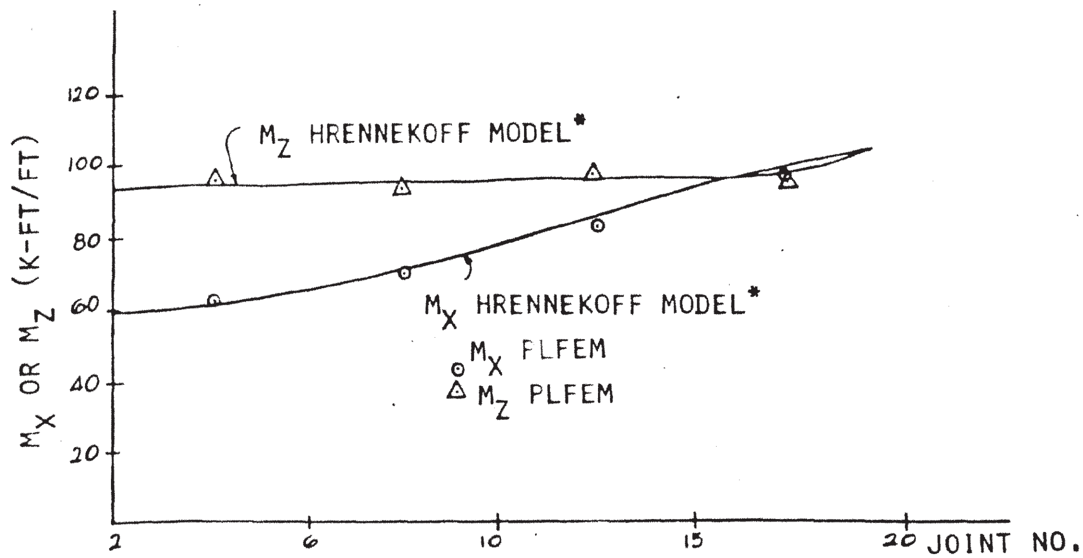
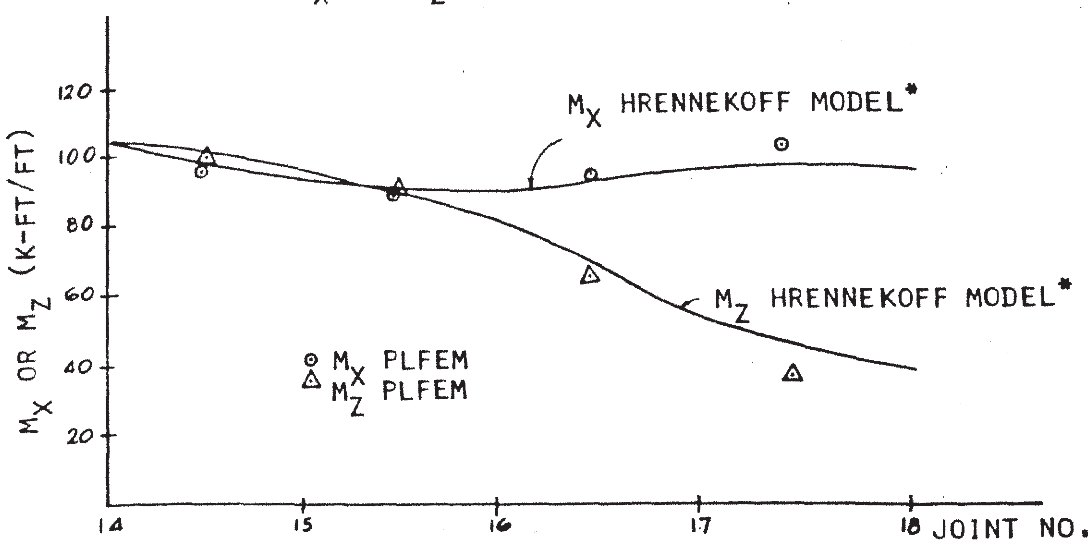
FIGURE F.42-1

UNITS 1 & 2
SQUARE PLATE WITH RECTANGULAR HOLE
SUBJECTED TO TEMPERATURE VARIATION



M_X AND M_Z ALONG THE 2-20 LINE



M_X AND M_Z ALONG THE 14-18 LINE

* HRENNEKOFF MODEL BASED ON A FRAMEWORK ELEMENT 0.875'SQUARE

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FIGURE F.43-1 UNITS 1 & 2 MOMENTS IN PLATE DUE TO TEMPERATURE VARIATION

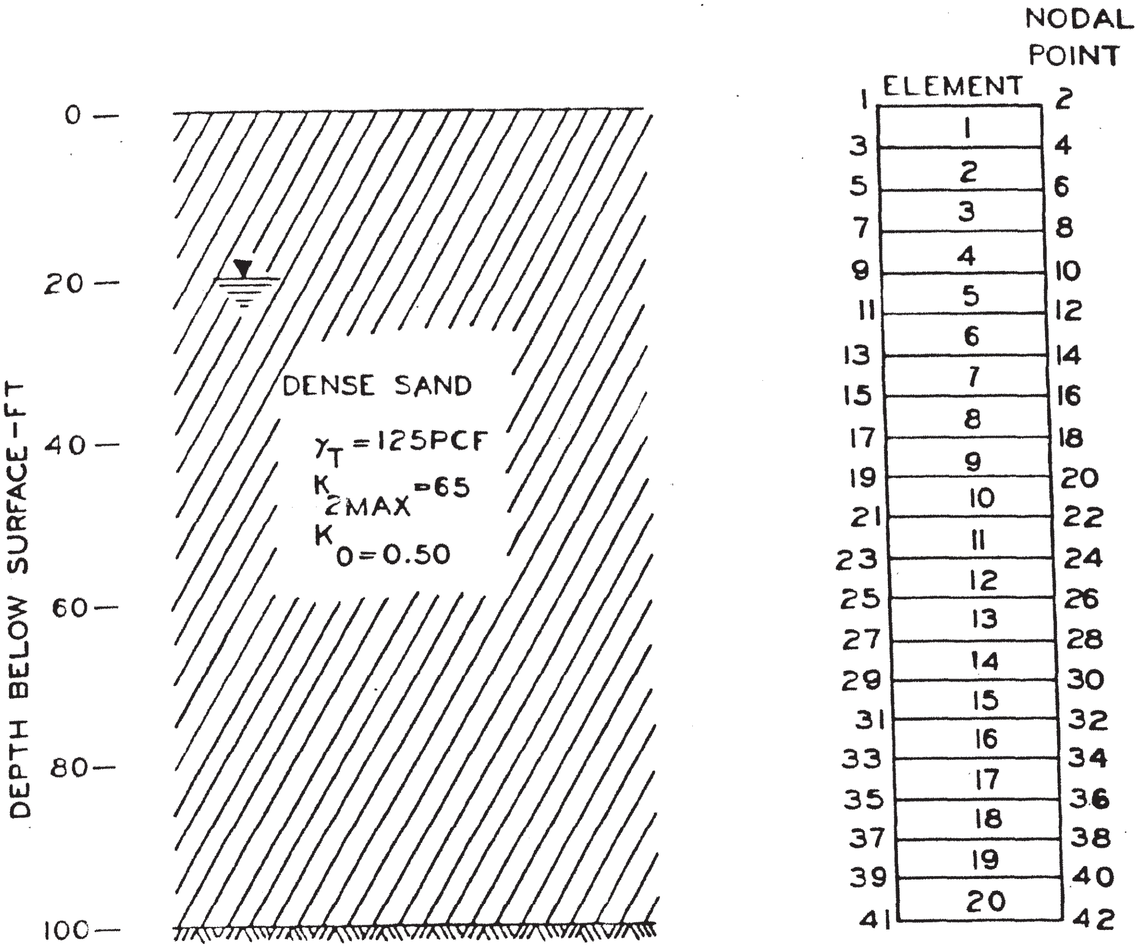

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FIGURE F.44-1

UNITS 1 & 2
SOIL PROFILE AND FINITE ELEMENT
REPRESENTATION USED FOR SAMPLE PROBLEM

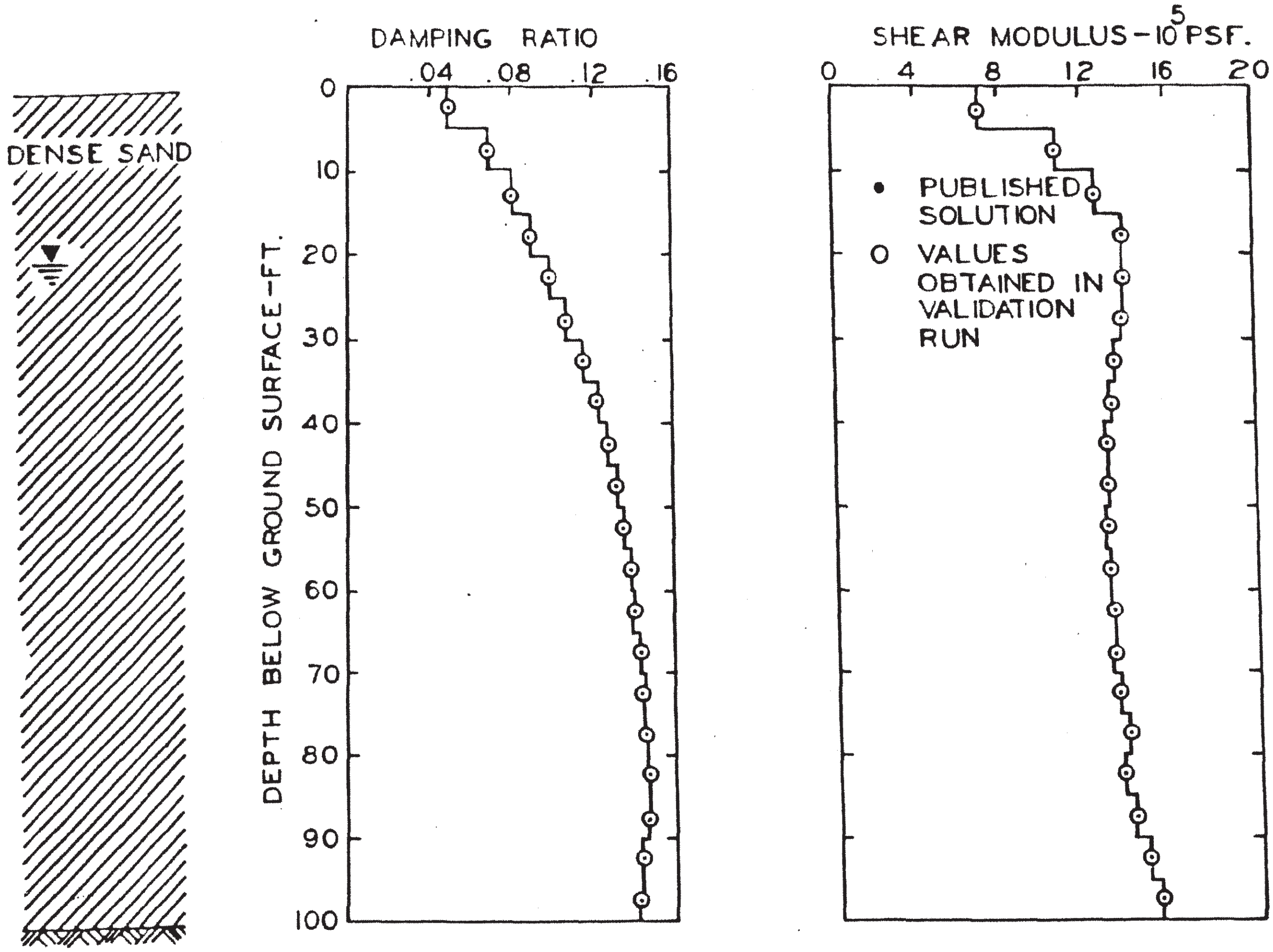

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FIGURE F.45-1

UNITS 1 & 2

STRAIN-COMPATIBLE DAMPING AND MODULUS VALUES USED IN ANALYSIS

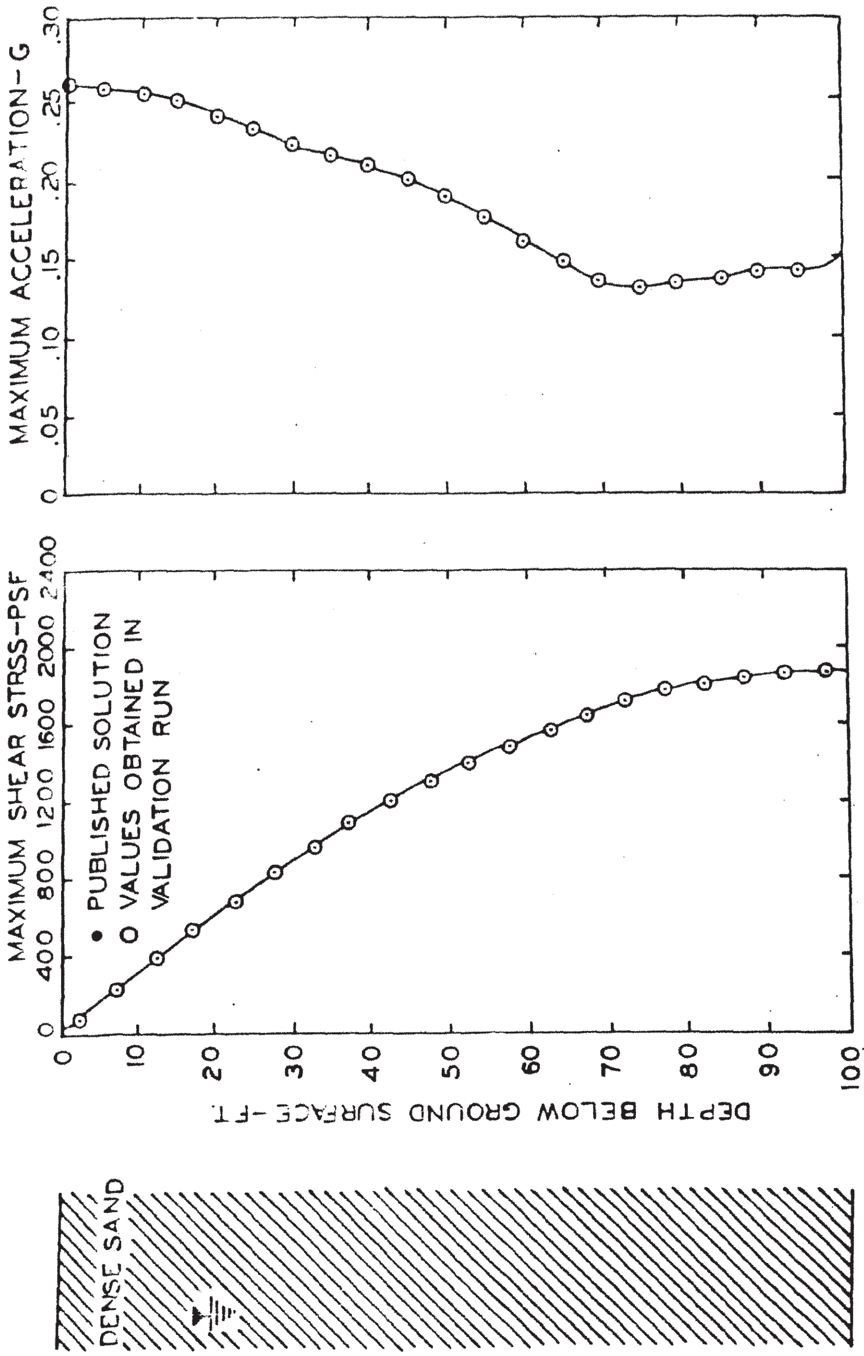

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FIGURE F.46-1

UNITS 1 & 2
DISTRIBUTION OF MAXIMUM SHEAR
STRESSES AND ACCELERATIONS

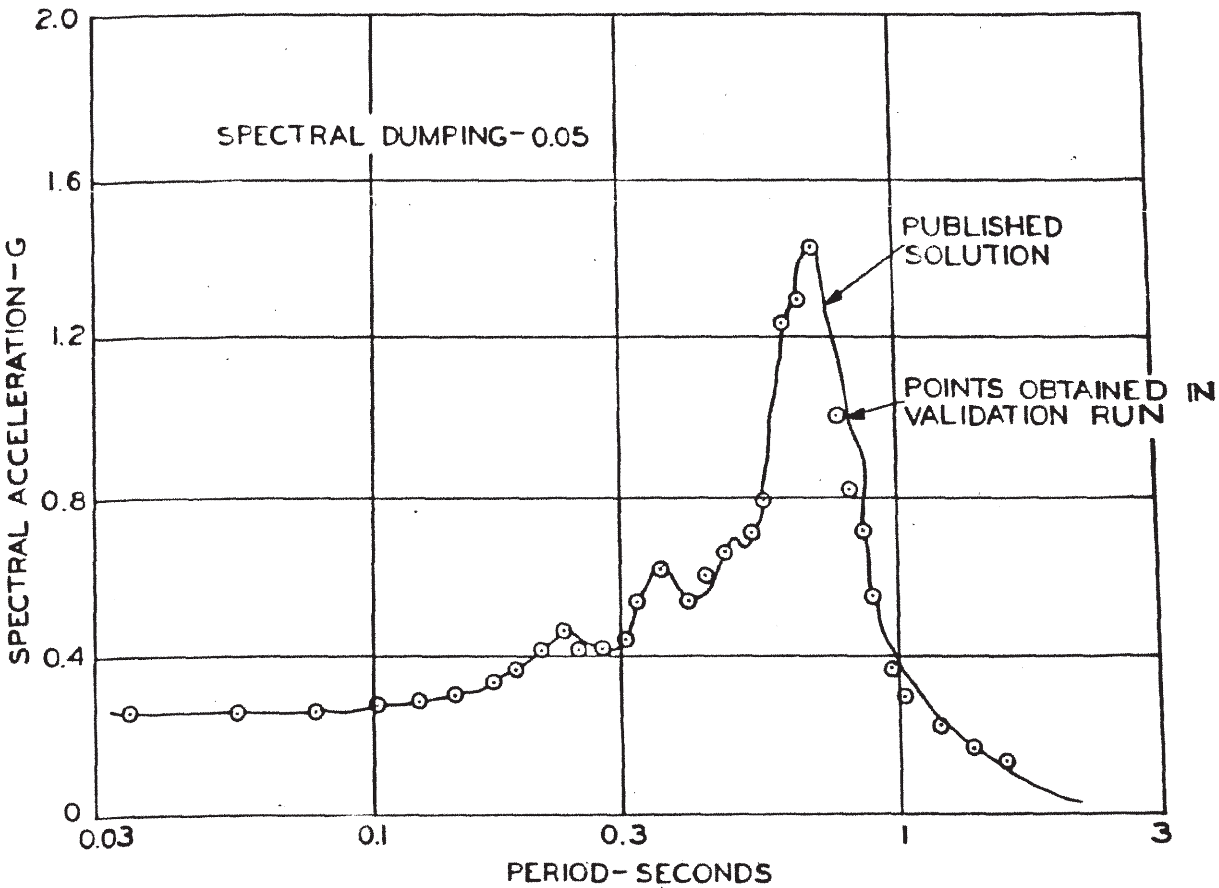

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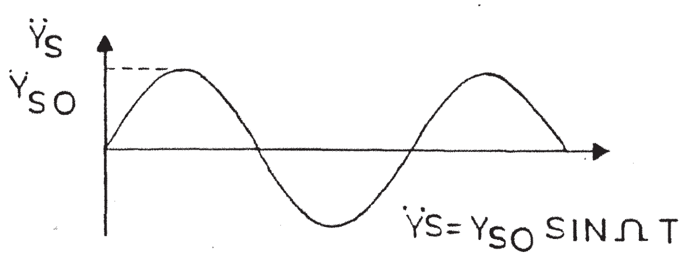
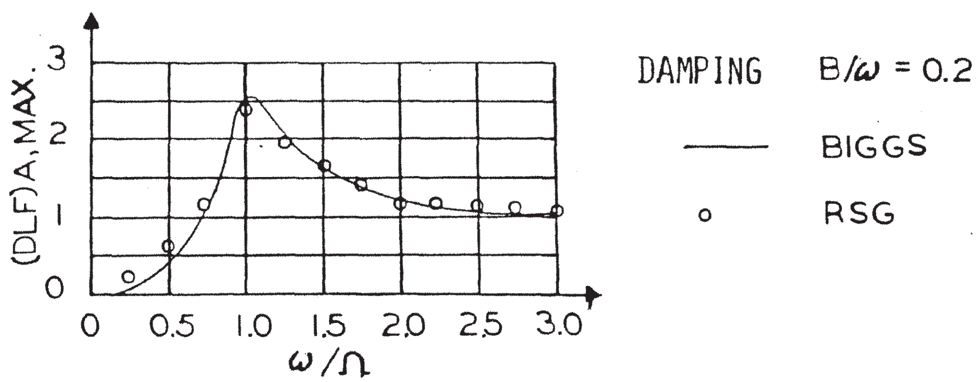
FIGURE F.47-1 UNITS 1 & 2 ACCELERATION SPECTRA FOR COMPUTED SURFACE MOTIONS



ACCELERATION TIME HISTORY



RESPONSE SPECTRUM

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FIGURE F.48-1 UNITS 1 & 2 VALIDATION FOR A ONE DEGREE-OF-FREEDOM DAMPED SYSTEM (RSG)

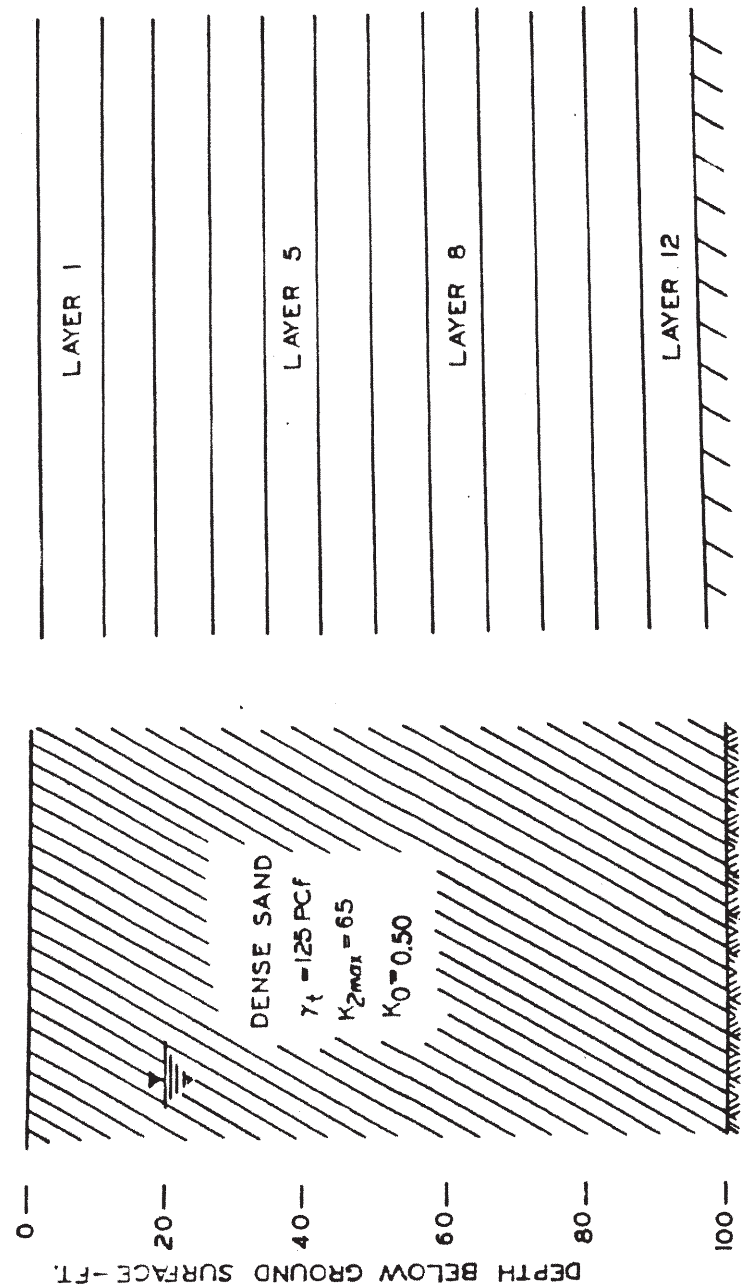

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FIGURE F.49-1 UNITS 1 & 2 SOIL PROFILE AND LAYERED REPRESENTATION USED FOR SAMPLE PROBLEM

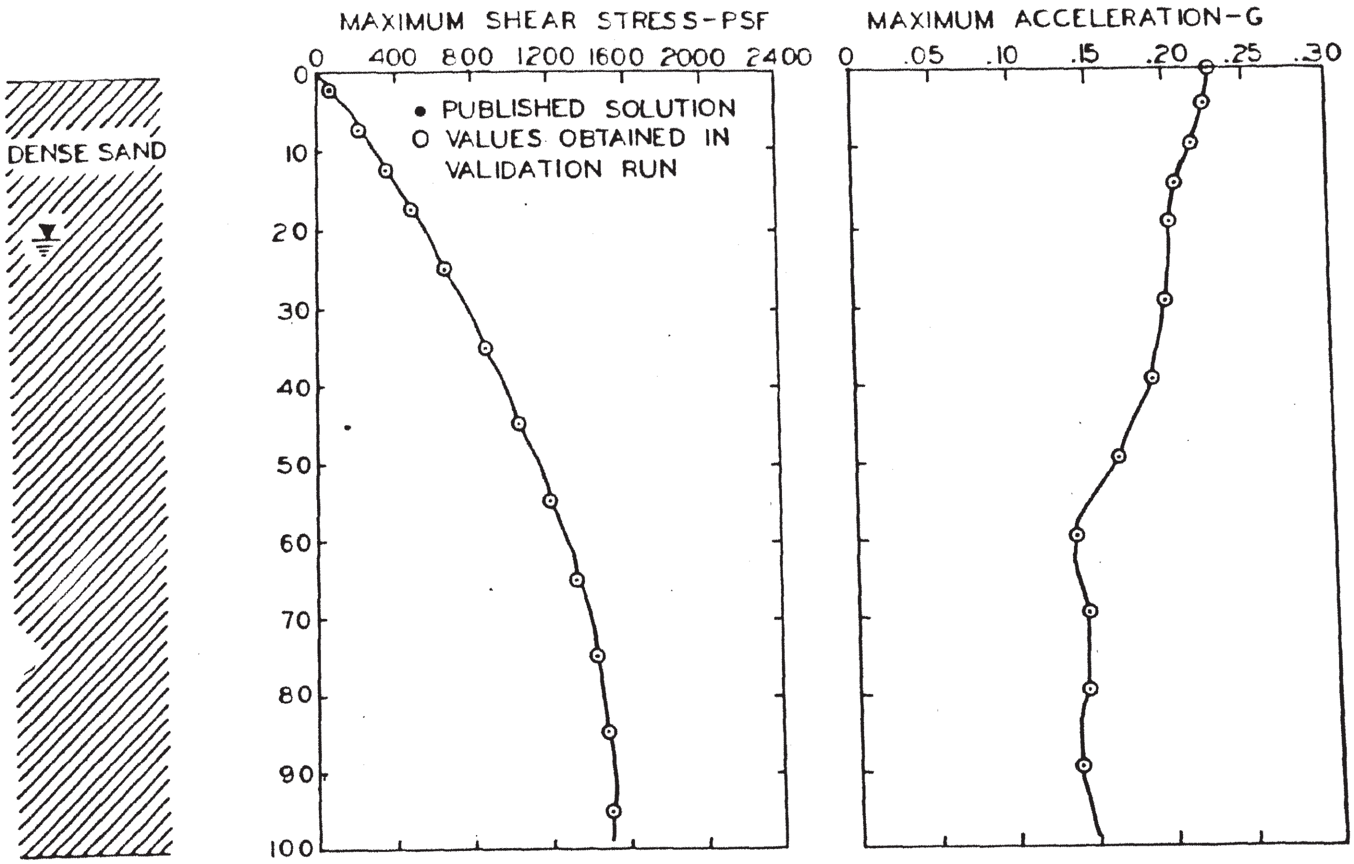

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FIGURE F.50-1

UNITS 1 & 2
COMPARISON OF SHEAR STRESSES
AND ACCELERATIONS

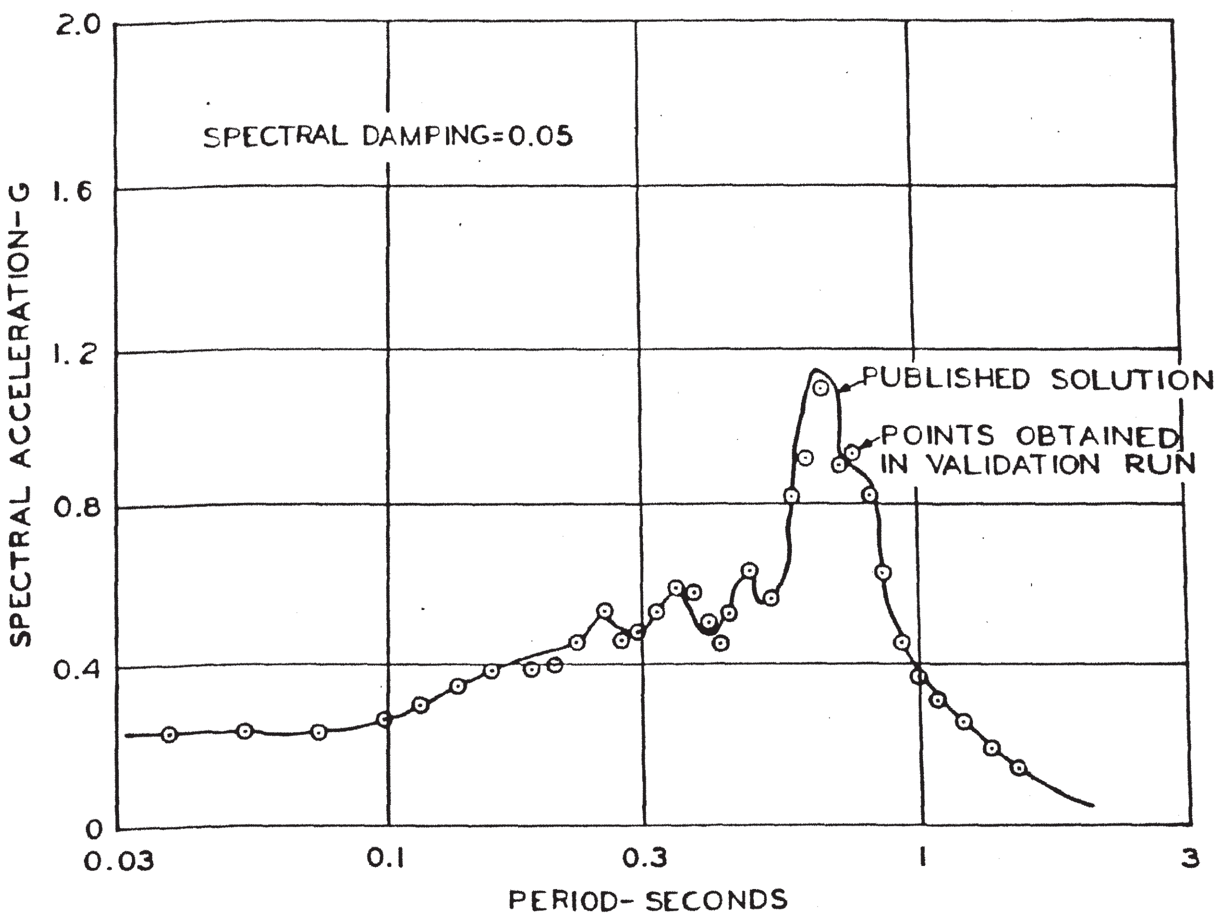

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FIGURE F.51-1 UNITS 1 & 2 COMPARISON OF SPECTRAL VALUES FOR SURFACE MOTIONS

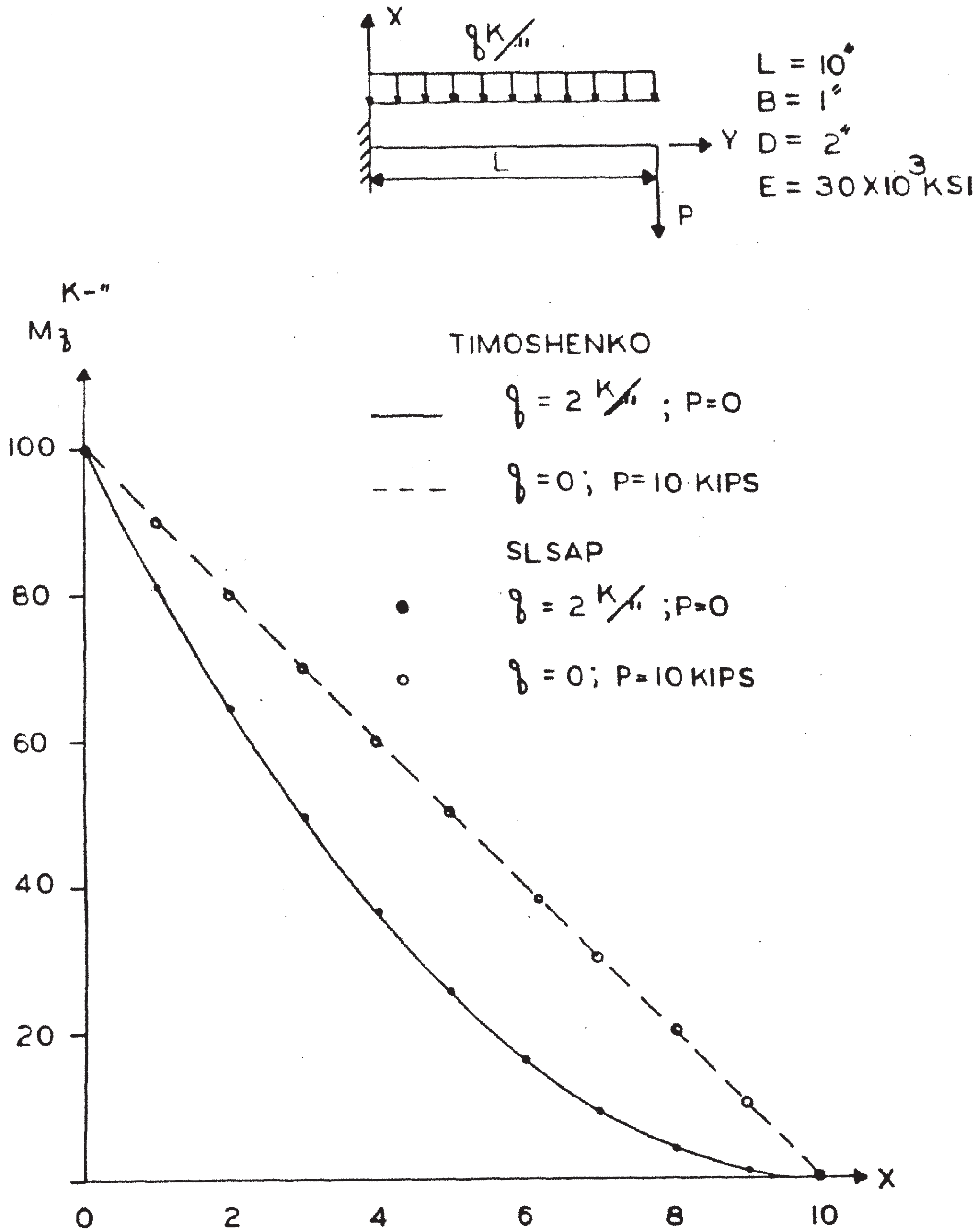

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FIGURE F.52-1

UNITS 1 & 2
BENDING MOMENTS IN A CANTILEVER BEAM

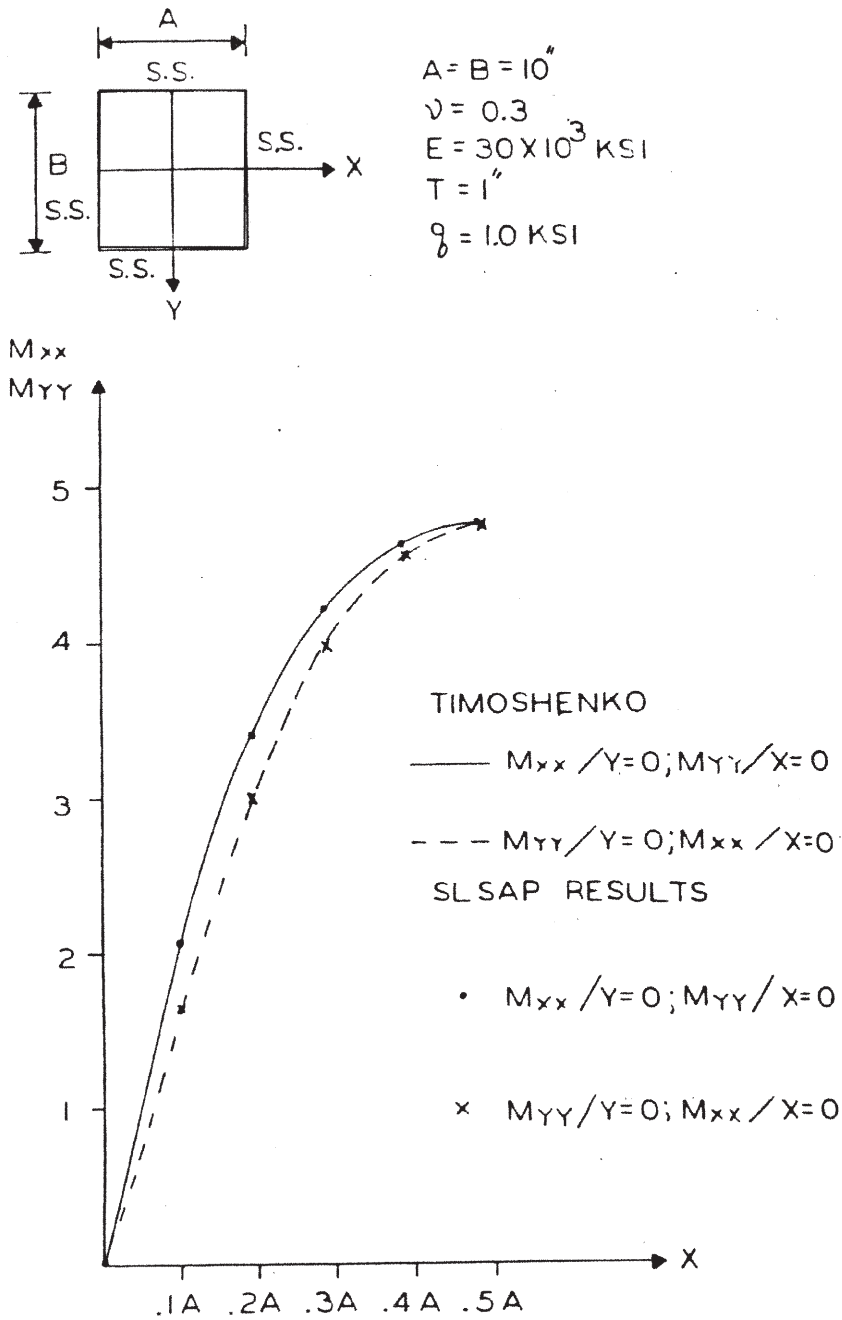

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FIGURE F.53-1

UNITS 1 & 2
BENDING MOMENTS IN A SIMPLY
SUPPORTED PLATE

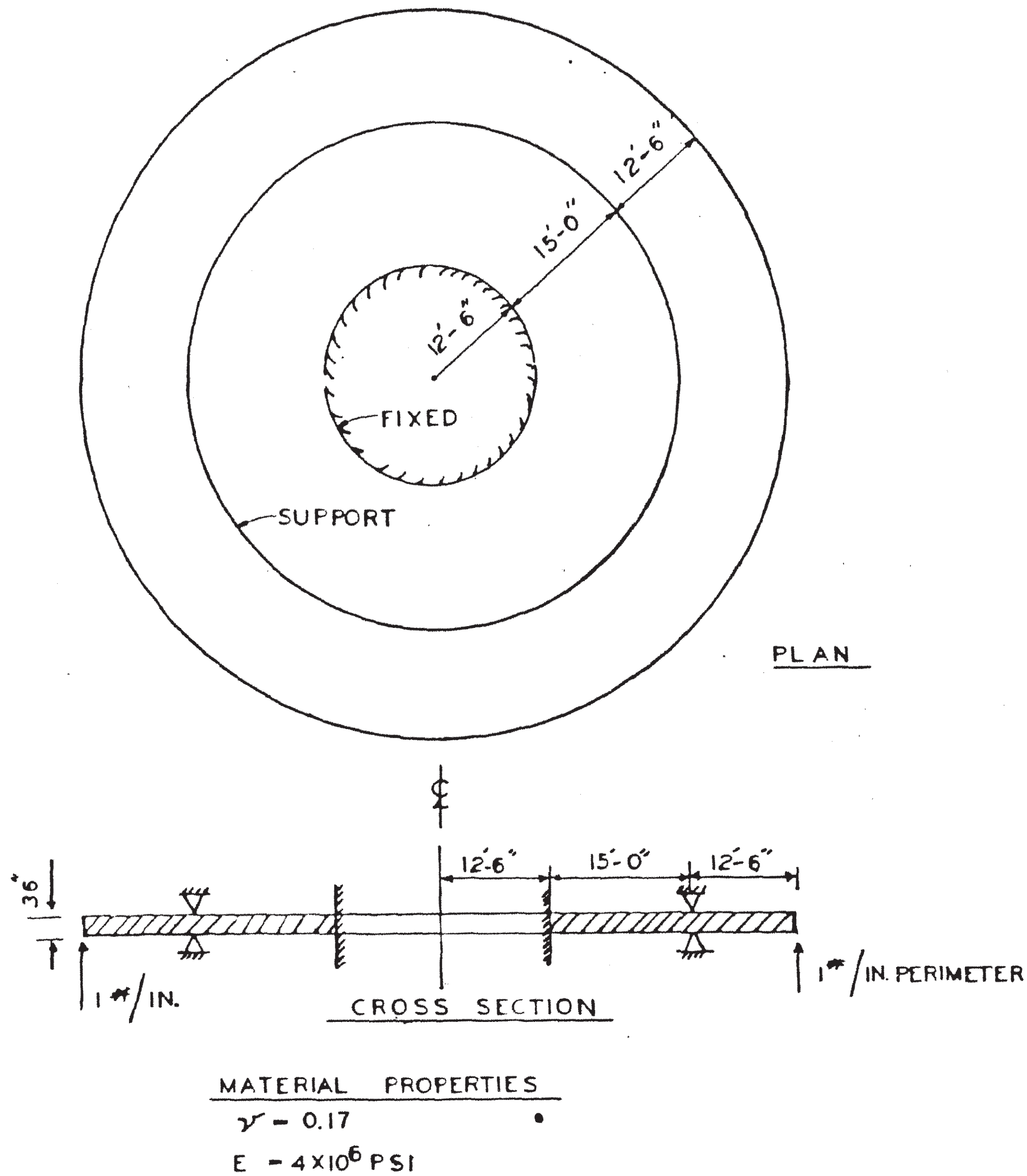

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FIGURE F.54-1

UNITS 1 & 2
CIRCULAR PLATE FOR SOR-III EXAMPLE

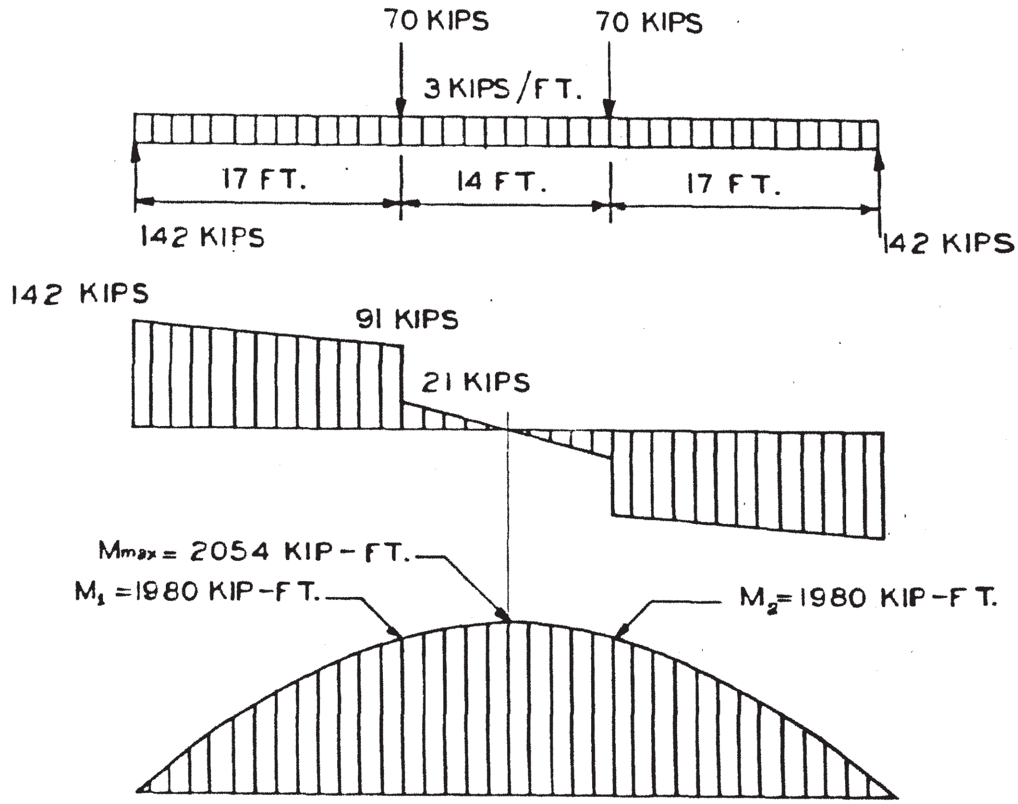

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FIGURE F.55-1

UNITS 1 & 2
SHEAR AND MOMENT DIAGRAMS

APPENDIX G

REACTOR RECIRCULATION SYSTEM

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DRAWINGS CITED IN THIS APPENDIX*

DRAWING*	SUBJECT
M-93	Nuclear Boiler & Reactor Recirculating System, Unit 1
M-139	Nuclear Boiler & Reactor Recirculating System, Unit 2
M-2093	Reactor Recirculating System “RR”, Unit 1
M-2139	Reactor Recirculating System “RR”, Unit 2
1E-1(2)-4205	Schematic Diagrams for Reactor Recirculation System “RR”

* The listed drawings are included as "General References" only; i.e., refer to the drawings to obtain additional detail or to obtain background information. These drawings are not part of the UFSAR. They are controlled by the Controlled Documents Program.

APPENDIX G - REACTOR RECIRCULATION SYSTEM

G.1 INTRODUCTION

G.1.1 Purpose

The purpose of this appendix is to describe the design of the La Salle County Station (LSCS) recirculation system in sufficient detail to show compliance with NRC nuclear reactor design requirements.

G.1.2 Discussion of Design Changes

LSCS has an improved recirculation system design that retains many of the features of the BWR/3 and BWR/4 product lines. The LSCS design is typical of the BWR/5 and /6 generation of plants. In order to put this design in perspective, the design changes made with the introduction of the BWR/5 are compared to the BWR/3 and /4 design in the following.

G.1.2.1 Jet Pumps

Jet pumps are common to BWR/3, /4, /5, and /6. The design of the jet pump was changed with introduction of the BWR/5 to implement a more efficient design which allowed a decrease in pump motor size and horsepower. The major difference in the jet pump design is the five-nozzle jet configuration, where five jets induce driving flow through each jet pump throat rather than the single jet typical of the BWR/3 and /4 generation.

G.1.2.2 Flow Control

The flow control design was changed with the BWR/5 to improve the response and load-following capabilities of the previous design. The control concept changed from recirculation pump speed control to direct recirculation flow control using a valve in each external recirculation loop. The salient design features are as follows:

BWR/3 and /4

Variable speed pump and motor

Drive motor - fluid coupler - generator - pump motor:

> The M-G varies the line frequency supplied to the pump motor. The fluid coupler provides slip between the motor and generator, thereby governing the output frequency of the generator. The control design incorporates both manual and automatic features, using feedback loops of generator speed and turbine load demand.

BWR/5 and /6

Constant speed pump and motor

Flow control valve (FCV):

> Varies the flow directly by increasing or decreasing the pressure drop in the recirculation loop.

Hydraulic actuator - power unit:

The FCV position is varied by a hydraulic cylinder mounted on the valve. Hydraulic lines lead from the cylinder to a power unit located outside the primary containment. The power unit controls the line pressure and the fluid flow to or from either side of the actuator piston, thereby changing flow control valve position.

Electrical Control:

The remainder of the system incorporates manual features using feedback loops of FCV position and velocity. Automatic load following capabilities have been removed.

G.1.2.3 Pump Motor

The coastdown characteristic of the recirculation system after the power supply is tripped is important to mitigate some transient and accident events. The factor that governs this coastdown for the BWR/5 design is the inertia of the pump impeller and motor rotor whereas the BWR/3 and /4 design was governed by the pump-motor inertia plus the inertia of the motor-fluid coupler-generator set. Since the MG set was deleted from the BWR/5 design, the inertia of the pump motor rotor was increased by approximately 20% in order to obtain a coastdown characteristic similar to a BWR/3 and /4.

G.1.2.4 Low-Frequency Motor-Generator Set

Since feedwater flow is low during plant heatup and low thermal power operation, there is little net positive suction head (NPSH) available to the recirculation pumps, flow control valves, and jet pumps. The NPSH available downstream of the recirculation pump is a function of the NPSH available to the pump and the heat addition to the fluid due to pump inefficiency. The LFMG set operates the pump at 25% of normal speed to minimize pump heat input, thereby avoiding cavitation in the loop during plant heatup and low-power operation. At high thermal power levels, there is enough feedwater subcooling to allow normal speed operation of the pump.

G.1.2.5 Cavitation Protection

The introduction of the flow control valve and five-nozzle jet pumps made it necessary to provide different cavitation protection interlocks, since these components produced different cavitation characteristics than the BWR/3 and /4 design. The design features are as follows:

BWR/3 and 4

Cavitation is prevented by a feedwater flow interlock set at 20% to 30% of normal flow, which reduces pump speed to below 30% of rated. The setpoint is determined by the thermal power level where cavitation would occur when the recirculation pump was operated at rated flow. This is conservative for all lower flows, since NPSH requirements decrease at lower flows, while available NPSH remains essentially the same for any given thermal power level.

BWR/5 and /6

The cavitation characteristics (thermal power level at which cavitation will occur for a given flow) of the jet pumps and recirculation pump are similar to a BWR/3 and /4 except that jet pump cavitation occurs at higher thermal power levels. Protection is provided by measuring the available subcooling and tripping the pumps to 25% speed when there is inadequate subcooling. A temperature element in the loop suction line and a pressure transmitter measuring vessel pressure in conjunction with a function generator measure the amount of subcooling available to the system.

When the recirculation pump is operating at 100% speed, the flow control valve will cavitate at low flows due to the large pressure drop across the valve. Cavitation protection is provided by a feedwater flow interlock that trips the pumps to 25% speed when there is not enough feedwater flow subcooling to allow operation at low flow control valve positions.

G.1.2.6 Piping

The recirculation system discharge piping configuration was changed to allow room for the flow control valve between the recirculation pump and discharge isolation valve. Pipe diameter was also reduced to minimize drywell space requirements.

G.1.2.7 Recirculation Pump Flow Measurement

The recirculation loop flow element is used to provide indirect core flow indication for the flow bias scram system and to indicate pump performance and jet pump drive flow. In general, these functions do not require high measurement accuracy, although repeatability is required. The flow bias scram system requires a signal that is proportional to pump flow. The signal is used as an indicator of core flow. The proportionality constant (calibration coefficient - pressure drop versus flow) is unimportant as long as that constant does not change, that is, the element is repeatable. Flow is controlled through a Manual/Auto (M/A) station. Individual FCV position demand is used as feedback for the ganged position control setpoint. For ganged position control, the common position setpoint is fed to each loop where it is compared to actual position to develop the FCV position demand signal.

These functional requirements are satisfied by an elbow flow element where the pressure from the inside to the outside of the elbow is proportional to flow. Consequently, the flow nozzle in the BWR/4 design was replaced by an elbow pressure tap flow element in the recirculation pump suction line for the BWR/5 and /6 design.

G.1.2.8 Recirculation Pump Trip (RPT)

The recirculation pumps are tripped for many reasons, among which are low NPSH, anticipated transient without scram (ATWS) mitigation, and electrical faults such as short circuits. Only one trip function is currently required to be safety grade, and that function is given the name RPT. The purpose of RPT is to mitigate the thermal consequences of the turbine trip and generator trip transients by tripping the recirculation pumps early in the event, producing rapid pump flow coastdown and additional core voiding, which results in a core reactivity reduction. This system is linked to the reactor protection system (RPS) such that both a scram and a pump trip occur when the turbine stop valves start to close and when turbine control valve fast closure occurs. Both scram and RPT are bypassed at low thermal power levels.

Since only one power source is available to a BWR/4 pump motor, RPT trips the pumps completely off. The BWR/5 activates the 25% speed source (the low-frequency M-G set) when the pump has coasted down to that speed.

G.1.2.9 Core Flow Measurement

The core flow measurement system is unchanged from the BWR/4 design. For BWR/5 and /6, as an operating convenience, individual jet pump pressure drop signals are fed to the process computer to calibrate the system and obtain the jet pump integrity surveillance data required by the Technical Specifications.

G.1.2.10 Recirculation System Operation

Due to the changes described in Subsections G.1.2.2 and G.1.2.5, the startup and operation of the BWR/5 recirculation system is significantly changed from previous systems. As a result, new control interlocks were necessary to prevent significant transients, equipment damage, or unnecessary scrams. Electrical interlocks were installed between the LFMG set and the normal power supply to prevent damage to the LFMG set and on the flow control valve to prevent cavitation damage. These interlocks also protect against flow-increase transients when starting the system or transferring to the normal power supply.

G.2 RECIRCULATION SYSTEM DESIGN DESCRIPTION

G.2.1 Plant Configuration Data

G.2.1.1 Recirculation System Configuration

G.2.1.1.1 Description

The function of the recirculation system is to circulate reactor coolant through the core and to provide a means of controlling reactor power output by varying the rate of flow of the coolant. In this system, water is circulated through the core by the use of jet pumps within the reactor pressure vessel. The arrangement of equipment and the flow paths are shown schematically in Figures G.2.1-1 to G.2.1-5.

Driving flow for the jet pumps is drawn from the bottom of the annular region surrounding the reactor core into two symmetrically oriented piping loops. Each loop contains a single-stage, vertical, centrifugal pump installed at an elevation several feet below the reactor vessel to provide the required net positive suction head (NPSH). Gate valves are located in both suction and discharge piping. The pump discharge flow return line connects to a manifold formed into a near semicircle to fit close to the reactor vessel. Smaller, vertical pipes from the manifold (one for every two jet pumps) return the high-pressure flow to the reactor at nozzles distributed circumferentially on the vessel wall. Inside the reactor vessel, the circuit is continued through internal risers which pass upward in the annular region (downcomer) between the core shroud and the vessel wall. Each internal riser is terminated with a special fitting which distributes the flow to the nozzle-supply passages of twin jet pumps.

Reactor recirculation system equipment that is part of the reactor pressure boundary is designed as Seismic Category I equipment. As such, it is designed to resist sufficiently the response motion at the installed location within the supporting structure for the safe shutdown earthquake. The pump is assumed to be filled with water for the analysis. Vibration snubbers located at the top of the motor and at the bottom of the pump casing are designed to resist the horizontal reactions.

The recirculation piping, valves, and pumps are supported by hangers to avoid the use of piping expansion loops that would be required if the pumps were anchored. In addition, the recirculation loops are provided with a system of restraints designed so that reaction forces associated with a split or circumferential pipe break do not jeopardize drywell integrity. This restraint system provides adequate clearance for normal thermal expansion movement of the loop.

The recirculation system piping, valves, and pump casings are covered with thermal insulation having a maximum average heat transfer rate of 65 Btu/hr-ft^2 of insulation extension surface with the system at rated operating conditions. This heat loss includes losses through joints, laps, and other openings that may occur in normal application.

The insulation is all-metal reflective type with one exception as described in the next paragraph. It is prefabricated into components for field installation. Removable insulation is provided at various locations to permit periodic inservice inspection of the equipment.

An alternate insulation detail allows a limited amount of removable FOAMGLAS® insulation to be installed around the recirculation flow control valve extension shaft under the metal reflective insulation on the valve. This material is intended to limit convective heat flow from inside the metal reflective insulation out around the valve extension shaft to limit the localized heat degradation of the non-safety related rotary variable differential transmitter associated with each valve. Use of the supplemental FOAMGLAS® is limited to this specific application and is not credited for maintaining the maximum average heat transfer rate.

The recirculated coolant consists of saturated water from the steam separators and dryers that has been subcooled by incoming feedwater. This water passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high-pressure flow into an external manifold from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the driven flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the driving flow. The flows, both driving and driven, are mixed in the jet pump throat section and result in partial pressure recovery. The balance of recovery is obtained in the jet pump diffusing section.

During startup and low-power operation, the recirculation pumps are run at 25% of rated speed to provide forced circulation. Two 15-Hz motor-generator sets provide the power to the pump motors during these periods. The pump is first started with the flow control valve in its minimum position (with the flow control valve at its minimum position, approximately 34% of rated pump flow will pass with the pump operating at high speed). The main (60-Hz) power source brings the pump up to full speed and is then tripped. During the following coastdown, the low-frequency motor generator (LFMG) set starts and energizes the pump at 25% speed. The flow control valve is then opened to provide at least 20% pump flow.

The allowable heatup rate for the recirculation pump casing is the same as for the reactor vessel. If one loop is shut down, the idle loop can be kept hot by leaving the discharge and suction gate valve open; this permits the active jet pump head to cause reverse flow in the idle loop.

Because the removal of the reactor recirculation gate valve internals would require unloading the core, or the use of alternate isolation devices such as the RR Suction plug or Jet Pump plugs, the objective of valve trim design is to minimize the need for maintenance of the valve internals.

When the pump is operating at 25% speed (15 Hz), the head provided by the elevation of the reactor water level above the recirculation pump and the jet pumps is sufficient to provide the required NPSH. When the pump is operating at 100% speed (60 Hz), most of the NPSH is supplied by the subcooling provided by feedwater flow. An accurate temperature element is provided in the recirculation line to measure recirculation loop temperature. A pressure transmitter measuring vessel pressure in conjunction with a function generator is provided to measure vessel water saturation temperature. The difference between these two readings is a direct measurement of the subcooling. If the subcooling falls below 10.1° F, the 60-Hz power supply is tripped to the 15-Hz supply to prevent cavitation of the recirculation pump and jet pump.

The transition of 100% speed is made after thermal power is sufficient to provide enough feedwater subcooling to prevent flow control valve cavitation. At this point the flow control valves are moved to the minimum positions and the pumps brought to 100% speed. The 25% to 100% speed transition is prevented by interlocks requiring sufficient thermal power and the control valve in minimum position.

During preparation for hydrostatic tests, the nuclear system temperature must be raised above the vessel nil ductility transition temperature limit. The vessel is heated by operating the residual heat removal pumps and by core decay heat. When NPSH is adequate, the recirculation pumps may be run at 100% speed to provide additional heat.

Each recirculation pump is equipped with mechanical shaft seal assemblies. Each of these assemblies consists of two seals built into a cartridge that can be readily replaced without removing the motor from the pump. Each individual seal in the cartridge is designed for pump design pressures so that any one seal can adequately limit leakage in the event that the other seal should fail. The pump shaft passes through a breakdown bushing to reduce leakage in the event of a gross failure of both shaft seals. The cavity temperature and pressure drop across each individual seal is monitored.

Each recirculation pump motor is a vertical solid-shaft, totally enclosed, air/water-cooled induction motor. The combined rotating inertias of the recirculation pump and motor determine the coastdown for both the loss-of-coolant accident and the turbine-generator trip transients. This inertia requirement is met without a flywheel.

The pump discharge flow control valve throttles the discharge flow of the pump. The recirculation loop flow rate can be changed rapidly within the expected flow range in response to rapid changes in system demand. The maximum required valve actuator stroking rate is 10% to 11% of full stroke per second.

A design objective for the recirculation system equipment is to provide units that will not require removal from the system for rework or overhaul. Pump casings and valve bodies are designed for a 40-year life and are welded to the pipe. The pump drive motor, impeller, wear rings, and flow control valve internals are designed for as long a life as is practical. Pump mechanical seal parts and the valve packing are expected to have a life expectancy which affords convenient replacement during the refueling outages.

The recirculation system piping is of all-welded construction and is designed and constructed to meet the requirements of either the ASME Section III Code or the ANSI B31.7 Code.

G.2.1.2 Thermal, Hydraulic, and Physical Data

G.2.1.2.1 Recirculation Loop Piping

G.2.1.2.1.1 Piping Description

a. Number of loops - 2

b. General arrangement - See Figures G.2.1-1 through G.2.1-5

c. Pump suction line nominal size - 24 in.

d. Pump discharge line nominal size - 24 in.

e. Ring header nominal size - 16 in.

f. Number of risers - 10

g. External riser nominal size 12 in.

h. Internal riser nominal size - 10 in.

G.2.1.2.1.2 Operating Conditions

a. Nominal loop velocity (at pump rated flow) - 41.5 ft/sec

G.2.1.2.2 Valves

G.2.1.2.2.1 Flow Control Valves

G.2.1.2.2.1.1 Valve Description

a. Nominal size - 24 in.

b. Type - Ball

c. Actuator - Hydraulic

G.2.1.2.2.1.2 Valve Operating Conditions

a. Valve wide-open flow - 45,600 gpm

b. Valve wide-open pressure drop - 22.7 psi

c. Fluid specific volume - 0.0212 ft^3/lb

d. Valve C_v - 7770

e. Stroke rate (% of stroke) - 10 ± 1% per second

G.2.1.2.2.1.3 Definitions

Closed Position - the fully closed position of the valve ball. This is a reference position for identifying valve ball angular position. Physically, the valve cannot reach this position with the actuator connected.

Minimum Actuator or Zero Position:

The actuator stop. This is the point where the valve pressure drop is largest. At this position, a flow path exists between the valve body and valve seat. This is defined as 0% position.

Minimum Position:

The position of the flow control valve to which the valve must be closed in order for the reactor recirculation pump start/upshift to be permitted to occur. The nominal value of the minimum position is 20%, relative to the zero and 100% valve positions.

Wide Open Position - the actuator stop. This is the point where valve pressure drop is smallest, and is defined as 100% position.

Stroke - the movement between zero and wide open positions of the actuator.

G.2.1.2.2.2 Isolation Valves

G.2.1.2.2.2.1 Suction Line Isolation Valve

a. Nominal size - 24 in.

b. Body type - Gate

c. Actuator - Motor operator

d. Valve wide-open loss coefficient (max) (f L/D) - 0.25

e. Stroke time - 2 min ± 10%

f. Closure pressure differential - 50 psid

G.2.1.2.2.2.2 Discharge Line Isolation Valve

a. Nominal size - 24 in.

b. Body type - Gate

c. Actuator - Motor operator

d. Valve wide-open loss coefficient (max) (f L/D) - 0.25

e. Stroke time - 12"/min ± 10%

f. Closure pressure differential - 400 psid

G.2.1.2.3 Recirculation Pump

G.2.1.2.3.1 Pump Description

a. Rated flow - 47,200 gpm; 17.85 x 10^6 lb/hr

b. Total developed head at rated flow - 805 ft

c. Nominal speed - 1,780 rpm

d. Specific speed - 2,559

e. Nominal low speed - 427 % 18 rpm

f. Type - Vertical centrifugal single stage

g. Seals - Mechanical shaft

G.2.1.2.3.2 <u>Operating Conditions</u>

a. Water temperature - 534° F

b. Pump suction static pressure - 1,023 psia

c. Suction velocity at rated flow - 41.5 ft/sec

G.2.1.2.3.3 <u>Performance at Normal Speed</u>

a. Continuous flow capability (min) - 20-115% of rated

b. Head-capacity characteristic type - See Figure G.2.1.-6

c. Minimum pump efficiency (rated flow) - 87%

d. Pump brake horsepower (rated flow) - 8,340 hp

e. Maximum pump brake horsepower - 8,900 hp

f. NPSH required at rated flow (see Figure G.2.1-6) - 115 ft

G.2.1.2.3.4 <u>Performance at Low Speed</u>

a. Low speed rated flow - 11,330 gpm

b. Head at rated flow - 46 ft

c. Minimum pump efficiency (at rated flow) - 67%

d. Maximum pump brake horsepower -hot - 170 hp

e. Maximum pump brake horsepower -cold - 226 hp

f. Available NPSH (minimum) - 69 ft

G.2.1.2.4 Recirculation Pump Motor

The recirculation pump decoupler has been eliminated from the design. (Note: See the GE report "Analysis of Recirculation Pump Motor Accident Conditions," January 14, 1977.)

G.2.1.2.4.1 Motor Description

a. Type - Totally enclosed air/water cooled

b. Minimum nameplate rating - 8,900 hp

c. Normal speed - 1,780 rpm

d. Rotational inertia - 21,500 lb-ft^2

G.2.1.2.4.2 Operating Conditions

a. Output range at normal speed - 5,310-8,189 hp

b. Output range at low speed - 101-161 hp

c. Starting power range, hot/cold - 96/135 hp

G.2.1.2.4.3 Required Performance

a. Minimum efficiency at rated power - 94%

G.2.1.2.5 Jet Pumps

G.2.1.2.5.1 Description

a. Number of jet pumps - 20

b. Type of pump - Five-jet nozzle

c. 180° bend internal diameter - 7.0 in.

d. Nozzle internal diameter at tip - 1.30 in.

e. Throat internal diameter - 6.40 in.

f. Discharge diameter - 19.0 in.

G.2.1.2.5.2 Design Operating Conditions

a. Driving flow per jet pump - 1.628 x 10^6 lb/hr; 4,307 gpm

b. M-ratio (suction flow/drive flow) - 2.33

c. Universal efficiency - 41.25%

d. Total developed head - 88.2 ft

e. Nozzle velocity - 208 ft/sec

f. Throat velocity - 143 ft/sec

g. Diffuser exit velocity - 16.2 ft/sec

G.2.1.2.5.3 Performance

a. Peak efficiency (design) - 43.36%

G.2.1.2.6 Equipment Interlocks

G.2.1.2.6.1 Pump Cavitation Protection

a. Function

To prevent cavitation damage to either the recirculation pumps or the jet pumps, the system is provided with an interlock to trip the pump motor to low speed if the temperature difference between the vessel water saturation temperature and the recirculation pump inlet temperature is less than the setpoint value. A high accuracy RTD is provided to measure recirculation pump inlet temperature. A pressure transmitter measuring vessel pressure in conjunction with a function generator is provided to measure vessel water saturation temperature.

b. Setpoint adjustment range: 0°-100° F

c. Setpoint: 10.1° F

G.2.1.2.6.2 Flow Control Valve Cavitation Protection

a. Function

1. To prevent flow control valve cavitation, the system is provided with interlocks which prevent operation of the plant at conditions where cavitation will occur.

2. In addition to the pump cavitation interlock, an interlock is provided to trip the pump motors to low speed when the feedwater flow rate is less than the specified setpoint value.

3. An interlock is provided to prevent increasing the pump motor speed above its low speed unless the flow control valve is in the minimum position and the feedwater flow rate is above the setpoint value.

b. Feedwater setpoint adjustment range: 1.4-5.0 x 10^6 lb/hr

c. Feedwater setpoint: 2.83 x 10^6 lb/hr

G.2.1.2.7 Recirculation Pump Flow Measurement

G.2.1.2.7.1 Description

a. Flow element type - Elbow taps

b. Rated flow type - 47,200 gpm

c. Flow element location - Pump suction line

G.2.1.2.7.2 Performance

a. Range - 20-115% rated pump flow

b. Flow element accuracy (% rated pressure drop) - ±5%

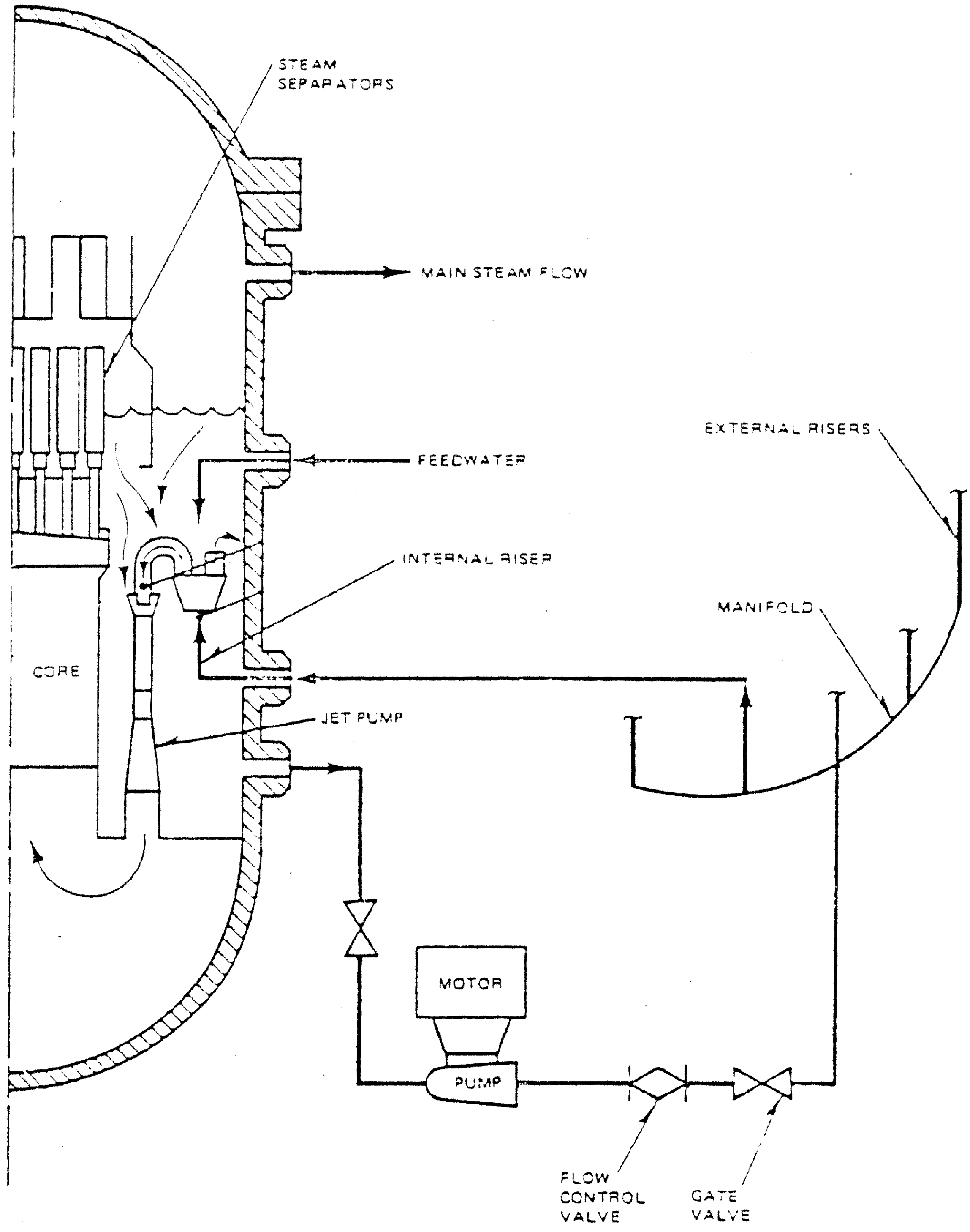

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FIGURE G.2.1-1

SCHEMATIC OF RECIRCULATION SYSTEM

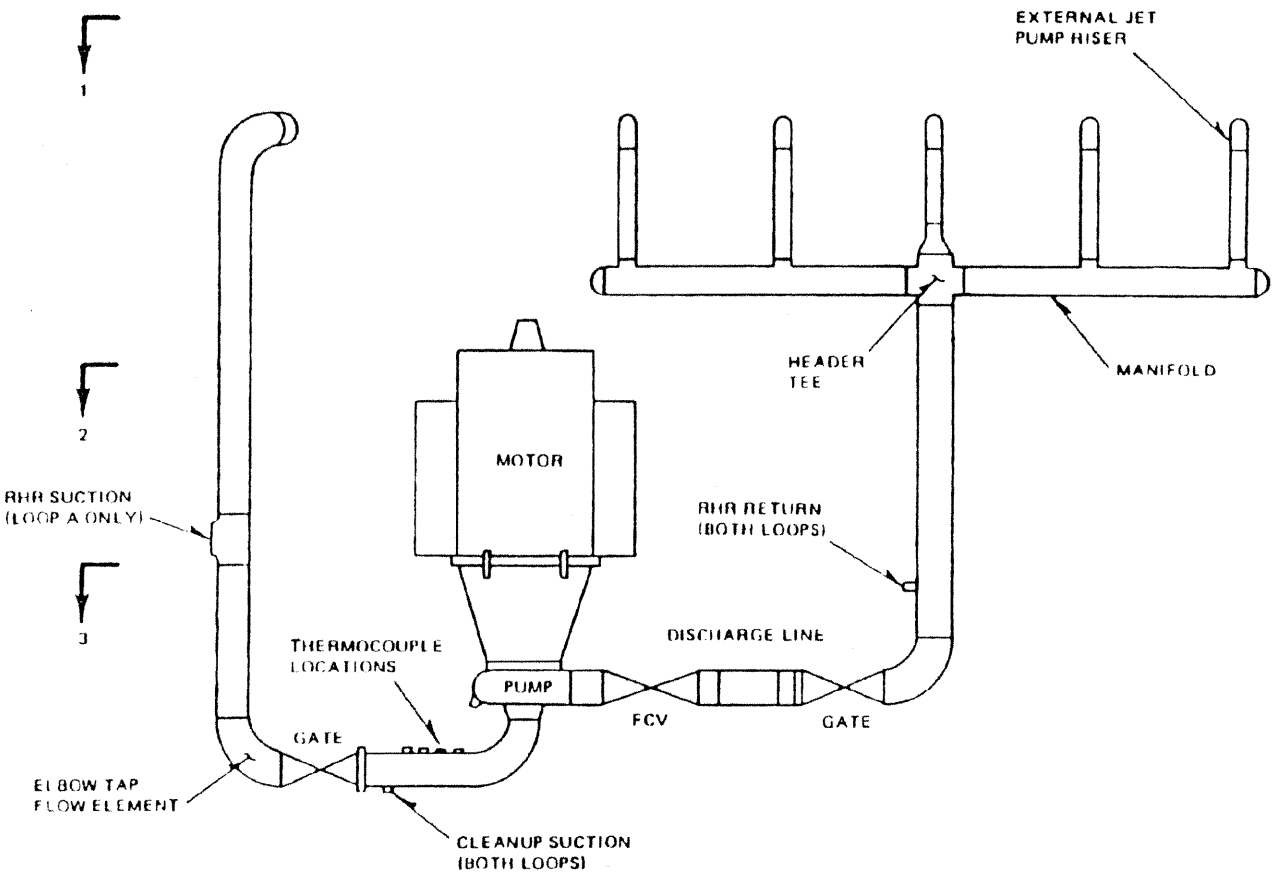

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FIGURE G.2.1-2

ELEVATION A-A: RECIRCULATION SYSTEM
EXTERNAL LOOP PIPING LAYOUT

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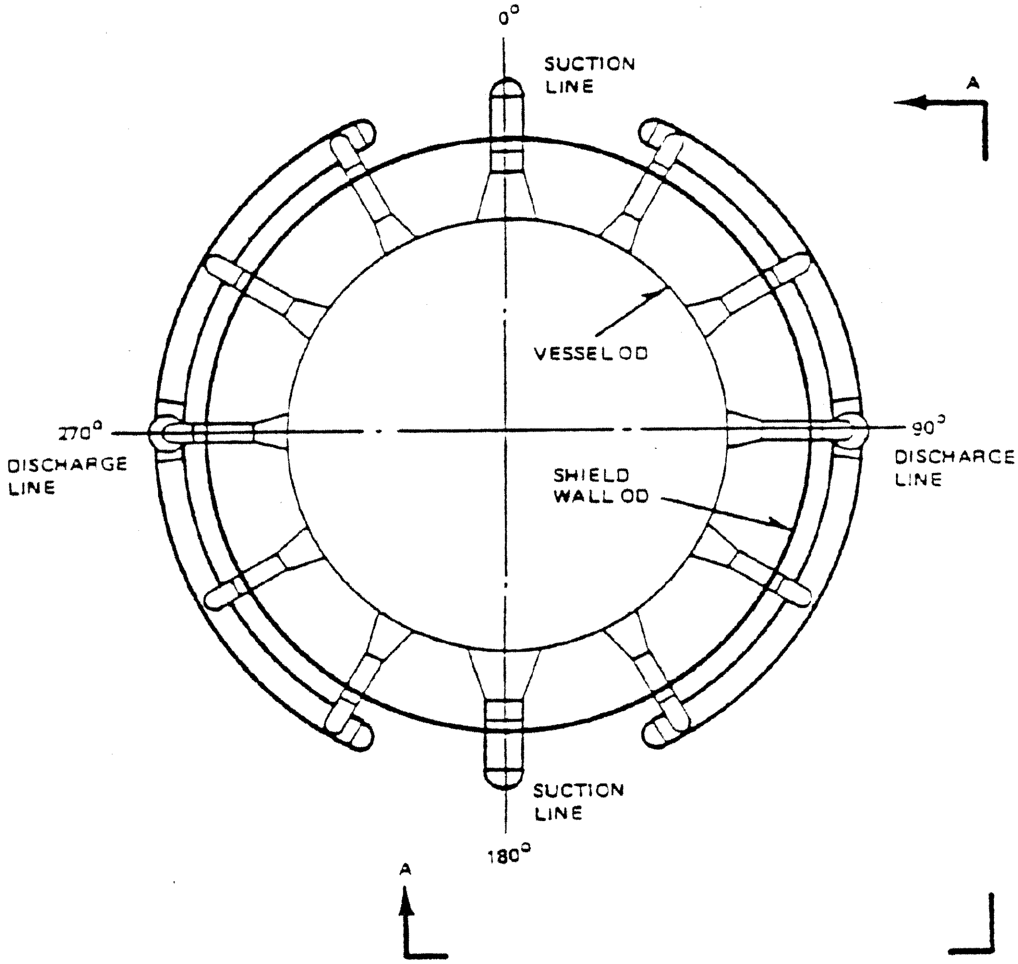

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FIGURE G.2.1-3

PLAN 1: RECIRCULATION SYSTEM
EXTERNAL LOOP PIPING LAYOUT

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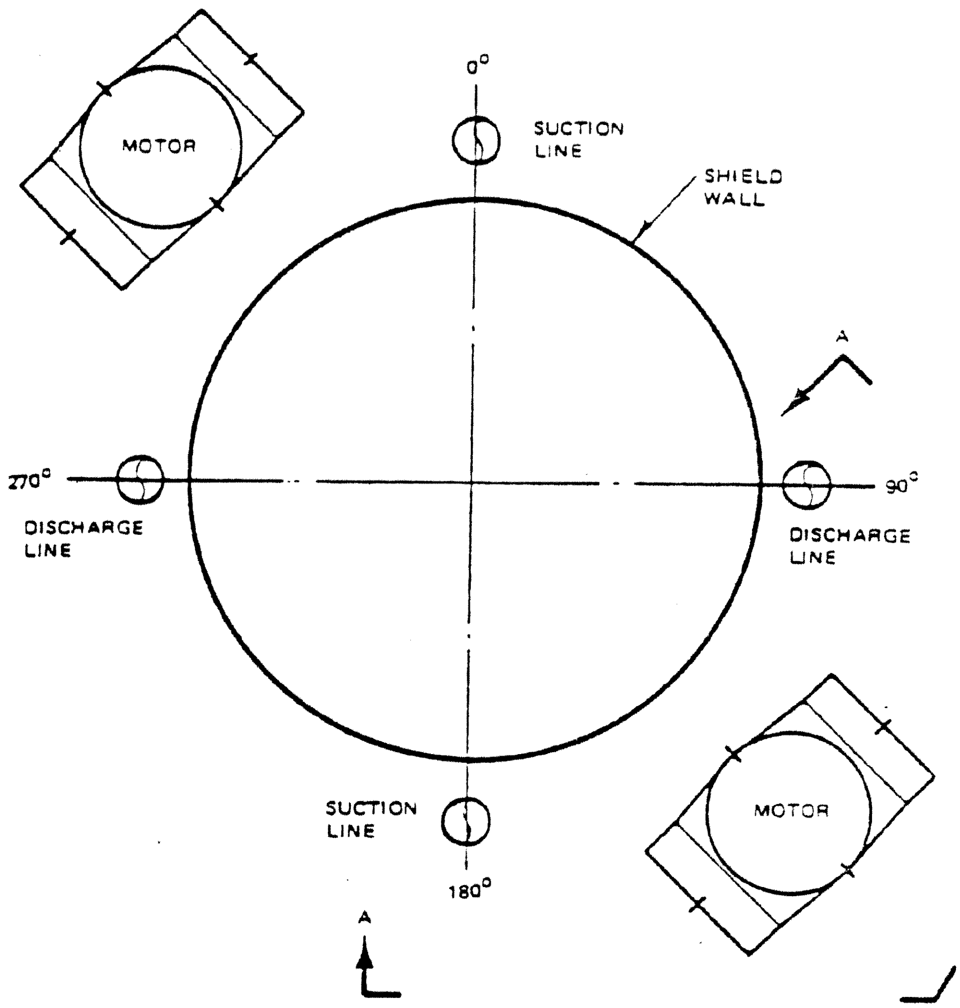

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FIGURE G.2.1-4

PLAN 2: RECIRCULATION SYSTEM
EXTERNAL LOOP PIPING LAYOUT

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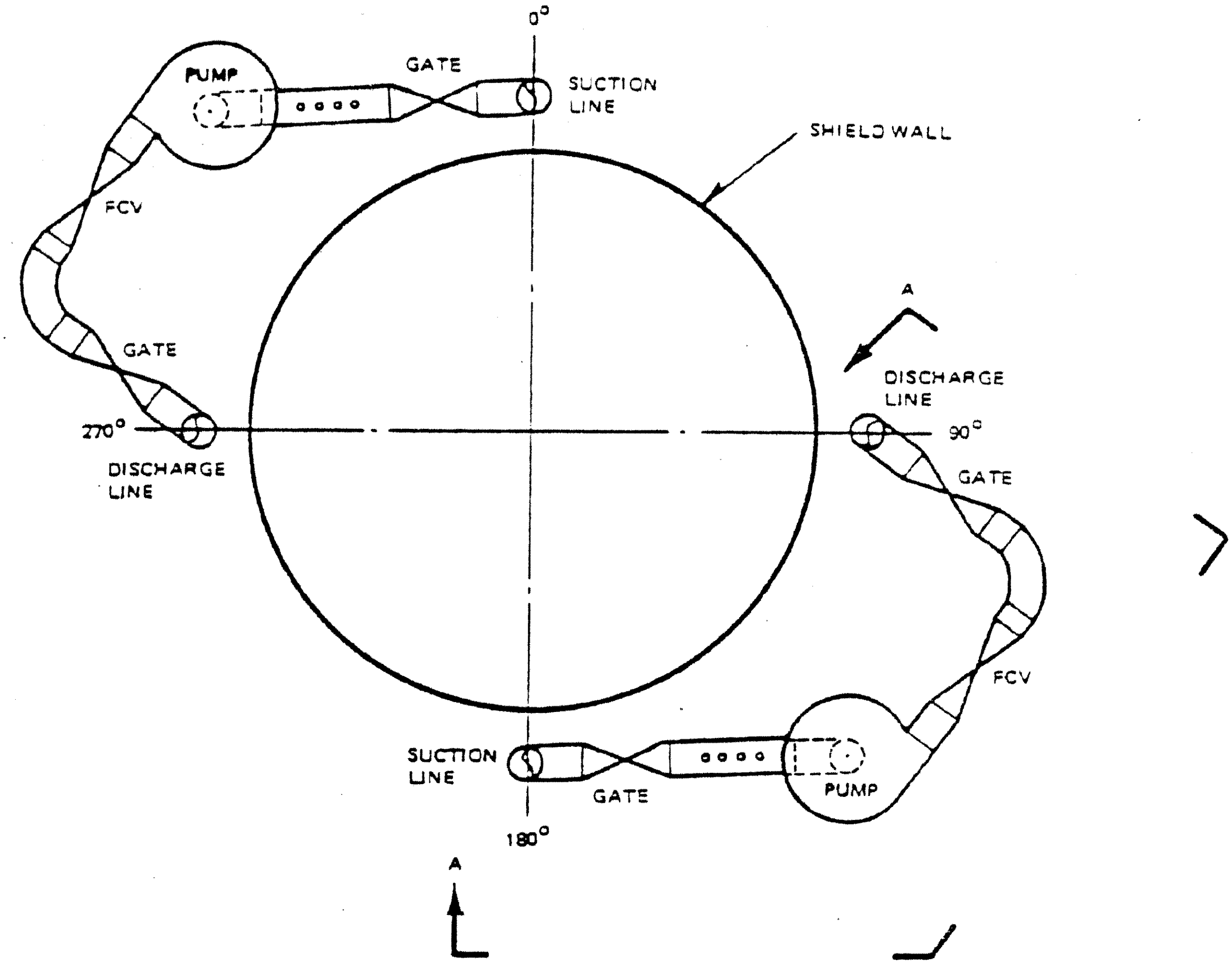

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FIGURE G.2.1-5

PLAN 3: RECIRCULATION SYSTEM
EXTERNAL LOOP PIPING LAYOUT

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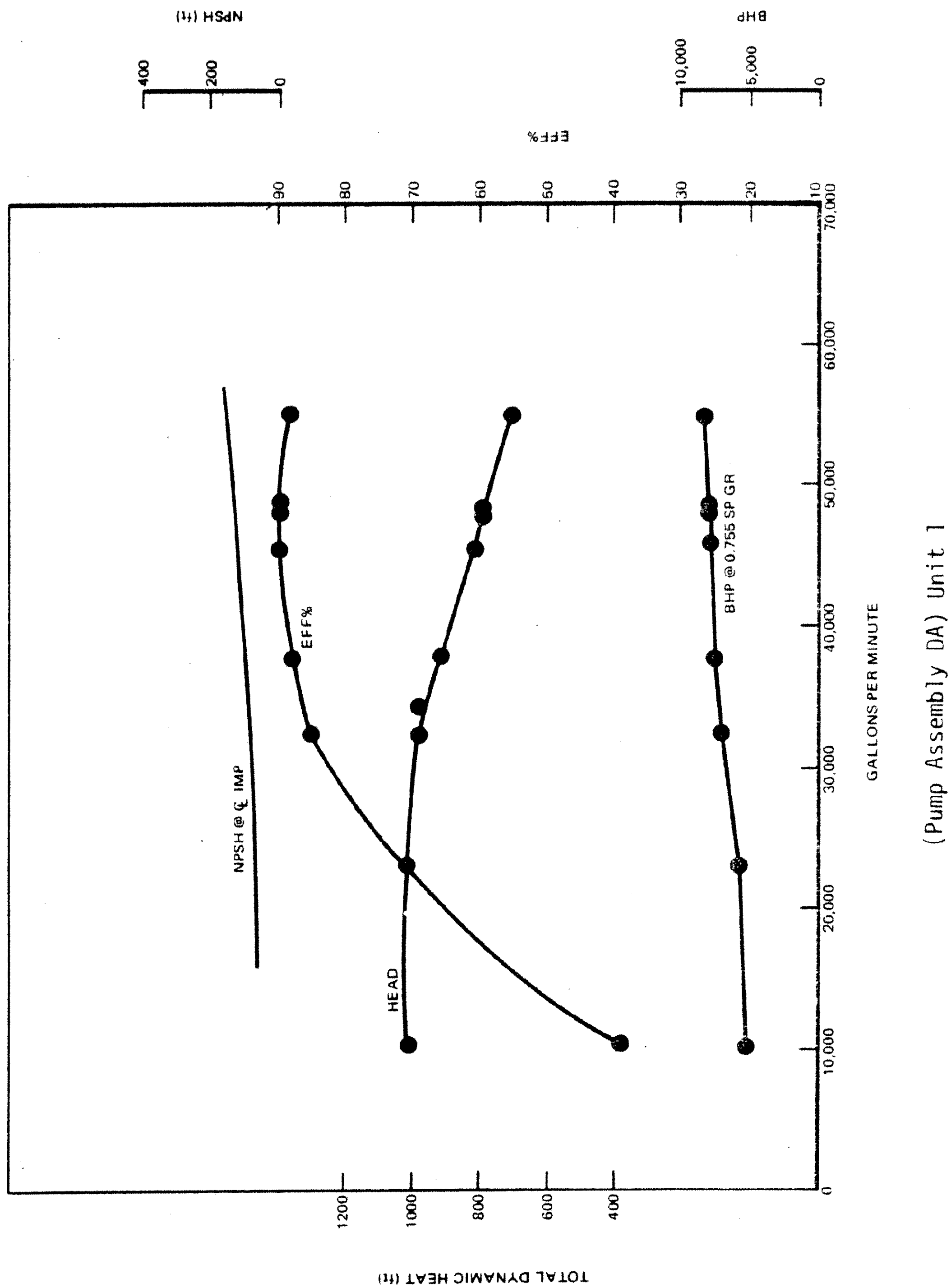

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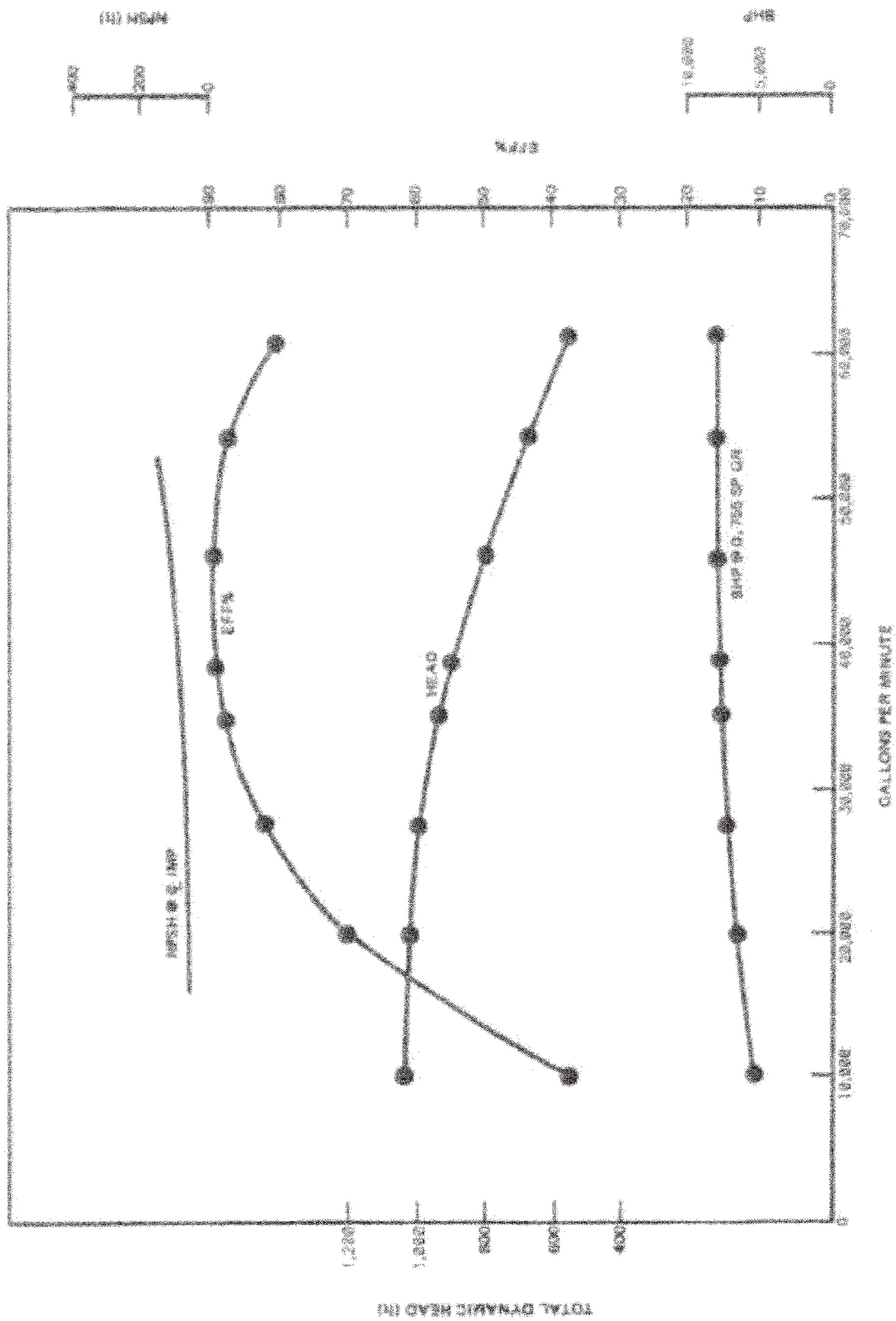
FIGURE G.2.1-6 RECIRCULATION PUMP CHARACTERISTICS (SHEET 1 of 4)



(Pump Assembly DB) Unit 1

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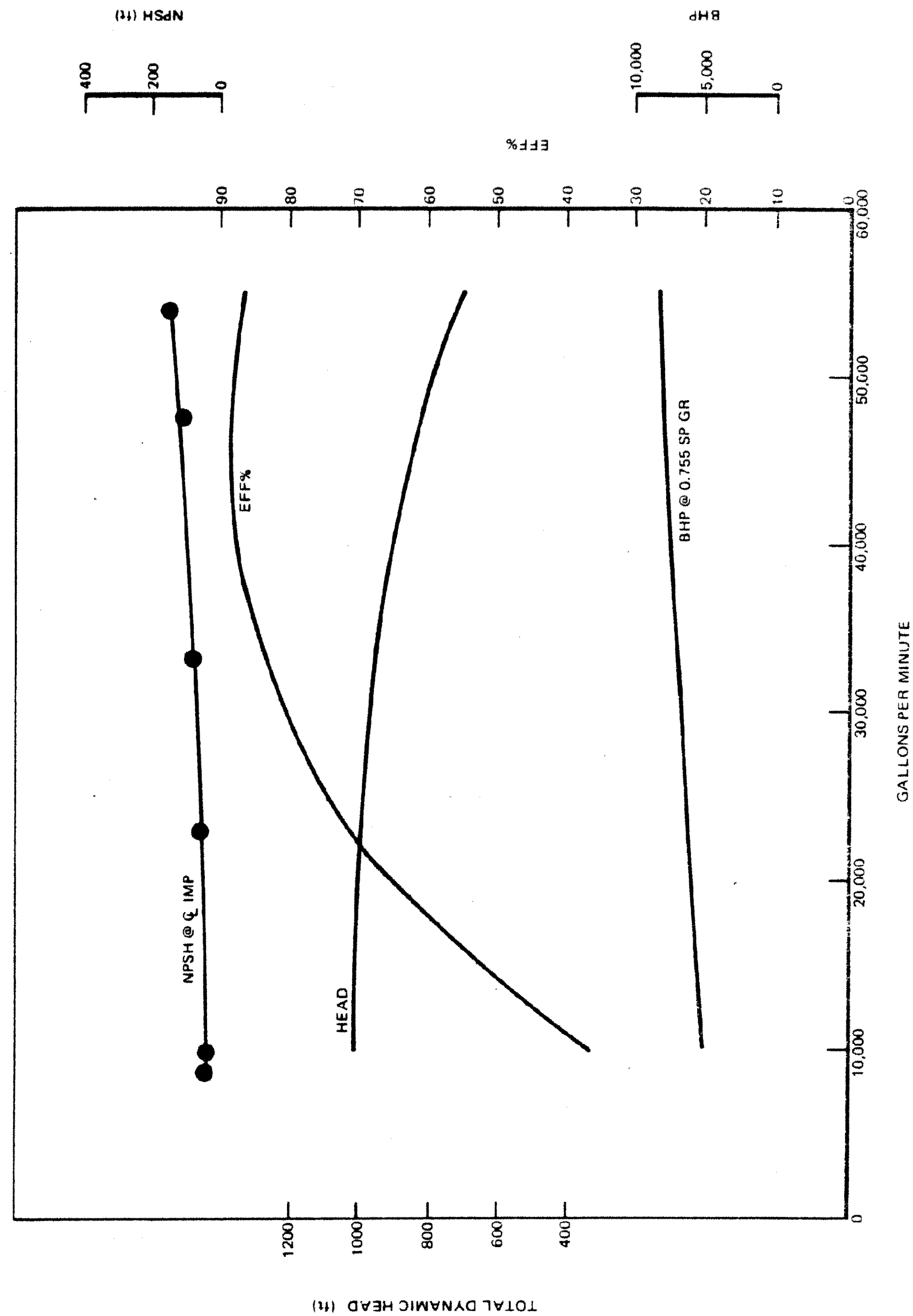
FIGURE 6.2.1-6 RECIRCULATION PUMP CHARACTERISTICS (SHEET 2 of 4)

REVISION 14



(Pump Assembly DC) Unit 2

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FIGURE G.2.1-6

RECIRCULATION PUMP CHARACTERISTICS
(SHEET 3 of 4)

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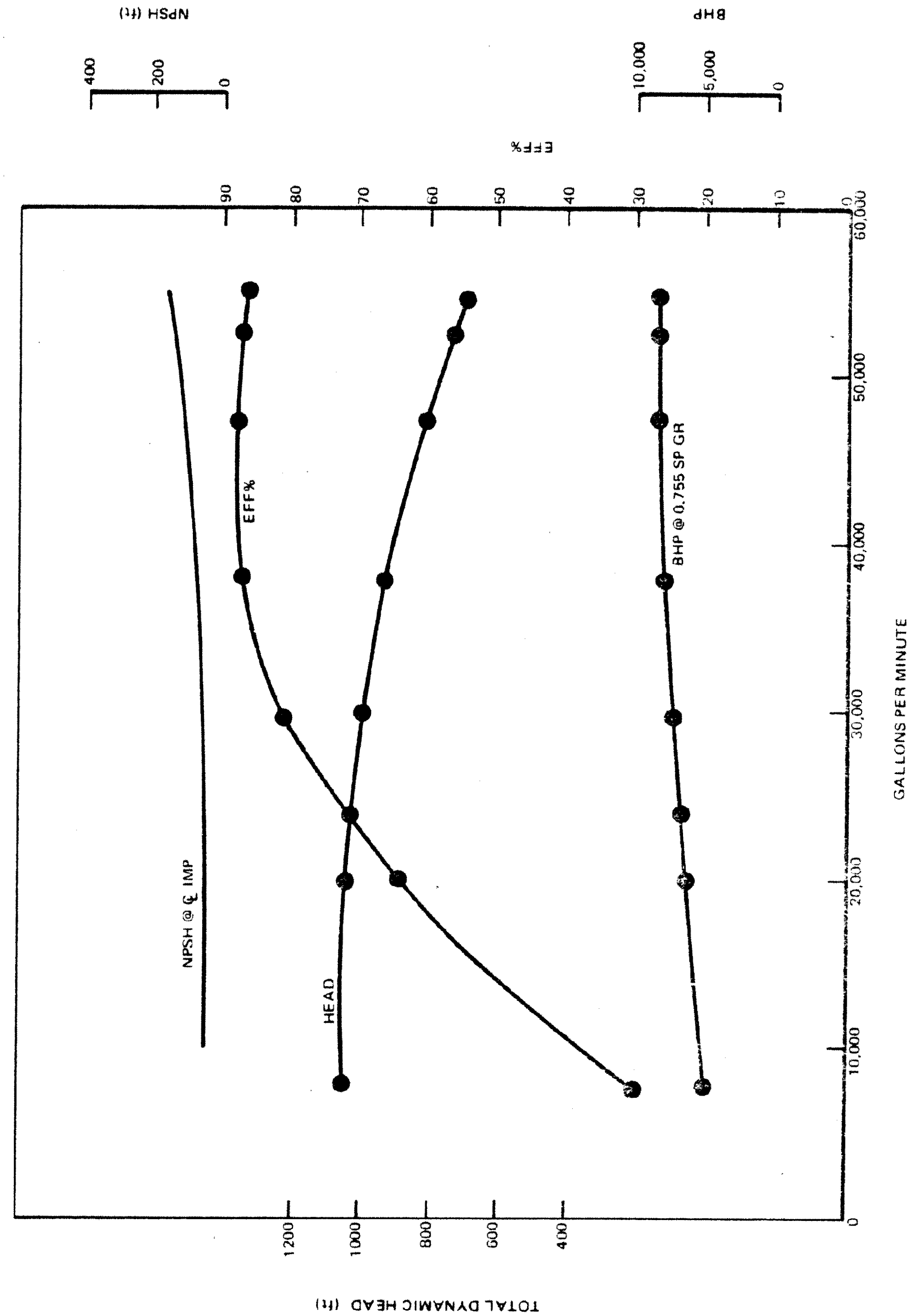

(Pump Assembly DD) Unit 2

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FIGURE G.2.1-6 RECIRCULATION PUMP CHARACTERISTICS (SHEET 4 of 4)

G.2.2 Operating Restrictions

G.2.2.1 Cavitation

G.2.2.1.1 Background

G.2.2.1.1.1 Cavitation

Cavitation is the term used to describe the formation and subsequent collapse of vapor bubbles in a fluid stream. Vapor bubbles can be formed in the stream when the local static pressure becomes less than the vapor pressure of the fluid. These bubbles then collapse when the local static pressure increases.

G.2.2.1.1.2 Net Positive Suction Head (NPSH)

NPSH is a term initially used by pump manufacturers to describe the minimum static to saturation vapor pressure difference that is needed at the pump suction to avoid cavitation in the pump. This term has been generalized to define the available pressure difference in a fluid stream. Consequently, NPSH is a measure of the pressure margin available before vapor bubbles start to form in the fluid.

G.2.2.1.1.3 Cavitation Damage

Cavitation damage is caused by the collapse of the bubbles. It will occur only when the static pressure increases downstream of the point where these bubbles form. While cavitation is usually referenced to the component that creates the vapor bubbles this is generally not where the bubbles collapse and the consequent damage occurs.

General practice is to avoid cavitation in hydraulic systems. While the mechanism is not well understood, industrial data available indicate that cavitation can result in severe material damage if it continues over a prolonged period of time. However, little is known about the damage potential versus bubble velocity, rate of static pressure increase, surface material properties, and density of the fluid.

The largest body of knowledge comes from the centrifugal-axial pump impeller and ship propeller design areas where cavitation has caused material pitting and erosion, leading in severe cases to failure of the impeller or propeller. The damage is attributed to rapid collapse of the vapor bubbles on or very near the surface of the impeller. Such bubble collapse shocks the surface, since it must absorb the energy of the fluid that is accelerated to close the vapor bubble void. The potential for severe shocks decreases as the distance of the bubble from the surface increases. The distance the bubble must be from the surface in relation to its size in order to avoid these shocks is essentially unknown. Consequently, little is known about the need to prevent free-stream cavitation relatively far from solid boundaries.

In cases of severe cavitation, abnormal vibration of the associated component could occur. Whether this vibration is excessive depends on the cavitation severity and the rigidity of the component.

G.2.2.1.2 Recirculation System Cavitation

G.2.2.1.2.1 Recirculation Pump

Low NPSH would produce vapor bubbles near the center of the impeller. These would collapse as they move toward the impeller exit. Consequently, cavitation affects the impeller surfaces and not the bowl or discharge region.

G.2.2.1.2.2 Flow Control Valve

Low NPSH would produce vapor bubbles in the fluid stream before it reaches the valve ball due to the decrease in static pressure. These bubbles would collapse in the pressure recovery areas at the ball and valve exits affecting material in these areas.

G.2.2.1.2.3 Jet Pump Nozzle

Vapor bubbles could form in the nozzle as the pressure decreases and velocity increases toward the nozzle exit. These would collapse in the pressure recovery area in the jet throat, affecting material in the throat area.

G.2.2.1.2.4 Jet Pump Suction

Vapor bubbles could form in the suction flow as pressure decreases and velocity increases from the downcomer annulus to the throat. These would also collapse in the throat pressure recovery area.

G.2.2.1.2.5 Cavitation Coefficients

The recirculation pump, jet pump, and flow control valve are tested to determine their cavitation coefficients so that prolonged operation in cavitating regimes can be avoided.

G.2.2.1.2.6 NPSH

System NPSH depends on reactor water level and water subcooling below vessel dome saturation temperature. During normal operation, feedwater flow provides the most subcooling by mixing with the steam separator return flow making the downcomer water temperature less than saturation temperature.

G.2.2.1.3 Operating Restrictions

Operation in cavitating conditions is prevented in most cases by automatic interlocks and in others by operating procedures.

G.2.2.1.3.1 Automatic Interlocks

As mentioned in Subsection G.2.1.2.6, cavitation interlocks are based on feedwater flow and the water temperature difference between the suction line temperature and the vessel water saturation temperature. These interlocks activate 15 seconds after the low feedwater flow condition is indicated, or 15 minutes after a low temperature difference is indicated, in order to prevent spurious trips from temporary changes in temperature or feedwater flow. These interlocks trip the pumps from 100% to 25% speed. Water level interlocks sequentially trip the pumps from 100% to 25% speed and then from 25% speed to zero if level drops to the point where a loss-of-coolant accident or anticipated transient without scram is indicated. (These interlocks are also discussed in Section G.5.)

A bypass function is provided on the Operator Station to bypass the cavitation interlocks. Use of these bypasses are administratively controlled.

G.2.2.1.3.2 Operating Procedures

G.2.2.1.3.2.1 One Recirculation Pump Operation

The automatic interlocks do not prevent jet pump cavitation in this operating mode. Reactor operation is limited to the areas above the jet pump nozzle cavitation line "N" of Figure G.2.5-2. The flow control valve cavitation interlock remains functional in this operating mode.

G.2.2.1.3.2.2 Residual Heat Removal System Operation

The RHR system can use the recirculation loop jet pumps to provide circulation through the reactor core. Operating restrictions limit RHR operation to regions where jet pump cavitation does not occur. NPSH requirements for RHR operation were established during preoperational testing.

G.2.2.1.3.2.3 Vessel Hydrostatic Testing

In preoperational and startup modes of operation, recirculation pump operation at 100% speed is needed to heat the vessel water up to the temperature required to perform hydrostatic tests. A bypass function is provided to disable the cavitation interlocks. Use of this function and system operation with the cavitation interlock disabled are governed by operating procedures. These procedures were determined during preoperational testing.

G.2.2.2 Loop Flow Balance

The design-basis loss-of-coolant accident analyses assume that the total flow through one bank of jet pumps is close to the total flow of the other bank. If it is assumed that one bank is at a higher flow than the other and a design-basis pipe rupture occurs in the high-flow loop, calculated peak cladding temperatures higher than those calculated in the DBA analysis could result, since the low flow loop could not provide as much flow during the first few seconds of the transient. Consequently, loop flow imbalance restrictions are placed on the system.

G.2.2.3 Thermal Shock

When the recirculation system is inactive and thermal power is too low to support much natural circulation flow, stratification of water in the recirculation loops or vessel bottom head can occur. If core flow is increased in such a condition, rapid replacement of cold with hot water shocks the adjacent components increasing their usage. Although the reactivity transient on the core is not significant, the increased usage is not acceptable. Consequently, restrictions are placed on allowable inactive loop suction temperature before startup and on core flow increases based on bottom head drain line to vessel saturation temperature.

G.2.2.4 Restart of One Recirculation Pump

In order to maintain plant availability, it is necessary to follow specific procedures for restart of one pump to return to two-pump operation. Otherwise, a scram will result. Specific procedures to avoid scram were determined during startup testing and subsequent analyses. The procedures include reducing active pump flow to at least 50%, reducing inactive flow control valve position to restart permissive level, and verifying that delta temperature between coolant in the idle loop and in the reactor pressure vessel is less than or equal to 50° F, before starting the inactive pump.

G.2.2.5 25% to 100% Speed Transfer

Transfer from low to high speed also causes an increase in reactivity. If this is done at high thermal power, a reactor scram will result. The specific thermal power range where transfer can be made without scram is determined based on analysis of startup test results and subsequent analyses and is included in plant operating procedures.

G.2.3 Operation and Performance

G.2.3.1 General Description

The reactor recirculation system pumps reactor coolant through the core. This is accomplished by two recirculation loops external to the reactor vessel but inside the primary containment. Each external loop contains one high-capacity motor-driven recirculation pump, a flow control valve, and two motor-operated gate valves for suction shutoff and discharge shutoff purposes. Each pump suction line contains a flow-measuring system.

The internal portion of the loop consists of the jet pumps, which contain no moving parts. The jet pumps provide a continuous internal circulation path for the major portion of the core coolant flow. The jet pumps are located in the annular region between the core shroud and the vessel inner walls. Any recirculation line break would still allow core flooding to approximately two-thirds of the core height, the level of the inlet of the jet pumps.

In the jet pump nozzle, the high static head developed by the centrifugal pumps is converted to a high-velocity jet at a low static pressure. The low pressure at the nozzle discharge draws the surrounding fluid into the jet pump throat, where it is mixed with the driving flow and then passes into the diffuser. In the diffuser, the velocity is reduced and the flow leaves the jet pumps at a pressure somewhat higher than the pressure in the downcomer.

The main resistances against which the jet pumps work are the core pressure drop and the steam separator pressure drop. There is a small, beneficial natural circulation driving head arising from the density difference between the two-phase mixture in the core and the subcooled liquid in the downcomer. Core flow is very nearly a linear function of the jet pump driving flow, except at low flow rates where natural circulation becomes significant.

Jet pump driving flow rate and thus core flow rate is varied by means of control valves located in each loop immediately downstream of the centrifugal pumps.

To avoid system cavitation during startup, hot standby and low-power operation, the recirculation system is provided with the capability of operating the centrifugal pumps at 25% of rated speed. Low-speed operation is accomplished by supplying power at a fixed low frequency to the pump-motor from a motor-generator set provided for this purpose.

G.2.3.2 Normal Operation

Normal plant operation presumes that no abnormal condition exists due to such events as scram, operator error, or turbine trip. In this context, an operator error is considered an action that is contrary to plant operating procedures.

Monitoring of recirculation and jet pump flow distributions is performed periodically to verify expected performance. Routine monitoring for potential jet pump nozzle plugging was implemented after Unit 2 Cycle 2 because a 2.7 inch diameter spacer was separated from the 2B discharge valve, 2B33-F067B. The potential operational and safety effects of the disk were evaluated and approved. No measurable effect was produced.

G.2.3.2.1 Start

The system is started after the core is loaded, the vessel internals installed, the head is in place and the water level is at least at the normal level. Residual heat removal operation through the recirculation system is terminated and the system valves are oriented to their startup positions as follows:

a. initial conditions (recirculation system):

   1. discharge valve closed (prevents RHR flow backwards through the pump);

   2. suction valve open (allows cleanup suction from the recirculation loops);

   3. flow control valve in minimum position; and

   4. pump off.

b. startup procedures (Figure G.2.3-1):

   1. terminate RHR operation through the recirculation loop discharge lines;

   2. open discharge gate valve;

   3. start pump (the automatic sequencing enables the 100% speed power supply to start the pumps, closes the LFMG feeder breaker to bring it up to speed, trips the 100% speed supply when 90 to 100% speed is reached and closes the LFMG output breaker when the pump has coasted down to 25% speed); and

4. open FCV to the wide open position (100% open).

G.2.3.2.2 Heatup and Pressurization

During vessel pressurization and heatup to rated conditions, the system is run at 25% speed with the FCV wide open (100% open).

G.2.3.2.3 Low Thermal Power

Power ascension up to 30% to 40% power is accomplished by pulling control rods. The recirculation system operating mode is 25% speed with the flow control valve wide open.

G.2.3.2.4 High Thermal Power

The system is transferred to 100% speed after the flow control valve cavitation interlock is cleared (approximately 25% thermal power/20% feedwater flow). This allows further power ascension by core flow control as well as by control rod movement.

a. transfer procedure (power ascension):

   1. Close the FCV to the minimum position.

   2. Transfer to 100% speed - FCV minimum position (the automatic sequencing opens the LFMG generator breaker, opens the LFMG feeder breaker and closes the 100% speed breaker).

b. transfer sequence (normal shutdown):

   1. Close FCV's to minimum position (100% speed).

   2. Transfer to 25% speed - FCV minimum position (the automatic sequencing opens the 100% speed breaker, closes the LFMG feeder breaker and closes the LFMG generator breaker when the pump has coasted down to 25% speed).

   3. Open FCV's to wide open position.

G.2.3.2.5 Natural Circulation

Power ascension without forced circulation creates the possibility of shocks to vessel components due to vessel fluid stratification. Although the piping systems are designed to minimize this effect, plant heatup must be more gradual than with forced circulation due to the need to prevent buildup of stratified layers. Consequently, natural circulation is considered an abnormal startup procedure.

G.2.3.2.6 Thermal Power - Core Flow Operating Map

A boiling water reactor must operate with certain restrictions because of pump net positive suction head (NPSH), overall plant control characteristics, core thermal power limits, etc. A typical power-flow map for the power range of operation is shown in Figure G.2.3-2. The nuclear system equipment, nuclear instrumentation, and the reactor protection system, in conjunction with operating procedures, maintain operations within the area of this map for normal operating conditions. The Core Operating Limits Report (COLR) contains cycle specific transient analyses that defines the operational boundaries of interest on the power to flow map. Points along the APRM rod block line and the increased core flow line are analyzed each cycle to determine appropriate thermal limit values. All other information on the power to flow map is considered reference only. A description of the power to flow map is as follows:

a. Natural Circulation Line (A) - The Operating state of the reactor moves along this line for normal control rod withdrawal sequence in the absence of recirculation pump operation.

b. 100% FCL - This line passes through 100% power at 100% flow. The operation state for the reactor follows this line for recirculation flow changes with a fixed control rod pattern. The line is based on a constant xenon concentration at rated power and flow.

c. Cavitation Protection Lines - 20% FW Flow for Flow control valve cavitation (reference 4) and 10.1 F for jet pump nozzle cavitation protection (reference 5).

d. APRM Rod Block Line - defined in Technical Specifications.

e. 105% flow line - defines the increased core flow (ICF) region.

f. REG A/B - Interim Corrective Action (ICA) regions for instability concerns. High power and low flow conditions are known areas where instabilities may occur.

G.2.3.2.7 Mechanical and Control Features

The following limits and design features are employed to maintain power conditions at the required values shown in Figures G.2.3-2:

a. Minimum Power Limit at Intermediate and High Core Flows - To prevent cavitation in the recirculation pump, jet pump, and flow control valves, the recirculation system is provided with an interlock to trip off the 60-Hz source and close in the 15-Hz power source if the difference between vessel water saturation temperature and recirculation pump inlet temperature is less than a preset value.

b. Minimum Power Limit at Low Core Flow - During lower power, low-loop flow operations, the temperature differential interlock does not provide cavitation protection to the flow control valves. Therefore, the system is also provided with an interlock to trip off the 60-Hz power source and close in the 15-Hz power source when the feedwater flow falls below a preset level.

Discussion

When reactor power is greater than approximately 25% of rated (20% feedwater flow), the low feedwater flow interlock is cleared, and the main recirculation pumps can be switched to the 100% speed power source. The flow control valve is closed to the minimum position before the speed change to prevent large increases in core power and a potential flux scram. The system is then brought to the desired power flow level within the normal operating area of the map by opening the flow control valves and withdrawing control rods.

Control rod withdrawal with constant flow control valve position will result in power-flow changes along lines of constant C_v (constant position). Flow control valve movement with constant control rod position will result in power-flow changes along, or nearly parallel to, the rated flow control line.

Large negative operating reactivity coefficients, inherent in the boiling water reactor, provide the following important advantages:

a. good load-following with well-damped behavior and little undershoot or overshoot in the heat transfer response,

b. load-following the recirculation flow control, and

c. strong damping of spatial power disturbances.

Because of the negative void coefficient, load-following is accomplished by varying the reactor recirculation flow. To increase reactor power, the recirculation flow rate is increased. The increased core flow sweeps some of the voids from the moderator and causes an increase in core reactivity. As reactor power increases, more steam is formed and the reactor stabilizes at a new power level, with the transient excess reactivity balanced by the new void formation. No control rods are moved to accomplish this change of power level. Conversely, when a power reduction is required, the recirculation flow rate is reduced. The resultant formation of more voids in the moderator automatically decreases the reactor power level to that commensurate with the new recirculation flow rate. Again, no control rods are moved to accomplish the power reduction.

Varying the recirculation flow rate (flow control) is more advantageous, relative to load-following, than using control rod positioning. Flow variations perturb the reactor uniformly in the horizontal planes ensure a flatter power distribution and reduced transient allowances. As flow is varied, the power and void distributions remain approximately constant at the steady-state end points for a wide range of flow variations. After adjusting the power distribution by positioning the control rods at a reduced power and flow, the operator can then bring the reactor to rated conditions by increasing flow, with the assurance that the power distribution will remain approximately constant.

G.2.3.3 Abnormal Operation

Abnormal operation is defined as any operation other than normal routine and includes such events as operator error, plant operational transients such as MSIV closure and turbine trip, or an accident condition, such as a recirculation line double-ended pipe break.

The recirculation system is designed to achieve the following two major objectives for abnormal operation:

a. Plant Safety - those system functions required to mitigate an abnormal operational event, that is functions for which credit is taken in the event analysis; and

b. Power Generation - those functions necessary to maximize power generation capability and ensure longevity of equipment. These functions prevent steady-state operation of system equipment in modes where damage can occur and prevent unnecessary reactor protection system (RPS) activation. Equipment damage would make it necessary to inspect or replace equipment during a schedule outage thereby increasing plant unavailability. Unnecessary scrams also increase plant unavailability.

G.2.3.3.1 Safety

G.2.3.3.1.1 Moderate and Infrequent Events

Trip of the recirculation pumps is needed to mitigate the effects of turbine or generator trip events.

G.2.3.3.1.2 Accident Events

Trip of the recirculation pumps is needed to mitigate the vessel overpressure transient and reduce core power level for the ATWS event.

G.2.3.3.2 Power Generation

G.2.3.3.2.1 Cavitation Interlocks

Cavitation interlocks are installed to provide protection against:

a. operator errors, and

b. rapid transients such as scram where NPSH conditions deteriorate rapidly and the operator is not expected to respond in time to prevent long-term operation in the cavitation region.

G.2.3.3.2.2 Recirculation FCV Runback

A recirculation FCV runback occurs if the total steam flow exceeds the feedwater flow capacity of the feedwater pumps running and reactor water level is less than L4. A 5% conservative bias is added to total steam flow, to keep total steam flow well within the capacity of the feedwater system. The reduction in recirculation flow helps avoid a low level reactor scram by reducing the steaming rate. The recirculation FCVs will runback as long as the condition is valid.

The recirculation FCV runback limits the valve position demand and the flow control A/B trouble alarm is annunciated in the 1(2)H13-P602 panel. When the FCV runback occurs, an automatic transfer to manual mode of the FCV's is performed.

G.2.3.3.2.3 Flow Control Valve Minimum Position Interlock

The interlock is installed to prevent system startup or transfer from 25% to 100% speed unless the valve is in the minimum position. The objective is to prevent scrams due to a rapid flow increase resulting from an operator failure to close the FCV prior to the start of speed transfer.

G.2.3.3.2.4 LFMG Output Breaker Control

The LFMG set output breaker closing logic includes interlock which prevent breaker closure until the 100% power supply is tripped and residual voltages in the motor have decayed. These interlocks prevent possible damage to the generator which could result if the generator was connected to the pump while the 100%-speed power supply was still active.

G.2.3.3.2.5 High Loop Flow Mismatch

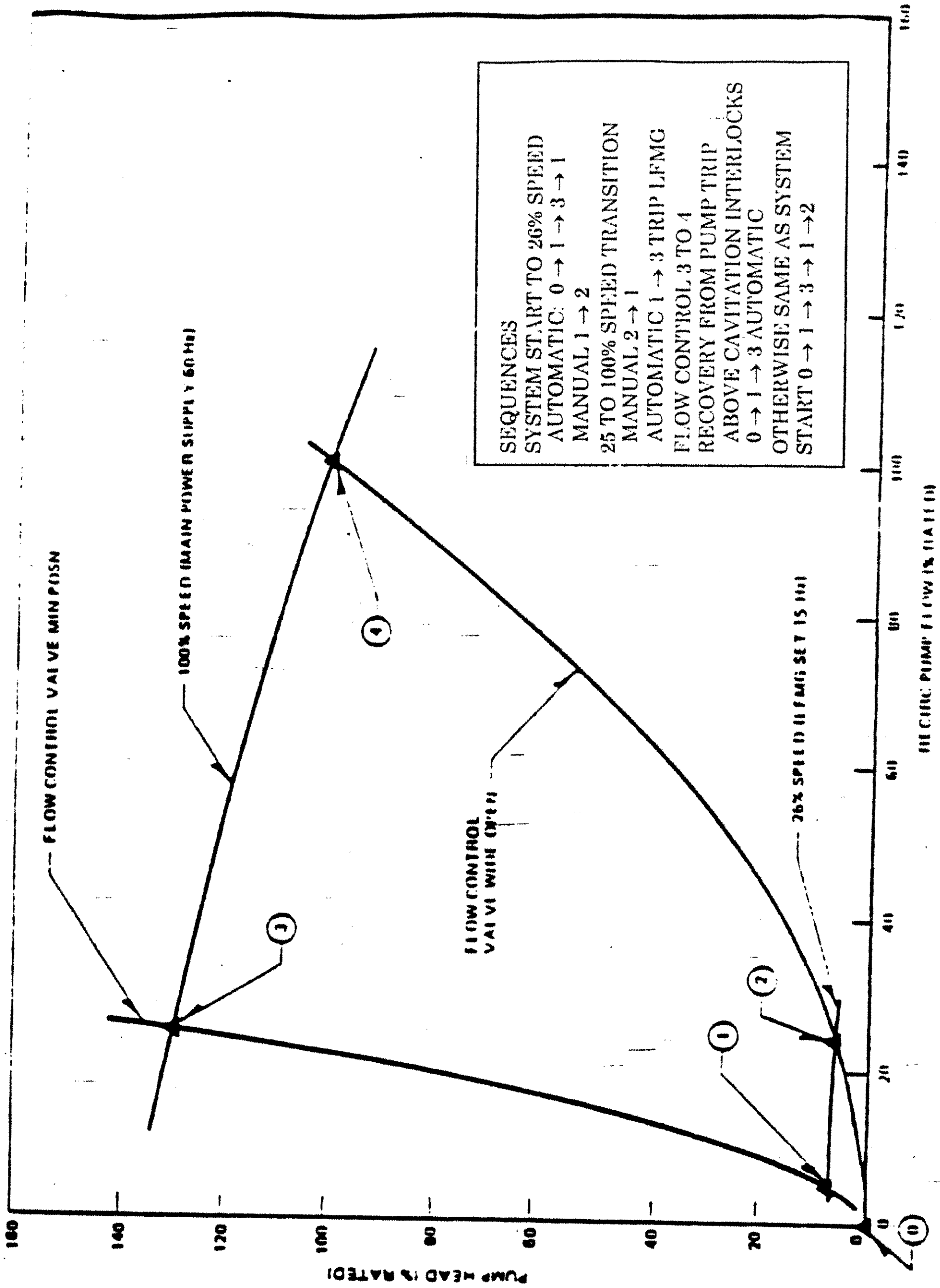
Mismatch of the flow in one loop to the other loop of greater than 50% is known to create abnormal conditions in the jet pumps having the lower flow. Such operation is normally precluded by operating procedures. Following a trip of one of two operating pumps, the tripped pump coasts down to zero speed rather than 25% speed to avoid automatic generation of a 100% - 25% mismatch condition.

G.2.3.3.2.6 Loop Suction and Discharge Isolation Valve Position

The pump is tripped at less than 90% open position to prevent pump damage from no flow if isolation valve closure is inadvertently initiated while the pumps are running.

G.2.3.3.2.7 Trip to 25% Speed

The LFMG set is activated in most 100% speed trip cases to avoid scram recovery delays due to vessel bottom head fluid stratification. The specific cases are discussed in Table G.2.5-3.



REPRESENTATIVE OF 20% FLOW CONTROL VALVE POSITION

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FIGURE G.2.3-1
LFMG SET: FLOW CONTROL VALVE RECIRCULATION SYSTEM

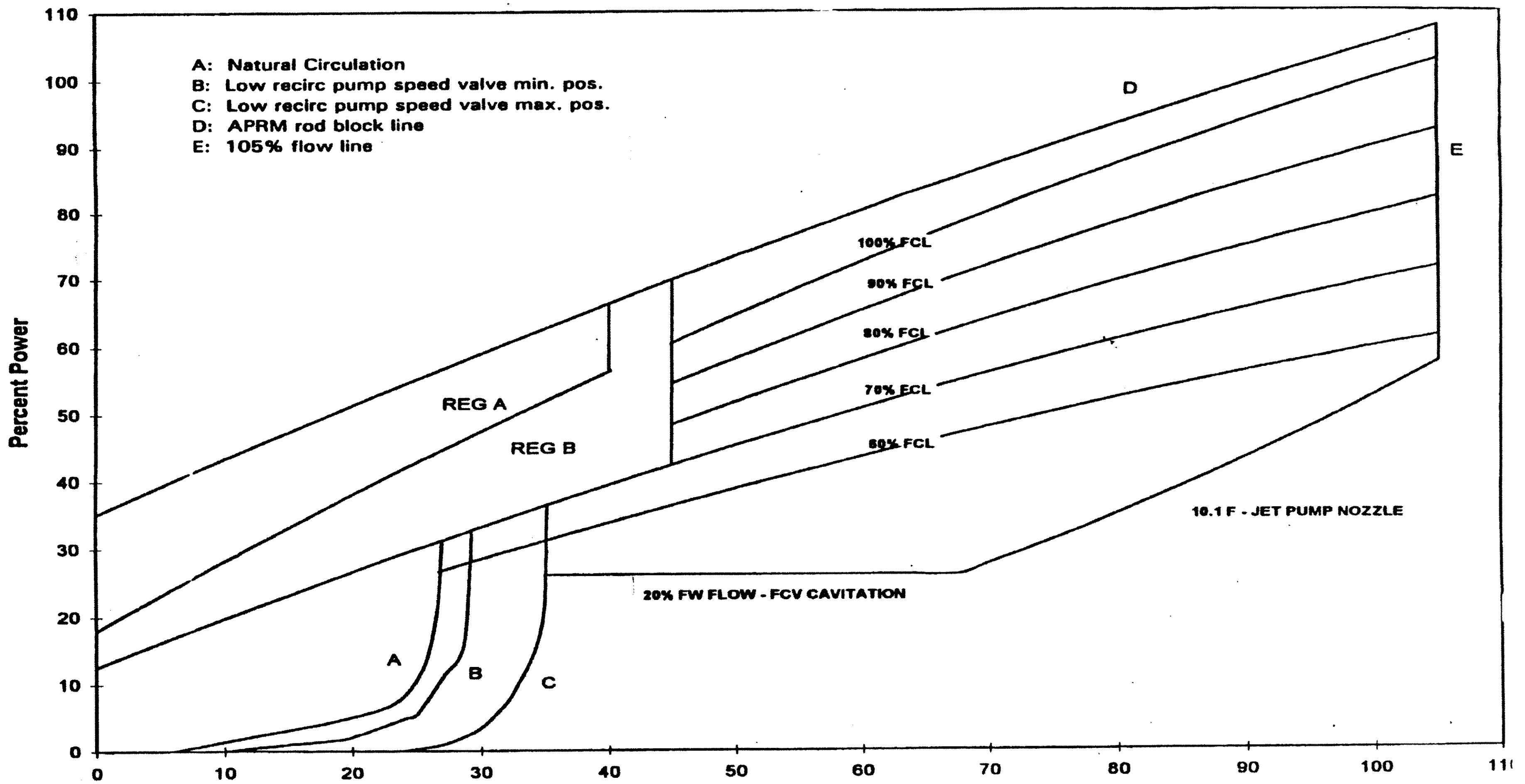

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FIGURE G.2.3-2

TYPICAL POWER TO FLOW MAP

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G.2.5 Thermal and Hydraulic Characteristics Summary: Tables and Figures

G.2.5.1 Table G.2.5-1 System Design Data at Rated Thermal Power and Core Flow

G.2.5.2 Figure G.2.5-1: Two Pump Thermal Power-Core Flow Operating Map

G.2.5.3 Figure G.2.5-2: One Pump Thermal Power-Core Flow Operating Map

The LSCS 1999 Power Uprate Project did not re-analyze the reactor operating domain for single loop operation (SLO).

G.2.5.4 Table G.2.5-2 Pressures and Temperatures at Selected Conditions

Key to Table:

POINT = location in the recirculation loop corresponding to numbers on Figure G.2.1-1

W = flow rate in percent of given "Normal Flow"

PDOME+ = static pressure above the given "Normal Dome Pressure" (psi)

TDOME+ = temperature above the given "Normal Dome Temperature" (° F).

G.2.5.5 Table G.2.5-3: System Trip and Start Functions

G.2.5.6 Figure G.2.5-3: Flow Control Valve Characteristics Curve

G.2.5.7 Table G.2.5-4: Reactor Recirculation System Design Characteristics

G.2.5.8 Table G.2.5-5: Plant Configuration Data

TABLE G.2.5-1

SYSTEM DESIGN DATA

a. Core thermal power, MW	3489
b. Dome pressure, psia	1020
c. Core flow, 10^6 lb/hr	114
d. Separator inlet quality, %	0.135
e. Core pressure drop, psi	25.4
f. Recirculation pump flow, % of rated	95.3
g. Recirculation pump flow, gpm	44,983
h. Recirculation pump flow, 10^6 lb/hr	17.0
i. Recirculation pump head, ft	830.7
j. Recirculation pump available NPSH, ft	345
k. Recirculation pump required NPSH, ft	108
l. Recirculation pump brake horsepower	8189
m. Flow control valve position, % open	95
n. Flow control valve C_v, gpm/psi	6931
o. Valve pressue drop, psi	31.8
p. Downcomer enthalpy, Btu/lb	528.2
q. Downcomer specific volume, ft^3/lb	0.02123
r. Jet pump nozzle velocity, ft/sec	217.2
s. Jet pump M-Ratio	2.356

TABLE G.2.5-2
SHEET 1 OF 2

PRESSURES AND TEMPERATURES AT SELECTED CONDITIONS

105. % ROD LINE - VALVES WIDE OPEN

NORMAL DOME PRESS. = 1030. PSIA NORMAL DOME TEMP. = 548. F NORMAL FLOW = 15554. GPM JP EFF. = 41.30 JP M RATIO = 2.326
PRESSURE IMPOSED ON PIPING MAY BE UP TO 13. PSI ABOVE NOMINAL TABLES FOR POINTS 5 THROUGH 16.

POINT	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
W	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	20.00	20.00	20.00	10.00	332.60	665.19
PDOME	6.58	0.45	6.72	5.03	2.74	271.52	271.52	248.87	247.25	245.56	235.43	230.23	226.47	216.91	214.89	45.83	13.34
TDOME	- 14.08	-14.09	-14.08	-14.08	-14.09	-12.79	-12.79	-12.82	-12.82	-12.82	-12.83	-12.84	-12.84	-12.86	-12.86	-13.88	1.57

100. % POWER - LOWER FLOW CONTROL LIMIT

NORMAL DOME PRESS. = 1020. PSIA NORMAL DOME TEMP. = 547. F NORMAL FLOW = 42845. GPM JP EFF. = 41.06 JP M RATIO = 2.350
PRESSURE IMPOSED ON PIPING MAY BE UP TO 14. PSI ABOVE NOMINAL TABLES FOR POINTS 5 THROUGH 16.

POINT	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
W	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	20.00	20.00	20.00	10.00	334.95	669.91
PDOME	6.61	1.62	8.50	6.99	4.89	283.89	283.89	222.43	221.00	219.50	209.69	205.08	201.45	192.47	188.56	41.91	11.92
TDOME	-14.42	-14.42	-14.42	-14.42	-14.42	-13.09	-13.09	-13.16	-13.16	-13.16	-13.17	-13.17	-13.18	-13.19	-13.19	-14.21	1.42

100. % POWER AND 100. % FLOW

NORMAL DOME PRESS. = 1020. PSIA NORMAL DOME TEMP. = 547. F NORMAL FLOW = 42845. GPM JP EFF. = 41.06 JP M RATIO = 2.350
PRESSURE IMPOSED ON PIPING MAY BE UP TO 14. PSI ABOVE NOMINAL TABLES FOR POINTS 5 THROUGH 16.

POINT	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
W	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	20.00	20.00	20.00	10.00	334.95	669.91
PDOME	6.61	1.62	8.50	6.99	4.89	283.89	283.89	222.43	221.00	219.50	209.69	205.08	201.45	192.47	190.68	41.91	11.92
TDOME	-14.42	-14.42	-14.42	-14.42	-14.42	-13.09	-13.09	-13.16	-13.16	-13.16	-13.17	-13.17	-13.18	-13.19	-13.19	-14.21	1.42

TABLE G.2.5-2
SHEET 2 OF 2

20. % ROD LINE - VALVES WIDE OPEN

NORMAL DOME PRESS. = 956. PSIA NORMAL DOME TEMP. = 539. F NORMAL FLOW = 46160. GPM JP EFF. = 37.32 JP M RATIO = 2.622
PRESSURE IMPOSED ON PIPING MAY BE UP TO 13. PSI ABOVE NOMINAL TABLES FOR POINTS 5 THROUGH 16.

POINT	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
W	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	20.00	20.00	20.00	10.00	362.20	724.41
PDOME	6.36	0.13	6.25	4.51	2.19	267.96	267.96	244.76	243.11	241.38	231.19	225.86	222.08	212.41	208.20	39.27	9.65
TDOME	-3.63	-3.64	-3.63	-3.63	-3.63	-2.34	-2.34	-2.37	-2.37	-2.37	-2.38	-2.39	-2.39	-2.41	-2.41	-3.44	1.21

HOT STANDBY - LOW SPEED, FLOW CONTROL VALVE WIDE OPEN

NORMAL DOME PRESS. = 947. PSIA NORMAL DOME TEMP = 538. F NORMAL FLOW = 11485. GPM JP EFF. = 38.82 JP M RATIO = 2.528
PRESSURE IMPOSED ON PIPING MAY BE UP TO 1. PSI ABOVE NOMINAL TABLES FOR POINTS 5 THROUGH 16.

POINT	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
W	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	20.00	20.00	20.00	10.00	352.80	705.59
PDOME	6.36	9.91	20.88	20.77	20.01	36.04	36.04	34.61	34.51	34.40	27.17	26.84	24.17	19.53	19.40	16.85	5.47
TDOME	-0.33	-0.33	-0.31	-0.31	-0.32	-0.22	-0.22	-0.22	-0.22	-0.22	-0.23	-0.23	-0.23	-0.24	-0.24	-0.80	0.69

TABLE G.2.5-3

RECIRCULATION SYSTEM TRIP AND START FUNCTIONS

ACTION					
EVENT	**A**	**B**	**C**	**D**	**E**
1	X		X		X
2	X		X		X
3		X		X	
4					X
5				X	
6	X			X	
7	X			X	
8	X		X		X
9	X			X	
10			X		

Events

1. Suction or discharge block valves less than 90% open.
2. Vessel high pressure (1135 psig; ATWS)
3. Turbine trip or generator load rejection.
4. 100% speed power supply trip of one of two operating recirculation pumps.
5. 100% speed power supply trip of all operating recirculation pumps.
6. Total feedwater flow less than approximately 20% nuclear boiler rated.
7. Temperature difference between the vessel water saturation temperature and recirculation pump suction temperature is less than 10.1^{o}F.
8. Pump motor or LFMG set electrical protection logic is activated.
9. Vessel low level (Level 3).
10. Vessel low-low level (Level 2).

Actions

A. Normal control trip 100% speed power supply.
B. RPT engineered Safety Class 3 trip of 100% speed power supply.
C. Trip of 25% speed power supply.
D. Automatic start of LFMG set during coastdown following a 100% speed trip.
E. No automatic start of LFMG set during coastdown following a 100% speed trip.

TABLE G.2.5-4
SHEET 1 OF 2

REACTOR RECIRCULATION SYSTEM DESIGN CHARACTERISTICS

EXTERNAL LOOPS

Number of loops – two

Pipe Sizes (nominal OD):

Pipe suction – 24 in.
Pump discharge – 24 in.
Discharge minifold – 16 in.
Recirculation inlet line – 12 in.

Design Pressure (psig)/Design Temperature (°F)

Suction piping and valve up to and including pump Suction nozzle – 1250/575
Pump, discharge valves, and piping between – 1650/575
Piping after discharge blocking valve up to vessel – 1550/575
Pump auxiliary piping and cooling water piping – 150/212
Vessel bottom drain – 1275/575

OPERATION AT RATED CONDITIONS

Recirculation Pump:

Flow – 44,983 gpm
Flow- 17.0 x 10^6 lb/hr
Total developed head – 831 ft
Suction pressure (static) – 1023 psia
Required NPSH – 108 ft
Water temperature (max) - 535°F
Pump brake hp – 8189
Flow velocity at pump suction – 39.5 ft/sec

Pump Motor:

Voltage rating – 6600
Speed – 1780 rpm
Motor rating – 8900
Phase – 3
Frequency – 60 Hz
Motor rotor inertia – 21,500 lb/ft^2

TABLE G.2.5-4
SHEET 2 OF 2

Jet Pumps:

Number – 20
Total jet pump flow – 114 x 10^6 lb/hr
Throat ID – 6.4 in.
Diffuser ID – 19 in.
Nozzle ID (5 each) – 1.30 in.
Diffuser exit velocity – 17.1 ft/sec
Jet pump head – 98.3 ft

Flow Control Valve:

Type – Ball
Material – Austenitic S/S
Type actuator – Hydraulic
Valve wide open C_v - 7770
Valve actuator opening and closing rate (min/max)
Percent of stroke per second – 10 to 11
Valve size diameter – 24 in.

Recirculation Loop Block Valves:

Type – Gate valve
Actuator – Motor
Material – Austenitic S/S
Shutoff leakage – 2 cm^3/in/hr
Valve size diameter – 24 in.

TABLE G.2.5-5

PLANT CONFIGURATION DATA

	FLOW PATH LENGTH (in.)	HEIGHT AND LIQUID LEVEL (in.)	ELEVATION OF BOTTOM OF EACH VOLUME* (in.)	MINIMUM FLOW AREAS (ft^2)
A. Lower Plenum	216	216		71.5
		216	-172.5	
B. Core	164	164	44.0	142.0
		164		
C. Upper Plenum and Separators	178	178 178	208.0	49.5
D. Dome (Above Normal Water Level)	312	312 0	386.0	343.5
E. Downcomer Area	321	321	-51.0	79.5
		321		
F. Recirculation Loops and Jet Pumps	108.5 ft (one loop)	403 403	-394.5	132.5 $in.^2$

*Reference point is recirculation nozzle outlet centerline

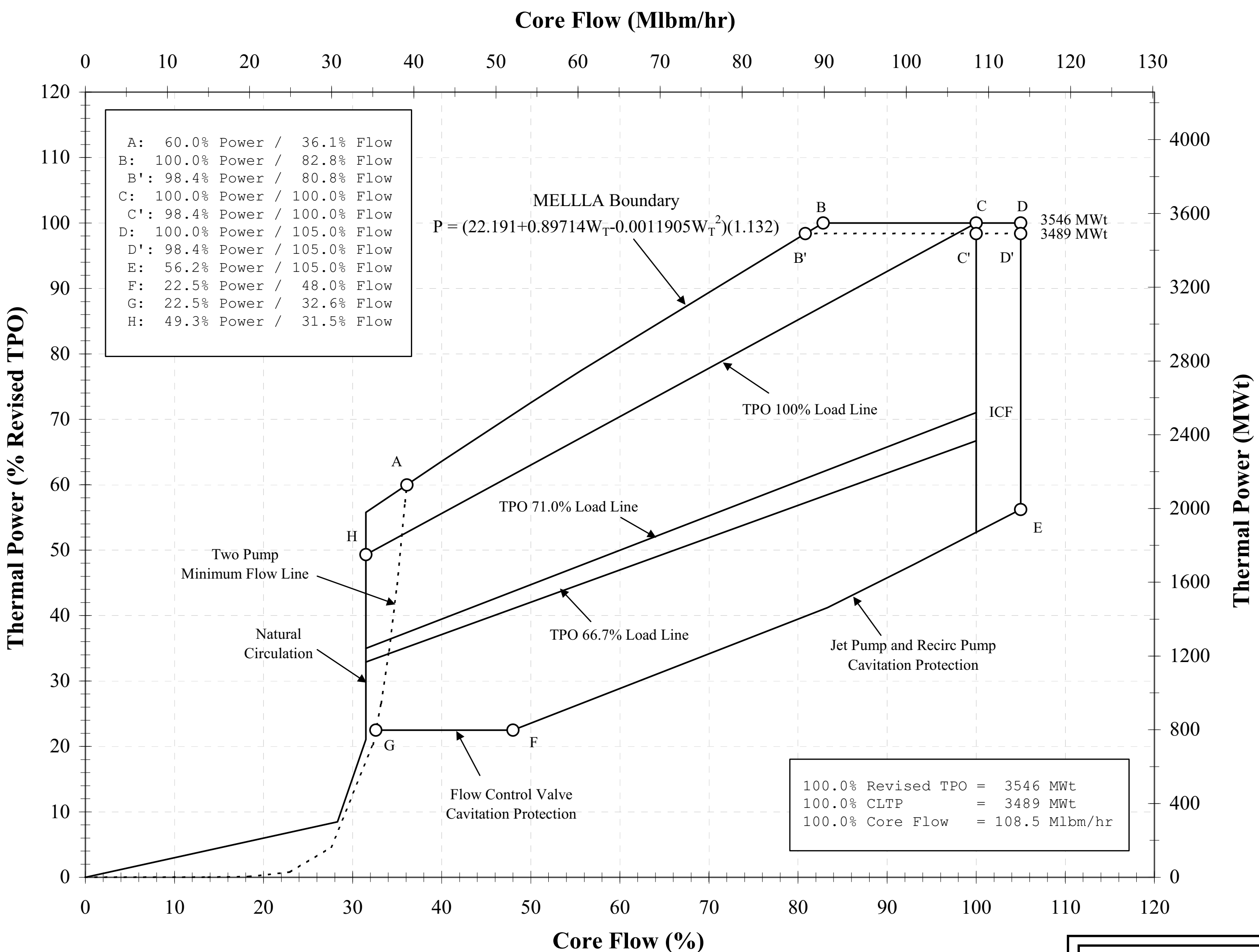

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FIGURE G.2.5-1

TWO-PUMP THERMAL POWER - CORE FLOW OPERATING MAP

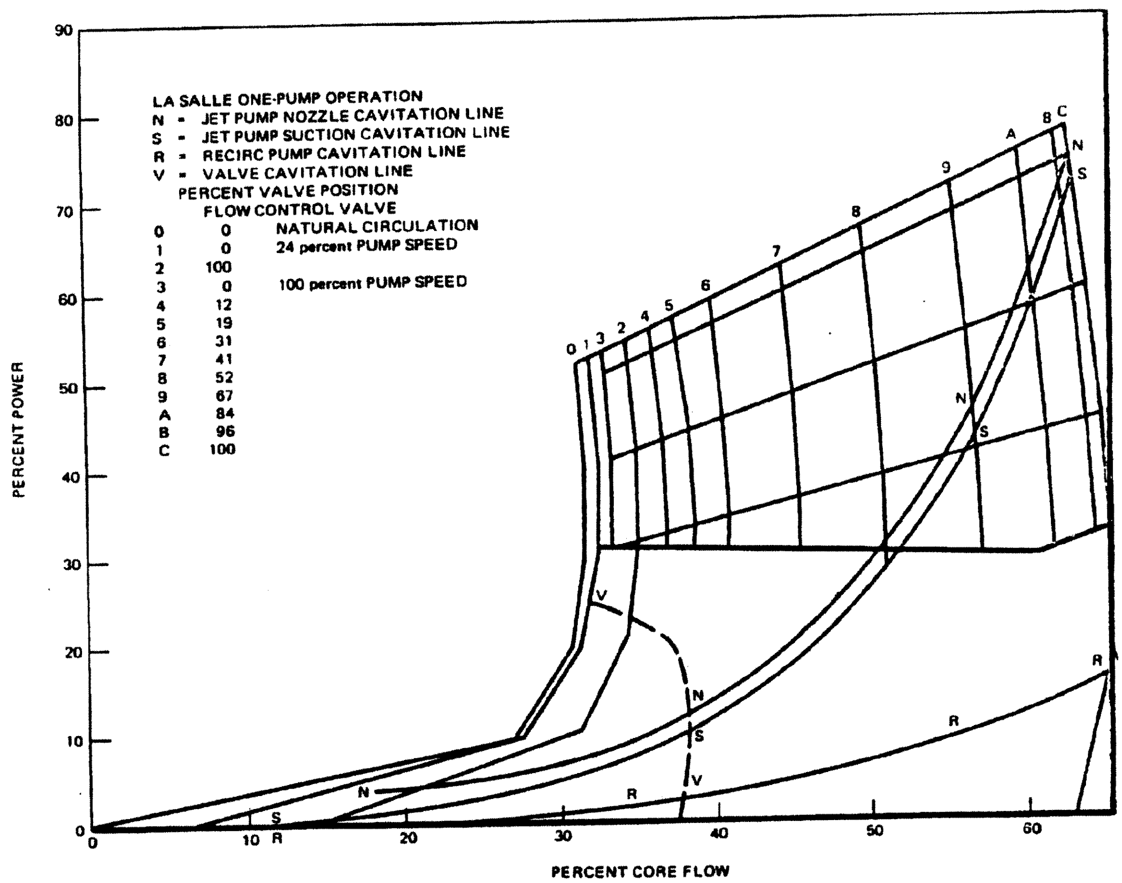

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FIGURE G.2.5-2 Typical ONE-PUMP THERMAL POWER – CORE FLOW OPERATING MAP

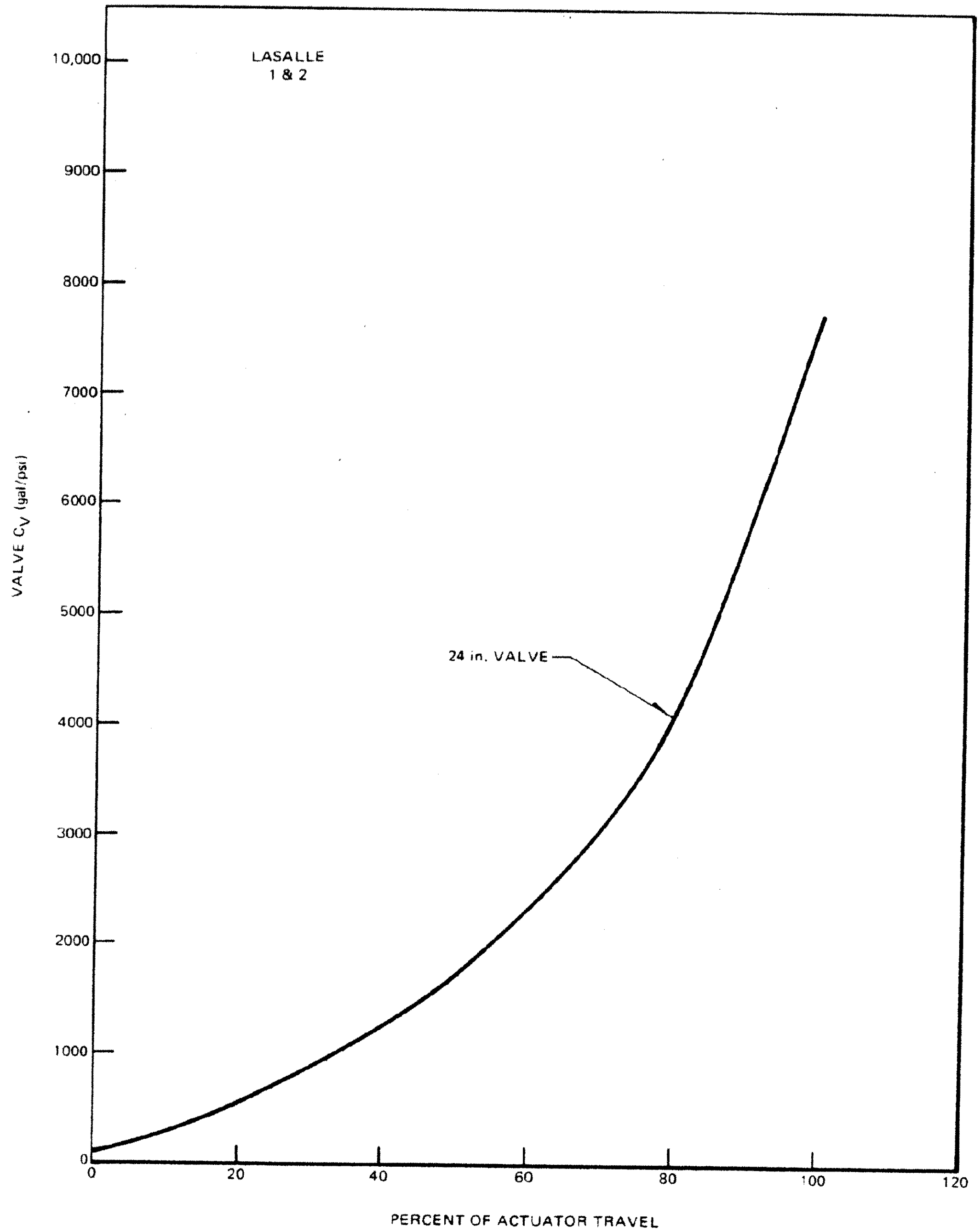

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FIGURE G.2.5-3 FLOW CONTROL VALVE CHARACTERISTIC CURVE

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G.3 SYSTEM DESIGN EVALUATIONS

This section discusses the factors that were considered in determining the system safety requirements (Subsection G.3.1), the system design bases for safety and power generation (Subsection G.3.2), and the system safety evaluation (Subsection G.3.3).

G.3.1 Thermal-Hydraulic Conceptual Design Studies for BWR/5 and BWR/6

Conceptual thermal-hydraulic design studies were performed on the BWR/5 and /6 preliminary system design to determine the system design parameters that would have an effect on the transient and accident event evaluations and to determine system functions that would be necessary to mitigate transients and accidents. Subsequently the individual plant design requirements for the parameters determined from these studies were determined (Subsection G.3.2). Reference 1 of Section G.7 describes the analytical methods.

G.3.1.1 Transients

G.3.1.1.1 Flow Increase Transient

The worst core effects generated by a recirculation system fault came from rapid core flow increases. The following table summarizes the nominal BWR/5 and /6 effects if one flow control valve is stroked to the wide open position at a constant rate.

	30%/sec Valve Stroking Rate *
Initial power	56% NBR
Initial core flow	35% NBR
Reactor scram time	1.15 sec
Peak neutron flux	281.9% NBR
Fuel surface heat flux	79.2% NBR
Peak heat flux time	2.15 sec
Fuel center temperature increase	349° F
ΔCPR	0.19
MCPR	>1.06

* Initial licensing analysis results

Figure G.3.1-1 is a plot of various plant parameters as a function of time and core flow for the 30% per second valve stroking rate. Consequently, it was important to limit valve opening rate in the system design.

G.3.1.1.2 Turbine Trip

Events during which a turbine trip or generator load rejection occurs result in a rapid pressurization condition which could lead to overpower conditions in the core. Trip of the recirculation pumps (RPT) and rapid coastdown during the first few seconds of this event mitigate the neutron flux excursion by creating additional core voids.

In order to act fast enough for this negative reactivity to benefit the event, it is also necessary to sense the event at the turbine stop and control valves just as the reactor protection system does for the scram function and to minimize the recirculation pump motor and pump inertia such that a rapid coastdown will occur when the pump is tripped.

G.3.1.1.3 Trip of One Feedwater Pump

Loss of a feedwater pump capability, as discussed in Section G.2.3.3.2.2, does not produce a severe thermal transient but does cause the vessel level to decrease to the scram setpoints if thermal power level is not decreased to the level that corresponds to the remaining feedwater pump capacity. Runback of the recirculation flow is needed to accomplish this function since a reduction in core flow reduces core power level and steam flow and thus requires less feedwater makeup.

G.3.1.1.4 Other Transients

During all other events, action by the recirculation system is not needed to mitigate the transients. The equipment protective interlocks are presumed to operate when the setpoints are reached, but this action is not critical to the events. Failure of these interlocks in conjunction with the event would be compared to an accident since multiple failures are required. Such events are normally bounded by the consequences of a loss-of-coolant accident.

G.3.1.2 Accidents

G.3.1.2.1 Design-Basis Loss-of-Coolant Accident (DB-LOCA)

For reasons discussed earlier in this UFSAR, the design-basis accident for the emergency core cooling system (ECCS) is an instantaneous double-ended pipe rupture in one of the recirculation loops. During the first few seconds of this postulated event, core flow is maintained by coastdown of the recirculation pump in the unbroken loop. This core flow is important to the subsequent core thermal response; consequently, the recirculation system must be designed so that flow in the unbroken loop is maintained to the extent assumed in the accident analyses. Specifically, high pump inertia is desired, and the unbroken loop recirculation valves must not close at a rate that will affect the coast down flow, as discussed in subsequent sections.

The ECCS design-basis permits no credit to be taken for a-c power other than that provided by the onsite emergency power supplies. Consequently, the recirculation pump in the unbroken loop is assumed to trip and coast down during the postulated DBA. In addition, the DBA analysis assumptions limit recirculation pump flow as a function of quality at the pump suction, and core flow is assumed to cease at the time of jet pump suction uncovery. (These assumptions and their conservatism are fully discussed in General Electric's ECCS analytical models (Initial Cycle discussed in Reference 2 of Section G.7; SAFER/GESTR analysis discussed in Reference 3 of Section G.7. These conservatism and assumptions are addressed in Reference 6 for FANP fuel. The EXEM BWR analysis is discussed in References 7 and 10 of Section G.7.) in conformance with 10 CFR 50 Appendix K. With postulated pipe breaks smaller than the DB-LOCA, the rate of increase in pump suction enthalpy is slower and the time of jet pump suction uncovery is extended; thus there is more credit for coast down flow. Consequently, the general requirement would be to design the recirculation system so that the coast down flow rate is unaffected.

As a measure of the sensitivity of the BWR5/6 DBA-LOCA analysis results to the flow in the unbroken loop, an analysis was conducted to bound the effect. In this analysis, the flow in the unbroken loop was assumed to decrease instantaneously from 100% to 25% concurrent with the pipe break. Instantaneous FCV closure was not physically realistic, since there would be some finite time for the valve to close; however, in order to determine how sensitive PCT was to the flow rate, the assumption served to indicate sensitivity.

The increase in calculated peak cladding temperature with this assumption was on the order of 300° F. Although such an increase is certainly not sufficient to cause a safety concern, it indicates the importance of ensuring through design that the valves would not affect the coast down flow rate.

If a-c power is not lost, the recirculation pumps are expected to trip when the cavitation interlocks are exceeded (low water level, or steamline to recirculation suction temperature) or recirculation pump trip (RPT) is actuated.

G.3.1.2.2 Anticipated Transients Without Scram (ATWS)

Trip of the recirculation pumps from 100% to zero speed would be needed to limit the initial vessel pressure rise and to reduce the steam flow rate to the suppression pool by reducing thermal power. Both recirculation pumps are tripped on high reactor dome pressure or on low-low reactor water level using one-out-of-two taken twice logic.

Both recirculation pumps are tripped on high reactor dome pressure or on low-low water level using one-out-of-two taken twice logic. The sensed variables originate at the level and pressure transmitters of the Alternate Rod Insertion (ARI) system. The trip relays of ARI are common to the ATWS-RPT logic such that initiation of ARI from either or both divisions occurs simultaneously with actuation of ATWS-RPT in the corresponding division(s).

Although the ARI and ATW-RPT logics share the trip relays utilized for automatic actuation, reset of the ATWS-RPT logic is completely independent of the reset of ARI. Manual initiation of ARI is also completely independent of the ATWS-RPT logic and will not actuate ATWS-RPT.

Initiation of an ATWS-RPT in either division annunciates an alarm window in the Main Control Room. The ATWS-RPT signal is sealed-in until reset by deliberate operator action in the Main Control Room.

A spurious level or pressure signal in any one ARI/ATWS-RPT instrument channel will not trip either one or both recirculation pumps. A failure in any one transmitter, trip unit, or trip relay will not result in inadvertent actuation of the ATWS-RPT.

The ATWS-RPT logic is designed to be tested during normal plant operation up to and including the relays that energize the trip coils of the recirculation pump breakers. Test switches are installed for each ATWS-RPT division on Auxiliary Electrical Equipment Room (AEER) panels to avoid inadvertent actuation. The test position of each switch annunciates an alarm window in the Main Control Room. When either switch is in the test position, a trip signal will illuminate an amber light located just above the switch.

Each division of the ATWS-RPT logic is powered by the corresponding 125 VDC divisional bus.

Although the ATWS-RPT equipment does not perform any safety function, NRC Generic Letter 85-06, "Quality Assurance Guidance for ATWS Equipment that is not Safety-Related", indicates special quality requirements must be applied. In accordance with this Generic Letter, the test and reset switches of the ATWS-RPT logic were procured safety-related with seismic qualification.

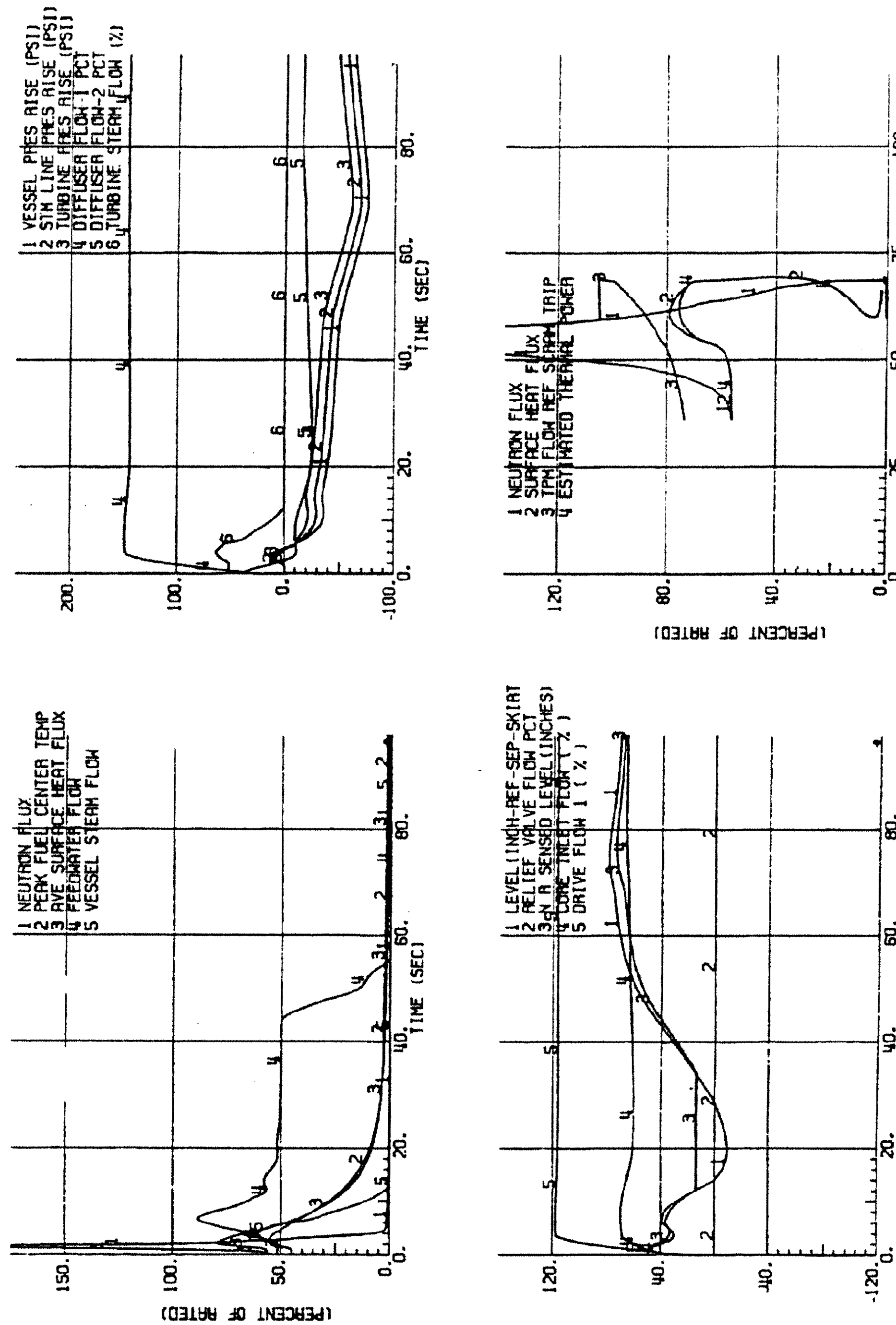

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FIGURE G.3.1-1

(Typical)

TRANSIENT ANALYSIS FIGURES
30%/SEC VALVE STROKING RATE

G.3.2 Design Bases

G.3.2.1 General Design Bases

G.3.2.1.1 Safety Design Bases

The reactor recirculation system is designed to meet the following safety design bases:

a. Provide adequate fuel barrier thermal margin during postulated transients. That is, all transients due to normal operation and to single operator error or equipment malfunction result in a minimum critical power ratio (MCPR) greater than the safety limit MCPR.

b. Failure of piping integrity does not compromise the ability of the reactor vessel internals to provide a refloodable volume.

c. System pressure integrity during abnormal and accident events to minimize consequences of such events.

G.3.2.1.2 Power Generation Design Bases

The reactor recirculation system is designed to meet the following bases in order to maximize plant availability and capacity factor:

a. Ability to achieve rated core flow throughout the design life of the plant.

b. Flexibility to adjust core power output over the range of 65% to 100% thermal power.

c. Minimize maintenance situations that require significant outage time such as those requiring core disassembly and fuel removal.

G.3.2.2 Safety Design Bases

G.3.2.2.1 Normal Operation

The following safety design requirements are met for normal (unfaulted) operation of the system such as flow control, system startup and power ascension:

a. Pressure integrity of all primary system components.

b. Normal flow control valve stroke rate in the open direction is less than effective rate of 30%/sec such that transient MCPR is greater than the MCPR safety limit.

c. One pump restart to return to two pump operation transient MCPR is greater than the MCPR safety limit.

G.3.2.2.2 Transient Operation

The following design requirements are met for all abnormal events, that is, those due to an expected operating transient such as turbine trip, any single equipment failure, or an operator error:

a. Pressure integrity of all primary system components.

b. Flow control valve stroke rate in the open direction is less than the effective rate of 30%/sec such that transient MCPR is greater than the MCPR safety limit.

c. One-pump restart to return to two-pump operation transient MCPR is greater than the MCPR safety limit.

d. Turbine trip:

   1. Trip of the recirculation pumps occurs in less than or equal to 190 msec after initiation of turbine stop or control valve fast closure.

   2. Recirculation Pump-Motor combined inertia is minimal to ensure fast pump coastdown.

G.3.2.2.3 Accident Operation

The following design requirements are met for accident events:

a. Pressure integrity of primary system components such that a loss-of-coolant accident does not result from the event.

b. LOCA:

   1. Unbroken loop valves do not close such as to affect the coastdown flow rate of the recirculation pump.

   2. Integrity of the jet pumps such that a refloodable volume is maintained up to the jet pump throat elevation.

3. Recirculation pump-motor combined inertia is maximized to ensure slow pump coastdown.

c. ATWS: Recirculation pump trip on high vessel pressure or low water level.

d. SSE: Establish containment by providing for isolation of the flow control valve hydraulic lines.

G.3.2.3 Power Generation Design Bases

G.3.2.3.1 General Requirements

Power generation deals with the ability of the plant to stay on line at high capacity. Two general concerns that address this objective are scram avoidance and equipment damage.

Scram Avoidance

Reactor protection system activation makes the plant unavailable. It is desirable to minimize the number of times that a scram is needed by designing with large margins to the RPS setpoints.

Equipment Damage

Damage to system equipment can call for replacement, repair, or inspection that increases outage time and decreases plant availability. It is desirable to avoid situations where damage can occur by including automatic features to protect the equipment.

G.3.2.3.2 Normal Operation

Scram Avoidance

FCV open/close rate is less than 11%/sec.

G.3.2.3.3 Transient Operation

Scram Avoidance Provisions

a. One-pump trip - pump inertia is greater than 21,500 lb_m -ft^2 to allow coastdown without an RPS trip.

b. Flow increase transients - position switch on the FCV prohibits speed increases unless the FCV is in the minimum position.

c. FCV runback given the loss of capability of one feedwater pump and vessel low level.

Equipment Damage Provisions

a. Cavitation interlocks for the recirculation pump, flow control valve and jet pumps: since cavitation produces material damage after long-term operation, and the damage potential decreases with an increase in water temperature, short periods of cavitation during a transient or accident are not a concern. However, long-term operation that might occur due to delayed operator action to trip the pumps is of sufficient concern to call for inspections during the next refueling outage. Consequently, to avoid the need for such inspections, automatic interlocks are installed.

b. Suction/discharge valve closure interlocks to protect the recirculation pump.

c. Electrical interlocks to prevent connection of the LFMG set while the 100%-speed power supply is still active.

d. Loop flow mismatch interlocks to prevent automatic generation of a high flow loop - low flow loop condition after a one-pump trip: this condition is known to produce acceptable but nevertheless abnormally high stresses in the low flow jet pumps and shall not be automatically generated by activating the LFMG set during a one-pump trip. Hence, it would require an operator error to produce this condition.

Other

Activate LFMG in most 100%-speed trip cases to avoid scram recovery delays due to vessel bottom head fluid stratification.

G.3.3 Safety and Power Generation Evaluations

G.3.3.1 Component Classification

The classification of the recirculation system equipment is shown in Table G.3.3-1. Active components are those components that are required to actuate mechanically or electrically during the transients or accidents.

Passive components are those components that are not required to actuate mechanically or electrically during the transients or accidents.

The active/passive designation is not applicable to those components that are not inherently capable of action, such as piping.

G.3.3.2 System Design

G.3.3.2.1 Transient Performance

Reactor recirculation system malfunctions that pose threats of damage to the fuel barrier are described and evaluated in Section G.6. It is shown in Section G.6 that none of the malfunctions results in fuel damage. The recirculation system has sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients.

G.3.3.3 System Component Design

G.3.3.3.1 General

G.3.3.3.1.1 Pressure Integrity (External Loop Piping, Recirculation Pump Bowl, Block Valve Body, Flow Control Valve Body)

Design pressures for the reactor recirculation system are based on peak steam pressure in the reactor dome, appropriate pump head allowances, and the evaluation head above the lowest point in the recirculation loop. Piping and related equipment pressure parts are chosen in accordance with applicable codes. Use of the code design criteria assures that a system designed, built, and operated within design limits has an extremely low probability of failure caused by any known failure mechanism.

G.3.3.3.1.2 Inspection and Testing

Quality control methods are used during fabrication and assembly of the reactor recirculation system to assure that design specifications are met. Inspection and testing are carried out as described in Section G.4. The reactor coolant system is thoroughly cleaned and flushed before initial fuel loading.

During the preoperational test program, the reactor recirculation system was hydrostatically tested at 125% reactor vessel design pressure. Preoperational tests on the reactor recirculation system also included checking operation of the pumps, flow control system (including valves), and gate valves.

During the startup test program, horizontal and vertical motions of the reactor recirculation system piping and equipment were observed; supports were adjusted, as necessary, to assure that components were free to move as designed. Nuclear and recirculation system responses to recirculation pump trips at rated temperature and pressure were evaluated during the startup tests, and plant power response to recirculation flow control was determined.

G.3.3.3.2 Jet Pumps

The core flooding capability of a jet pump design plant is discussed in detail in the emergency core cooling systems document filed with the NRC as a General Electric topical report. The ability to reflood the BWR core to the top of the jet pump applies to all jet pump BWR's and does not depend on the plant size or product line.

G.3.3.3.3 Recirculation Pump and Motor

In order to assure functional performance of the recirculation pump and motor, the following additional requirement is met:

The pump and motor bearings shall have sufficient dynamic load capability at rated operating conditions to withstand the safe shutdown earthquake and to be able to coast down to 40% of rated speed on loss of power.

G.3.3.3.3.1 Analytical Methods for Evaluation of Pump Speed and Bearing Integrity

Tests and procedures used to evaluate critical speed problems in pumps and to assure the integrity of the bearings for the transient conditions are briefly discussed in the following subsections.

G.3.3.3.3.1.1 Pump Shaft Critical Speed

The first critical speed of the recirculation pump shaft has been calculated to be above 130% of the operating speed. The absence of shaft vibration has been verified by testing the pump under rated speed conditions in the supplier's test loop. The absence of vibration was further verified in the plant during preoperational testing.

G.3.3.3.3.1.2 Pump Bearing Integrity

Adequacy of the bearing design has been verified by full temperature and pressure tests in the supplier's test loop.

G.3.3.3.4 Recirculation System Piping

G.3.3.3.4.1 Pipe Rupture

Protection against dynamic effects of pipe rupture is described in Section 3.6 of the UFSAR. Protection has been provided for the postulated break locations and break types that are specified in Subsection 3.6.1.2.

G.3.3.3.5 Suction and Discharge Block Valves

G.3.3.3.5.1 Safety Requirement

Valves shall not close such as to affect the coastdown flow rate of the pump.

G.3.3.3.5.2 Design Description

The 24-inch motor-operated gate valves provide pump and flow control valve isolation during maintenance. The operators take 2 minutes to either fully open or close the valve. The suction valve is capable of closing with up to a 50-psi differential, while the discharge valve can close with up to a 400-psi differential. Both valves are remote-manually operated.

G.3.3.3.5.3 Safety Evaluation

An FMEA performed on the block valves is presented in Attachment G.B.1. In addition, an analysis was made to determine the effect of block valve closure on recirculation pump coastdown. The analysis postulates that coincident with a recirculation pump trip, the block valves begin to close. It was concluded that any closure time greater than 1 minute will have no effect on coastdown times. The consequence of an inadvertent closure without a coincident pump trip is covered in the FMEA, Attachment G.B.1.

G.3.3.3.6 Flow Control Valve, Actuator and Hydraulic Power Unit

G.3.3.3.6.1 Safety Requirements

Worst single failure or operator error shall result in an effective stroke rate of less than 30% of stroke per second in the open direction.

G.3.3.3.6.2 Design Description

G.3.3.3.6.2.1 Flow Control Valves

This is a ball valve with a linkage connected to the actuator shaft (see Subsection G.2.1.2.2).

G.3.3.3.6.2.2 Flow Control Valve Actuator

The actuator is a modified commercial hydraulic cylinder. Deviations from the standard unit include special seals, special rod bearings and the necessary interface dimensional modifications. The cylinder is rated for operation at 3000 psig.

A linear velocity transformer is mounted on the actuator rod and provides the velocity feedback to the flow control system. The position feedback signal is provided by a transducer (Unit 1 only) connected to the ball shaft of the valve.

Hydraulic sealing of the actuator rod is provided by a primary seal which provides the full pressure containment and a secondary seal which prevents any leakage outside the unit. Between the seals is a cavity which has a drain line connected back to the supply reservoir. A flow switch mounted in this line will visually indicate and annunciate a high drain flow condition to the operator.

G.3.3.3.6.2.3 Hydraulic Power Unit

Units are proven commercial devices rated for operation up to 3000 psig. Actual operating pressure is 1800 psig. Qualification of the units required a hydrostatic pressure test at 3000 psig on the high pressure circuits.

The units are located outside the drywell. Only the hydraulic lines, the pilot-operated check valves, and the hydraulic actuator are located inside the drywell.

Refer to Subsection G.5.2.3.4.9 for a complete description of the hydraulic power unit.

G.3.3.3.6.2.4 Hydraulic Line Design

All high-pressure skid piping is ASTM A106, Grade B, Schedule 80. Actual operating pressure is 1800 psig (units are rated by the manufacturer for service up to 3000 psi). The skid equipment and actuator are interconnected using piping and flexible metallic hose. The piping is ASME SA312 Type 304, Schedule 80. The flexible hose is ASTM A240 Type 316L with ASTM A312 Type 304, Schedule 80 welding nipples.

G.3.3.3.6.2.4.1 Design Pressure Rating of Interconnecting Piping

a. Drain: 150 psig

b. Remainder: 3,000 psig

G.3.3.3.6.2.5 Hydraulic Line Isolation Valves

Two solenoid operated automatic globe isolation valves are provided on each hydraulic line, inside and outside the containment, as close to the containment as practical. These valves form part of the containment isolation system for lines that penetrate the primary containment boundary and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere; as such they prevent or limit the escape of fission products from postulated accidents.

G.3.3.3.6.2.6 Hydraulic Fluid

The hydraulic fluid used is Fyrquel EHC. It has an autoignition temperature of 1150° F. When heated to 455° F, the vapor will flash with an ignition source. If heated to 700° F, burning of vapor will last at least five seconds after ignition.

G.3.3.3.6.3 Safety Evaluation

G.3.3.3.6.3.1 Flow Control Valve

G.3.3.3.6.3.2 Flow Control Valve Seizure: Pressure Boundary Integrity

The Flow Control Valve (FCV) with the enclosed topworks consists of the body, the bonnet, shaft/ball assembly, the upper cover, the packing cartridge, the yoke and the actuator with the hydraulic lines.

The forces required to actuate the valve vary with the position:

a. To open the valve, the extend force varies from ~19,000 lb to ~11,500 lb.

b. To close the valve, the force varies from ~11,000 (retract) to ~8,000 lb (extend) due to the flow torque force.

Should the valve ball seize, the hydraulic actuator is capable of producing a maximum force of 50,000 lb. The most severely stressed part would be:

a. The link. At the maximum actuator force the stress would be about equal to the yield stress.

All other parts are stressed below their yield points, and no failure is therefore expected.

The result of the above failures would not affect the pressure integrity of the valve or allow the ball to break away from the shaft.

G.3.3.3.6.3.3 Hydraulic Actuator and Hydraulic Power Unit

G.3.3.3.6.3.3.1 Actuator Velocity Limits

Actuator velocity is normally controlled by the electronic circuits to limit any sustained motion to 10%/sec. A "hard-over" servovalve malfunction and/or an electronic malfunction which applies a hard-over signal to the servovalve will result in higher velocities, referred to as "excess" velocity.

When this occurs, velocity is limited initially only by the restriction of the hydraulic circuit. Once the accumulator has discharged all the fluid stored in it, velocity is limited by the pump flow capacity. Due to high friction forces, the FCV will remain motionless unless driven by the actuator.

Accordingly, two velocities are of interest, the peak instantaneous velocity which occurs while the accumulator is discharging, and the average velocity (i.e., full stroke/time to travel full stroke).

Since the actuator incorporates unequal areas, velocities must be calculated for both direction of motion (all analysis assumes failure of all interlocks which might shut the unit down).

The worst single failure, servovalve hard-over, will result in the following actuator transients:

a. peak velocity during accumulator discharge;

b. peak extending velocity ≤62%/sec (extending is in the direction to open the FCV); and

c. peak retracting velocity ≤54%/sec (retracting is in the direction to close the FCV).

For the rod extending event, the accumulator discharge lasts for 0.14 seconds, while the retracting duration is 0.22 seconds.

Average velocity over full stroke (including accumulator discharge):

a. extending ≤18%/sec (open FCV)

b. retracting ≤25%/sec (close FCV)

G.3.3.3.6.3.3.2 Summary of Failure Analysis

A failure mode and effects analysis (FMEA) for the flow control valve and the hydraulic power unit are given in Attachment G.B.2.

For the flow control valve, there are two major limiting single features. One is valve failure which is bounded by the LOCA event; the second is a momentary stroke rate that exceeds an effective 30%/sec average rate. This latter value is the average continuous rate used in the transient analysis such that the transient MCPR is greater than the safety limit MCPR. The single failure that results in the highest stroke rate was determined to be a hard-over servovalve. A conservative design analysis produces a stroke transient as shown in Figure G.3.3-1. This transient was analyzed and found to be less severe than the effective 30%/sec rate.

Multiple failures, such as high pump pressure, servovalve hard-over, or loss of pressure, can result in instantaneous velocities exceeding 60%/sec. These events are considered to be in the same category as accident events.

G.3.3.3.7 Flow Control System

G.3.3.3.7.1 Safety Requirements

Worst single failure or operator error shall result in an effective stroke rate of less than 30% of stroke per second in the open direction.

The control system shall not close the flow control valve in the unbroken loop at a rate that will affect the LOCA recirculation pump coastdown.

G.3.3.3.7.2 Design Description

During normal operation, the recirculation flow control system provides the recirculation flow demand signal to the hydraulic power unit (HPU) servovalves.

The flow control system has been designed to limit the maximum demand signal to an actuator stroke rate of less than 11%/sec. This is to ensure that the reactor protection system will not initiate scram. Various interlocks are installed to ensure that this limit is not exceeded.

The design is discussed in more detail in Subsection G.5.2.

G.3.3.3.7.3 Safety Evaluation

G.3.3.3.7.3.1 Flow Control Valve Stroke Rate

A failure modes and effects analysis (FMEA) for the flow control system is given in Attachment G.B.3.

Single failures can result in an actuator stroke rate exceeding 11%/sec, which may result in reactor protection system activation. However, there are no single failures of the flow control system which would result in an effective stroke rate exceeding 30%/sec. This value is the average continuous rate used in the transient analysis.

The worst single failure would be a single demand for a hard-over servovalve. A conservative design analysis produces a stroke transient as shown in Figure G.3.3-1. This transient was analyzed and found to be less severe than the effective 30%/sec rate.

G.3.3.3.7.3.2 Loss-of-Coolant Accident

For postulated pipe breaks in the recirculation system, only the response of the valve in the unbroken loop is of concern. The event would have the following effects on the system:

The recirculation flow loops would not interact with one another. Thus, the unbroken loop would see no significant change.

The valve in the unbroken recirculation line will not move in response to a recirculation loop pipe break.

G.3.3.3.8 Recirculation Pump Trip (RPT) System

G.3.3.3.8.1 Safety Requirements

The RPT system is required to trip both recirculation pumps from their normal power source within 190 msec after a turbine/generator trip or load rejection event occurs with reactor power level greater than or equal to 25% of rated core thermal power. This trip cannot be prevented by any single component failure in the system.

G.3.3.3.8.2 RPT Design

The RPT logic is derived from relay's in the reactor protection system (RPS) which are activated by turbine stop valve limit switches and turbine control valve oil pressure switches. Contacts from these relays are arranged in a logic scheme so that deenergizing various combinations of the relays will energize the RPT trip coil in each of two main power source circuit breakers (see Figure G.3.3-2).

Each main circuit breaker has two trip coils. One trip coil is activated by an RPT signal only. The other trip coil is activated by all other breaker trip functions. The dual trip coils serve to separate the breaker safety function from its non-safety functions.

The design of the sensor/logic/circuit breaker chain is such that the response time between initiation signal and suppression of circuit breaker arc is less than 190 msec.

G.3.3.3.8.3 Failure Mode and Effects Analysis (FMEA)

An FMEA was performed for each power supply, fuse, sensor/relay pair, test switch, diode, test lamp, indicating lamp and circuit breaker in the RPT logic circuit. The detailed FMEA sheets are presented in Attachment G.B.4. The results of this FMEA are as follows:

a. A fail-open or fail-closed single random failure of individual fuses, trip sensors or relays in Channel A1, A2, B1, or B2 will not prevent or initiate RPT.

b. A fail-open or fail-closed single random failure of power supplies, test switches, diodes or test lamps in trip channels A or B will not prevent or initiate RPT.

c. A fail-open in an individual circuit breaker status indicating lamp or a fail-closed of an individual circuit breaker will not prevent RPT when required.

d. A short circuit in an individual circuit breaker status indicating lamp or fail-open of an individual circuit breaker will initiate RPT in a single loop.

G.3.3.3.9 Low-Frequency Motor-Generator (LFMG) Set

G.3.3.3.9.1 Special Design Requirements

G.3.3.3.9.1.1 Safety

The LFMG set is nonessential equipment and has no safety function. However, design requirement is that the LFMG set cannot interfere with a recirculation system safety function (e.g., RPT).

G.3.3.3.9.1.2 Power Generation

The basic design requirements for the LFMG set are power generation oriented, that is, the purpose of the LFMG set is to prevent cavitation of the flow control valve at low recirculation flows while maintaining enough flow to prevent reactor pressure vessel temperature stratification. Thus, most events that cause the recirculation pumps to trip from the main power source will also start the LFMG set and close it in at 25% pump speed to maintain circulation.

G.3.3.3.9.2 Design Parameters

G.3.3.3.9.2.1 General

A single-line diagram of the recirculation pump motor power sources is shown in Figure G.3.3-3. The pump motors can be supplied from a nonessential 60-Hz power source through two essential class 1E, Seismic Category I, Safety Class 3 circuit breakers or from the 15-Hz output of the nonessential LFMG set. Interlocks are provided to prevent powering of the pump motor from both power sources simultaneously.

G.3.3.3.9.2.2 Interlocks

During plant startup, if it is necessary to start a recirculation pump and reactor power is less than 25% of rated power/20% feedwater flow (FCV cavitation interlock), a start signal to the pump motor closes the main power source breaker, accelerating the pump from zero to 100% speed, starts the LFMG set, trips the main power breaker at 100% speed, and closes in the LFMG set generator breaker when the pump coasts down to 25% speed.

Once reactor power is greater than 25%, the pumps may be transferred to the main power source. The transfer can only be accomplished if the flow control valve is at its minimum position. This prevents a large flow increase transient. The power transfer signal trips the LFMG set and, once the generator breaker is open, closes the main power source breaker which causes the pump to accelerate from 25% to 100% speed. The FCV can then be used to control flow.

During a plant shutdown, when reactor power level drops below 25%, with recirculation flow less than 60%, operating procedures and automatic interlocks require that the recirculation pumps be tripped from their main power source and powered from the LFMG set.

In addition, the LFMG set will automatically start and close in at 25% speed following a recirculation pump trip due to RPT, normal trip of both recirculation pumps, low temperature differential between main steam and recirculation pump suction, or vessel low level.

The LFMG set will not automatically start and close in if the recirculation pump trip is due to suction or discharge valve less than 90% open, vessel high pressure, normal trip of one of two operating pumps, or if the LFMG set electrical protection logic is activated.

G.3.3.3.9.3 Failure Mode and Effects Analysis (FMEA)

An FMEA was performed on the LFMG components and interlocks. The details of the component FMEA are presented in Subsection G.3.3.3.11. The details of the interlock FMEA are presented in Subsection G.3.3.3.10.

These FMEA's show that the LFMG set cannot interfere with the RPT safety function and that the system interlocks are sufficient to prevent FCV cavitation event with a single random failure (i.e., operator error or equipment failure).

G.3.3.3.10 Control Interlocks

G.3.3.3.10.1 Power-Generation Design Requirements

Operating conditions have been identified which could result in cavitation of the flow control valve, recirculation pump, and/or jet pumps. Operating procedures require the plant operator to avoid these conditions. In the event that, due to operator error, these conditions are not avoided, automatic interlocks exist which will prevent operation at these conditions.

G.3.3.3.10.2 Design Description

G.3.3.3.10.2.1 Reactor Low Power Level - LFMG Set Auto Sequence

When reactor power level is not high enough (~25%) to provide sufficient subcooling to prevent cavitation of the flow control valve, a start of the recirculation pumps will result in closure of the main power source breaker to accelerate the pump from zero to 100% speed, start of the LFMG set, trip of the main power source breaker when the pump reaches 100% speed and closure of the LFMG output breaker when the pump reaches 25% speed. The start can only occur if the recirculation suction and discharge valves are open, the FCV is at minimum position, the M/A station is in manual, control power is available to the LFMG set breakers and the electrical protection relays on the LFMG set have not actuated.

G.3.3.3.10.2.2 Reactor Normal Power Level Low-Speed to High-Speed Transfer

Once reactor power level is greater than 25%(20% feedwater flow), the operator may transfer the recirculation pumps to the main power source from the LFMG set or, if the pumps are tripped, they may be started by the main power source and will not transfer back to the LFMG set. These sequences can only occur if the suction and discharge valves are fully open, the FCV is at minimum position, the M/A station is in manual, pump speed is less than 20%, feedwater flow is normal, reactor water level is normal, vessel water saturation temperature to recirculation suction ΔT is normal, and RPT has not occurred.

G.3.3.3.10.2.3 High-Speed to Low-Speed Auto Transfer Sequence

During plant shutdown, when reactor power level approaches 30% decreasing, the operator will transfer the recirculation pumps from high-speed to low-speed operation. Interlocks require this transfer to be performed on both pumps simultaneously, thus avoiding an undesirable loop flow mismatch. Placing the control switch for both pumps into the transfer position automatically starts both LFMG sets, trips the main power source breakers, and closes the LFMG generator breakers when the pumps reach 25% speed. High to low speed transfer will automatically occur on RPT, less than 20% feedwater flow, reactor water low level, and low ΔT between vessel water saturation temperature to recirculation suction temperature. During high-to-low speed transfer, a tripped pump will not be energized by the LFMG set at 25% speed if the other pump has failed to trip.

G.3.3.3.10.3 Failure Mode and Effects Analysis (FMEA)

An FMEA was performed on the recirculation system control interlocks. The detailed FMEA sheets are presented in Attachment G.B.5. The results of this FMEA are as follows:

> No single random failure (i.e. operator action or equipment failure) will result in sustained operation in a region which will produce cavitation of the flow control valve, jet pumps, and/or recirculation pumps.

G.3.3.3.11 Power Supplies

G.3.3.3.11.1 Design Requirements

The recirculation system power supplies are designed so that no single failure of a power supply can prevent the action of the RPT safety function, nor can the single failure of a power supply prevent the protection of equipment required under the power-generation design-basis.

G.3.3.3.11.2 Power Supply Arrangement

The arrangement of the various power supplies needed to power the recirculation system is shown in Figure G.3.3-4. A single-line diagram of the auxiliary power system showing the recirculation pump motor and LFMG set power supplies, the 120-Vac instrument bus, and the 125-Vdc control power to the 4.16-kV and 6.9kV circuit breakers is presented on Figure G.3.3-5.

G.3.3.3.11.3 Failure Mode and Effects Analysis (FMEA)

An FMEA was performed for each power supply or circuit breaker which has an effect on the safety design and power generation design bases for the recirculation system. The detailed FMEA sheets are presented in Attachment G.B.6. The results of this FMEA are as follows:

No single random failure of a power supply or circuit breaker will prevent the action of the RPT safety function, nor can the single failure of a power supply or circuit breaker prevent the protection of equipment required under the power generation design-basis.

Multiple failures are required to parallel the LFMG set with the main power source. Even if these multiple failures occurred, the ability of RPT to function would not be impaired, although the LFMG set could be damaged.

TABLE G.3.3-1

RECIRCULATION SYSTEM EQUIPMENT CALASSIFICATION

COMPONENT	SAFETY FUNCTION	SAFETY CLASS	SEISMIC CATEGORY	FUNCTION
External loop piping	NA	1	I	Pressure integrity
Jet pumps	Passive	1	I	Core reflood height
Recirculation Pump	Passive	1	I	Pressure, Integrity, bearings
Recirculation Pump motor	Passive	2	I	Bearings only
Block valves F023/F067	Passive	1	I	Pressure integrity
Block valve motor operator	Passive	NA	NA	
Flow control valve F060	Passive	1	I	Pressure integrity
Flow control valve actuation:				
a. hydraulic actuator;	Passive	NA	NA	
b. hydraulic lines (inside containment);	Passive	NA	NA	
c. hydraulic power unit;	Passive	NA	NA	
d. control system	Passive	NA	NA	
Hydraulic line isolation valves	Active	2	I	Containment isolation
RPT breakers and controls	Active	3	I	Turbine trip
All other controls	Passive	NA	* NA	ATWS, system starts, trips
LFMG	Passive	NA	NA	

* Digital control components are seismically mounted.

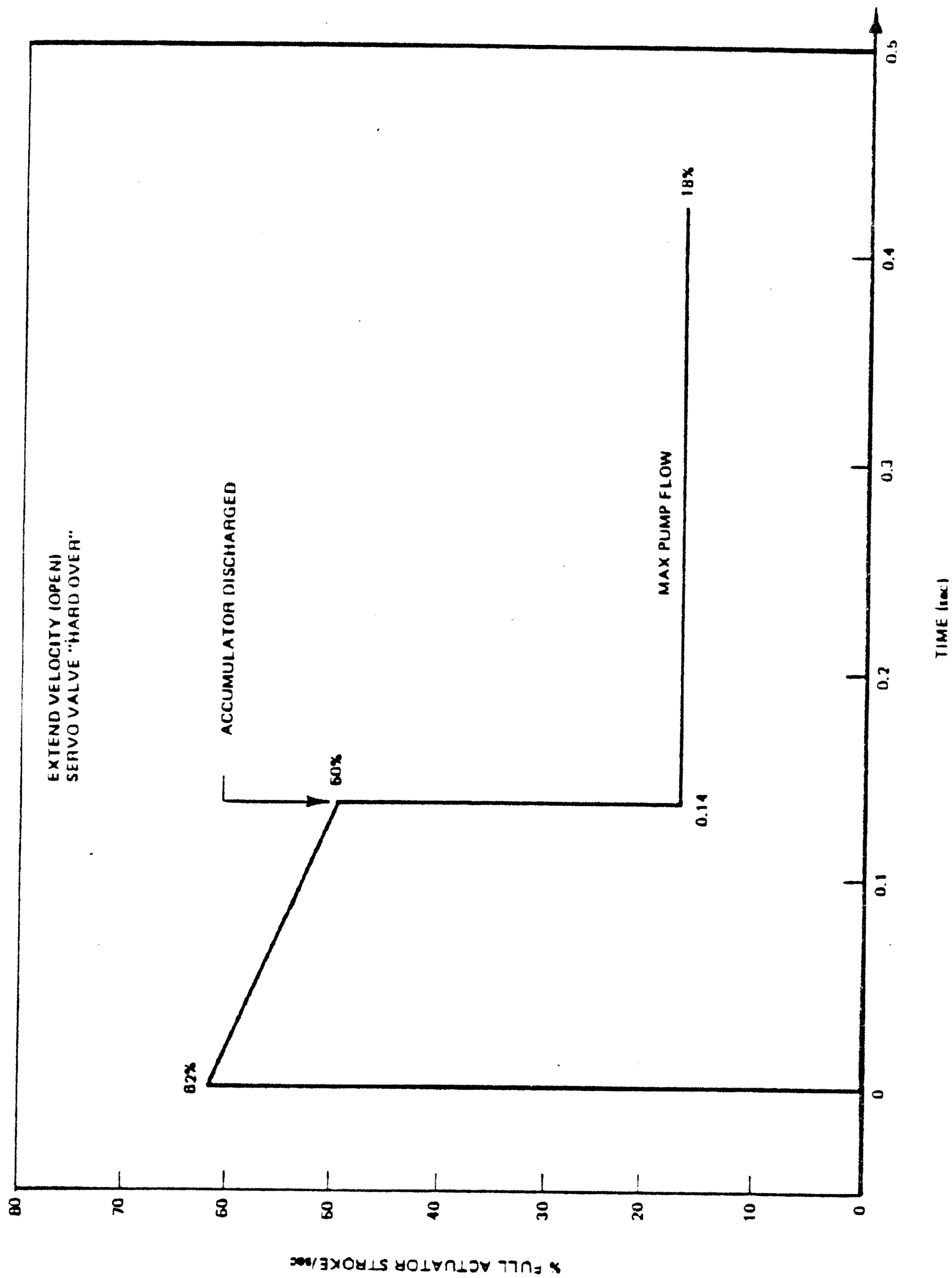

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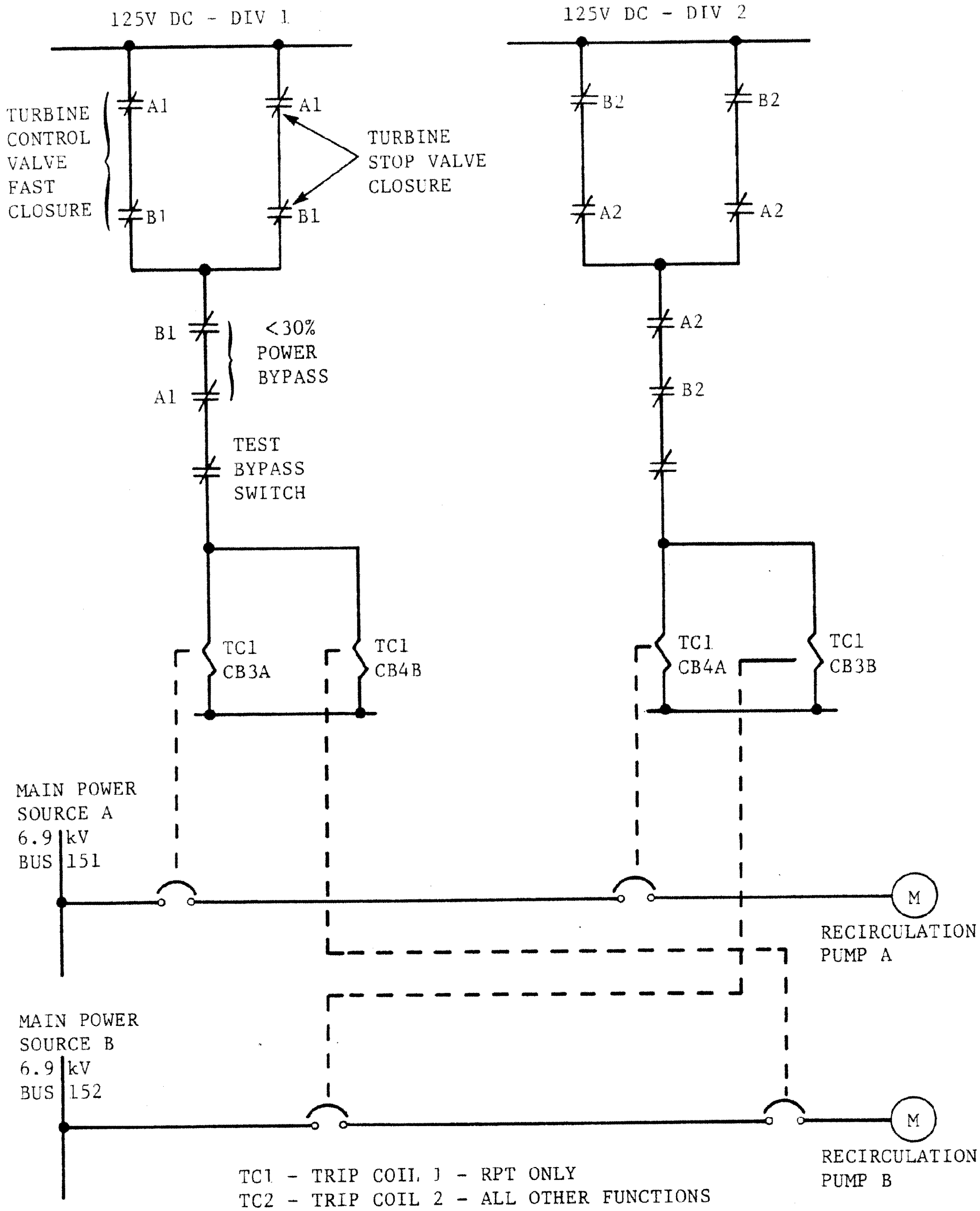
FIGURE G.3.3-1

HIGHEST SINGLE-FAILURE STROKE RATE



TC1 - TRIP COIL 1 - RPT ONLY
TC2 - TRIP COIL 2 - ALL OTHER FUNCTIONS

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FIGURE G.3.3-2 RPT LOGIC DIAGRAM

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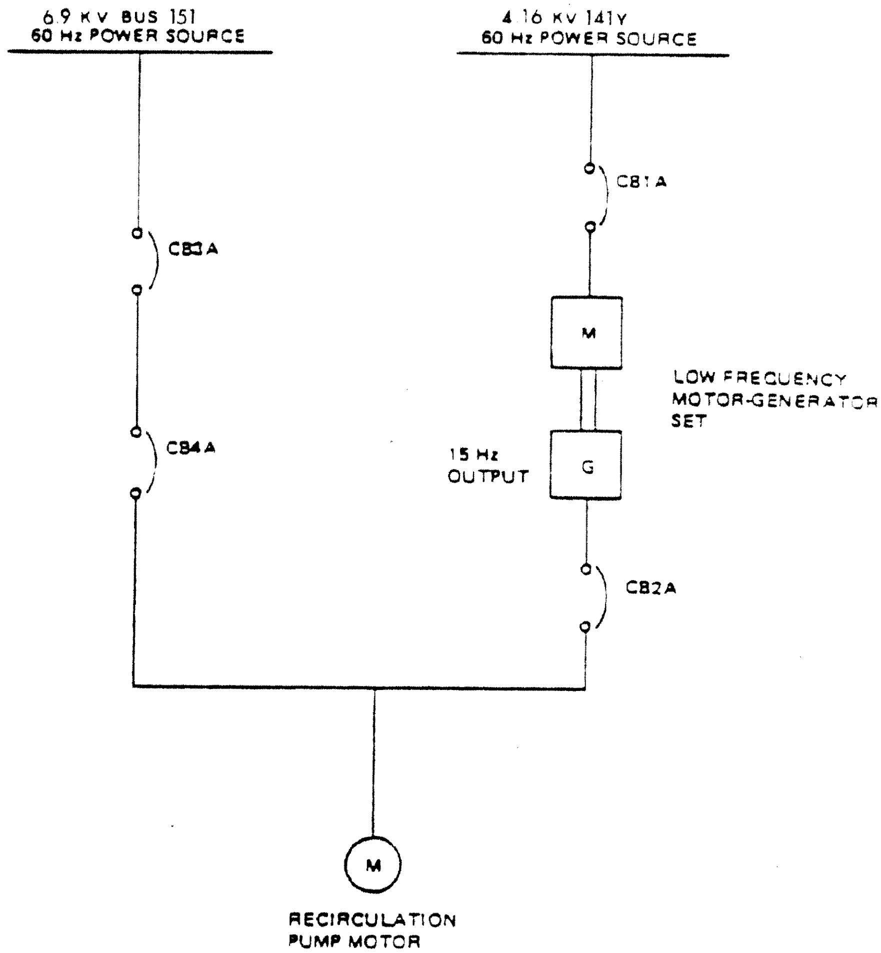

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FIGURE G.3.3-3

BREAKER DESIGNATION

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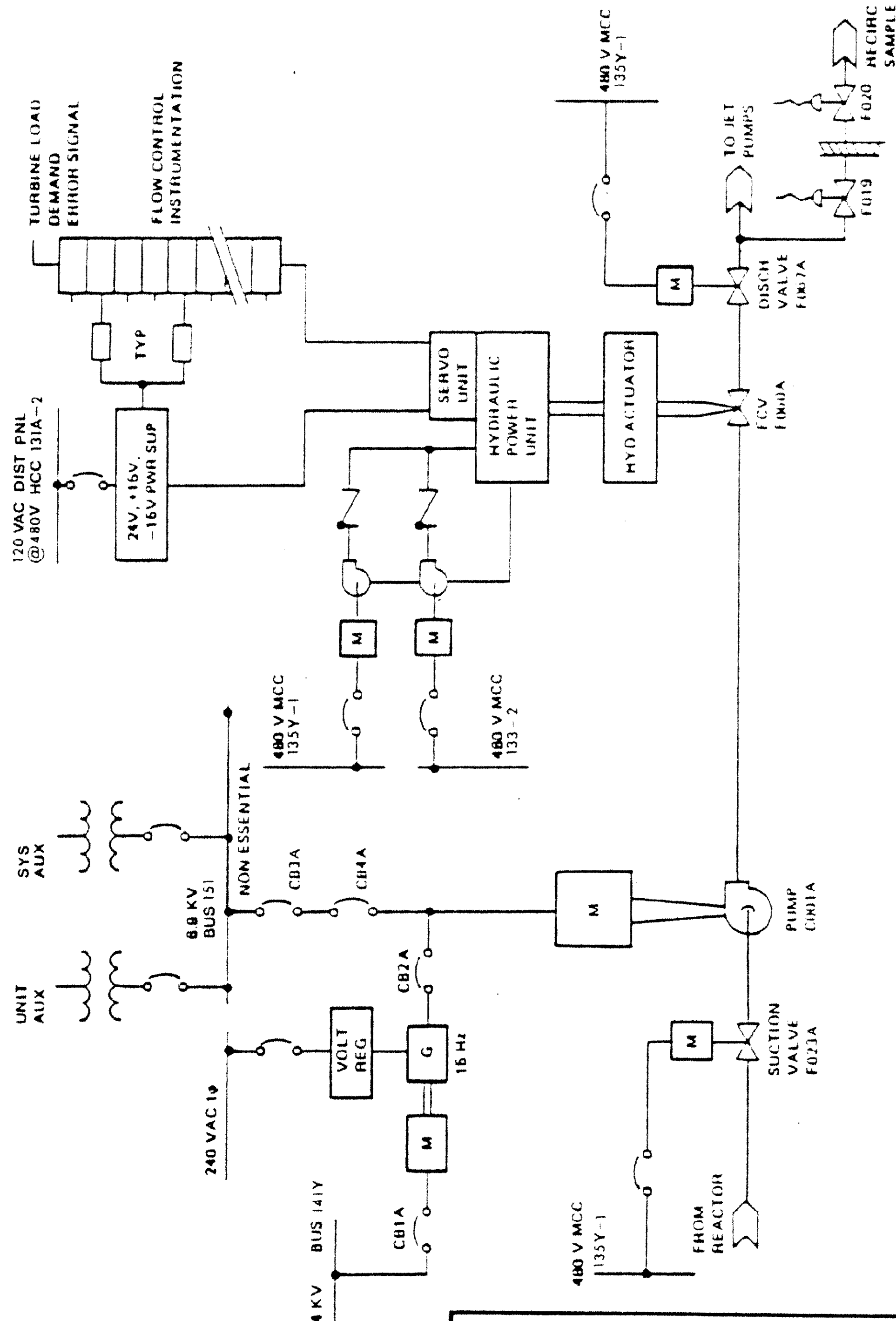

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FIGURE G.3.3-4

POWER SUPPLY DEVELOPMENT "A"
RECIRCULATION LOOP

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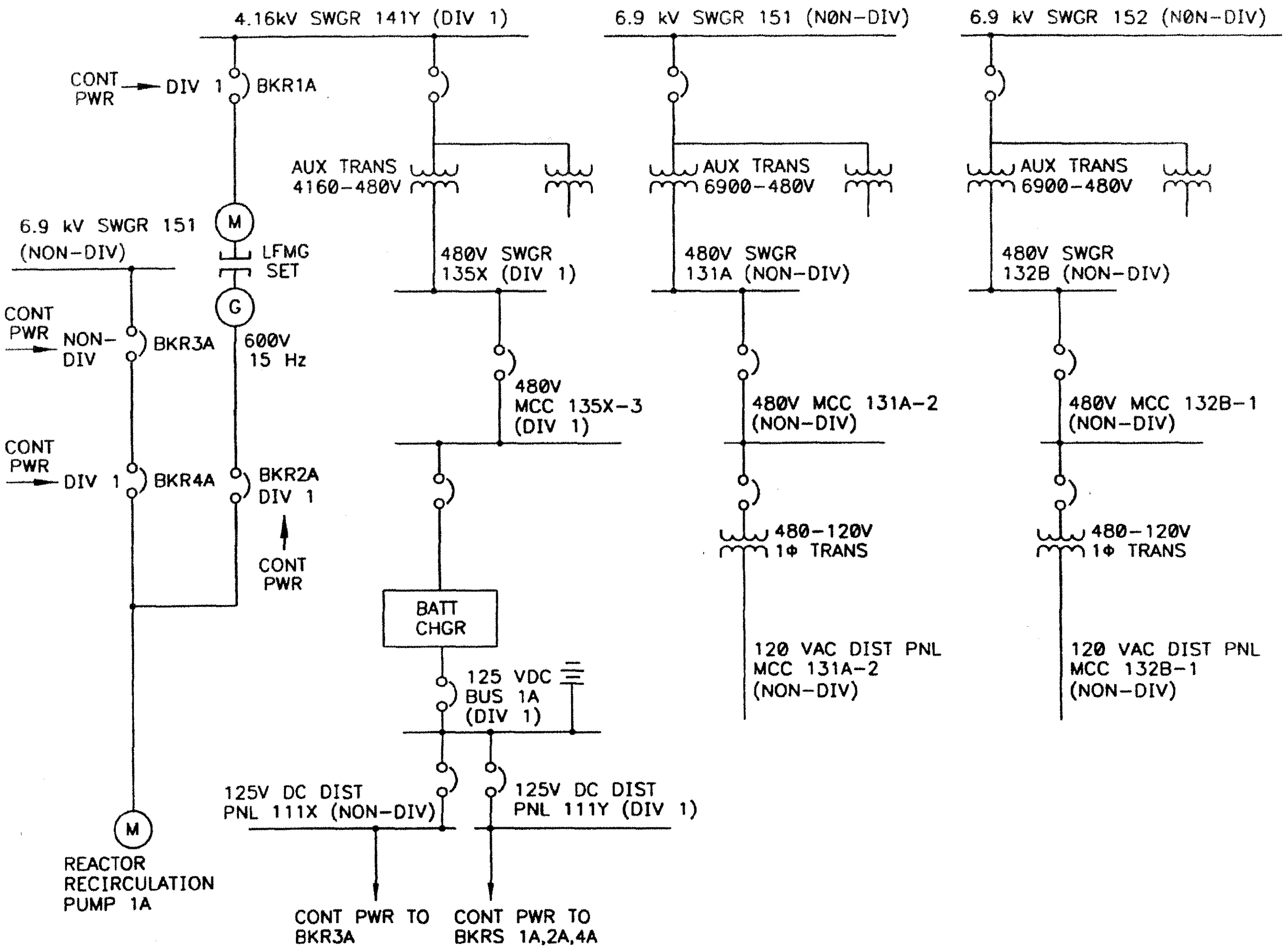

1. ANNUNCIATION IS PROVIDED IF ANY OF THE BREAKERS FEEDING THE 125 VDC DISTRIBUTION PANELS ARE TRIPPED (AUTOMATICALLY).
2. UNDERVOLTAGE MONITORING IS PROVIDED AT THE 125 VDC BUS AND THE CONTROL ROOM.
3. INCOMING 125 VDC CONTROL POWER FAILURE AT THE SWITCHGEAR OR SUB-STATIONS ARE ANNUNCIATED ON UNDERVOLTAGE.
4. THE LAYOUT OF DIVISION 2 CONTROL POWER IS SIMILAR TO THE DIVISION 1 SHOWN ABOVE AND IS PROVIDED WITH SAME ALARMS AND MONITORING.

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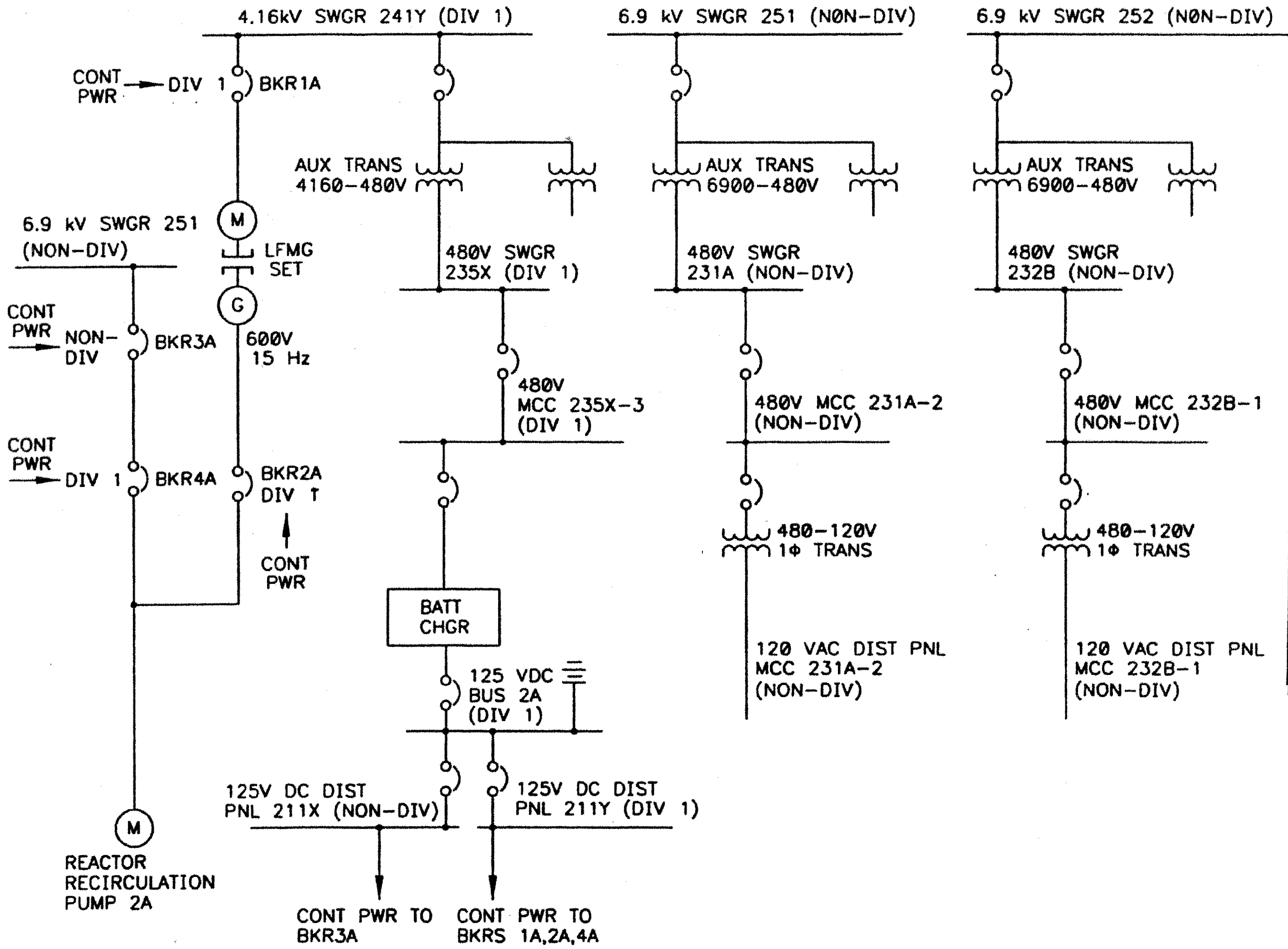
FIGURE G.3.3-5 LAYOUT OF AUXILIARY POWER SYSTEM FOR UNIT 1 REACTOR RECIRCULATION SYSTEM SHEET 1 OF 2



1. ANNUNCIATION IS PROVIDED IF ANY OF THE BREAKERS FEEDING THE 125 VDC DISTRIBUTION PANELS ARE TRIPPED (AUTOMATICALLY).
2. UNDERVOLTAGE MONITORING IS PROVIDED AT THE 125 VDC BUS AND THE CONTROL ROOM.
3. INCOMING 125 VDC CONTROL POWER FAILURE AT THE SWITCHGEAR OR SUB-STATIONS ARE ANNUNCIATED ON UNDERVOLTAGE.
4. THE LAYOUT OF DIVISION 2 CONTROL POWER IS SIMILAR TO THE DIVISION 1 SHOWN ABOVE AND IS PROVIDED WITH SAME ALARMS AND MONITORING.

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FIGURE G.3.3-5 LAYOUT OF AUXILIARY POWER SYSTEM FOR UNIT 2 REACTOR RECIRCULATION SYSTEM SHEET 2 OF 2

G.4 TESTING

This section describes the development tests, preoperational tests and startup tests conducted on individual recirculation system components and the entire system.

G.4.1 Development Tests

G.4.1.1 Five-Nozzle Jet Pumps

G.4.1.1.1 Development of Jet Pumps

G.4.1.1.1.1 Improvement and Evaluation of Performance

Initial jet pump feasibility tests were performed in 1962 when the first quarter-scale jet pump was designed, built, and tested at reactor operating conditions. This jet pump was designed using information available in literature and performed with a peak efficiency of 30% at reactor conditions. Subsequent work was directed toward increasing jet pump efficiency by developing improved components using quarter-scale models. A streamlined nozzle and a streamlined suction inlet with low losses were developed. In addition, nozzle-to-suction inlet spacing and the mixing throat length were optimized for the geometry intended for reactor application.

This development program produced a jet pump that performed with 32% equivalent reactor peak efficiency. In mid-1967 a full-size prototype jet pump was tested at reactor conditions to demonstrate that it performed as well as or better than the quarter-scale model previously tested. The quarter-scale model was scaled linearly to full size. The full-size prototype performed with a peak application efficiency of 34%. Four production models for the Dresden-2 plant were tested extensively at reactor conditions to confirm that performance fully met requirements and to calibrate the diffuser pressure differential for use in measuring reactor core flow.

The prototype jet pump design was refined somewhat (partly to improve performance) when the Dresden-2 jet pump was designed. Thus, the design used for the Dresden-2 jet pump is considered as the reference, first-generation design for BWR's. It is used here for comparison with the improved, second-generation jet pumps.

Jet pumps for reactors are designed with substantial margins to avoid cavitation during normal operation. To acquire a thorough understanding of jet pump operation, tests were performed with the full-size prototype and Dresden-2 jet pumps at off-design conditions which produced cavitation. This was accomplished by reducing the subcooling in steps to zero and then injecting measured amounts of steam into the recirculation flow upstream of the jet pump suction inlet to simulate steam carryunder from steam separators at zero subcooling.

The tests show that jet pump efficiency is not significantly reduced as cavitation is increased. No abrupt change in performance is produced by cavitation, and suction flow rate may be increased when cavitation is present. Even when suction flow contained 0.06% steam by weight (in thermodynamic equilibrium), the dropoff in efficiency was less than 10% of its value at design conditions. No changes in the well-polished jet pump surfaces exposed to flow were observed even after many hours of cavitation testing.

G.4.1.1.1.2 First-Generation Jet Pumps

The main features of the Dresden-2 jet pump are: an inlet pipe equipped with a turning vane in the 180° bend; an efficient single-jet converging drive flow nozzle with a 3.3-inch outlet tip diameter; an elliptical suction inlet blended into the mixing throat; a mixing throat 8.1-inch ID and 10.5 diameter long; a conical diffuser with an included angle of 8° and an area ratio of 6:1 which is blended into the throat and into the tailpipe; and a tailpipe approximately one diameter in length.

The 180° bend is equipped with a single, full passage width turning vane (based on GE development tests) to minimize bend losses. The nozzle and suction inlet are finished from a single casting, including the three struts which maintain the drive nozzle and the suction inlet in alignment. These struts are streamlined to minimize losses at the suction inlet. This combination and arrangement of components results in a jet pump that performs well at reactor conditions.

When the performance of the Dresden-2 jet pump, as well as all other jet pumps, was applied to plant design, sufficient margin was allowed to provide for the possible influence in such factors as manufacturing and installation tolerances, flow distribution patterns in a reactor, and surface finish changes during reactor lifetime. Confirmatory data from Dresden-2 and other plants using similar jet pumps show that jet pump performance in plants meets or exceeds the performance requirements established from production jet pump tests of the type discussed here.

G.4.1.1.1.3 Second-Generation Jet Pumps

There was substantial incentive to improve the efficiency of BWR jet pumps, because as their efficiency is increased, plant heat rate is improved, and the cost and size of recirculation pumps and motors used to drive the jet pumps are reduced. Therefore, there has been a continuing development effort to improve the efficiency of jet pumps. This effort has been directed toward optimizing jet pump components and introducing new components or geometry configurations. One of the components investigated, for example, has been the multijet drive nozzle.

The multijet nozzle provides more surface area for mixing of the drive and suction flow streams than does a single-jet nozzle of the same cross-sectional area. The mixing of the two streams can therefore be accomplished in a shorter throat. Efficiency is improved because comparatively the friction losses in the throat are reduced to a greater extent than they are increased by using the multijet nozzle. The multijet nozzle also produces a wake-type velocity profile at the entrance of the diffuser, which enhances diffuser performance.

Development tests with multijet nozzles showed that a five-jet nozzle, with the jet outlets located on a circle and spaced equally apart, produces the best performance. The circle on which the jet outlets are located is concentric with the jet pump throat, and its diameter has been optimized experimentally with respect to the throat diameter. The jets are designed to discharge parallel to the axis of the mixing throat as a result of experimental investigation of discharge directions.

An outline (Figures G.4.1-1 and G.4.1-2) of the second-generation jet pump used in BWR/5 and BWR/6 plants consists of: an inlet bend with a turning vane used to convey the drive flow from the inlet pipe to the drive nozzle; a five-jet drive nozzle; five streamlined struts for maintaining the nozzle and throat in alignment; a streamlined suction inlet to minimize suction flow inlet losses; a constant-area mixing throat in which the high-velocity drive stream mixes with the low-velocity suction stream; a 3° included-angle diverging throat where both mixing and pressure recovery take place; a 5° included-angle diffuser in which the kinetic energy of the mixed streams is converted to pressure energy; and the tailpipe, which further converts kinetic energy to pressure and leads to the high-pressure discharge flow from the jet pumps to the core inlet plenum.

The performance of a full-size prototype of this jet pump was thoroughly examined at reactor design and off-design conditions in a reactor downcomer annulus mockup. The performance obtained at design drive flow rate and subcooling gives a peak efficiency of 41.5%. Performance at other test conditions was essentially the same within the accuracy of the experiment, which is estimated at ±2% (two-sigma confidence limits), i.e., (41.5% ± 0.9)% efficiency.

The effect of varying drive flow rate from 50% to 100% of design flow on performance was studied. The pump was operated near the peak efficiency value with about 25 Btu/lb subcooling. The results indicate that no significant change in performance occurs.

Comparison shows that the second-generation jet pump performs with substantially higher efficiency than the first-generation jet pump.

G.4.1.1.2 Cavitation Tests

G.4.1.1.2.1 Effects of Cavitation

Visual examination of the jet pumps of three BWR/5's (Laguna Verde, ZPS-1, and LaSalle) and one BWR/6 (Perry) revealed no damage after being subjected to cavitation for 1 to 2 hours in the test loop.

G.4.1.1.2.2 Cavitation Tests of the Laguna Verde Production Jet Pumps

a. Introduction

Due to changes in operating and startup conditions for General Electric's BWR/5 and /6, a need arose to obtain rather extensive jet pump cavitation data.

It was decided to integrate the cavitation tests with the performance and diffuser calibration tests of the Laguna Verde (201-BWR/5) production jet pump performed in March 1976 at GE's facilities located in the Pacific Gas and Electric (PG&E) Power Plant at Moss Landing, California.

The cavitation tests were performed for a range of drive flow rates. Onset of cavitation was determined by monitoring changes in jet pump performance, vibration, and hydraulic noise level. The purpose of tests at reactor conditions was to verify that cavitation will not reduce jet pump performance or cause damaging vibration levels.

b. Summary and Conclusions

A cavitation coefficient (Kc), defined as the ratio of net jet pump suction head to jet pump suction velocity head, was used to correlate jet pump cavitation tests results. Cavitation inception was not characterized by a constant value of Kc. The cavitation coefficient depended on jet pump suction-to-drive-flow ratio and the type of cavitation.

Tests were performed at 540° F, which is the rated water temperature. For these conditions, cavitation did not raise the acoustic noise level, and jet pump acceleration levels actually decreased. Jet pump performance started dropping at Kc value from ~0.62 to 0.85, with the average being Kc = ~0.75. These Kc values are essentially the same as those obtained during cavitation tests at 540° F with the ZPS-1 and the La Salle production jet pumps. This illustrates, as expected, that all BWR/5 jet pumps have the same cavitation characteristics.

c. Discussion

The start of cavitation is defined through its effect on jet pump performance. It is well known that cavitation in the jet pump reduces jet pump efficiency. Therefore, the start of cavitation is defined as the point where jet pump efficiency starts dropping below its value measured far away from possible cavitation conditions (i.e., at high subcooling values).

It is realized that some cavitation may exist before any influence on performance starts showing up. However, "noise cavitation" is most likely limited to the shear layer between the drive and the suction streams. Since it does not extend to the walls, it does not affect performance and should not produce any material damage on the jet pump mixer walls.

The start of jet pump cavitation can usually be conveniently correlated using a jet pump cavitation parameter, Kc defined as

$$Kc = NPSH / [V^2/2g]$$

Cavitation or formation of vapor bubbles first occurs in the shear layer between the jet and suction flow. When these vapor bubbles collapse, acoustic noise is generated, resulting in "noise cavitation." If the pressure is reduced further, the vapor bubbles occupy more and more volume until the bubbles touch the jet pump wall. At this point, the suction flow will begin to be choked and the jet pump performance will decrease. Also, when the bubbles touch the wall, more intense acoustic noise is generated due to the solid boundary and vapor-fluid interface interacting. Therefore, the drop in performance coincides with a large increase in acoustic noise.

G.4.1.1.3 Production Tests

Hydraulic testing of production jet pump models for specific nuclear plants is performed at the Moss Landing Large Steam-Water Test Facility to establish efficiency and flow-measurement characteristics. Determination of efficiency performance is necessary, since the capability to produce rated core flow with minimized auxiliary power consumption is essential for achieving warranted power levels and heat rates. Flow measurements must be obtained because of the dependence of permissible core power levels upon direct measurement of core flow rate.

The method of determining core flow rate in a reactor is based upon measuring the differential pressure applied to all jet pump diffusers in the reactor. Air calibration of four of the jet pump diffusers is required because variations in performance result from complex but subtle manufacturing variations which develop between seemingly identical units. Hydraulic calibration of only one jet pump unit of a particular design is performed because related air-test data from the four calibrated diffusers is available. Such data establish the probable water calibrations within acceptable accuracy limits after one unit has undergone both air and water testing. These four calibrated diffusers are then used to calibrate the remaining diffusers after installation in the reactor.

G.4.1.2 Flow Control Valve Testing

A series of tests was run in 1970 at the GE Moss Landing Facility on an 8-inch scale model to establish the hydraulic performance of the valve. The valve flow characteristic C_v vs. position and cavitation index K_c vs. position were established.

Tests were run at the Byron Jackson Pump Facility in July and August 1974 on the hydraulic power and servocontroller system to establish and verify velocity, step response linearity, dead-band, interlock, etc. These tests were run on a 24-inch valve.

Tests were run at the Bingham-Willamette Pump facility from June to September 1974 in Portland, Oregon, in their recirculation loop on a 24-inch flow control valve, consisting of a "Floor Test" and "Hot Loop Test" to establish: valve feedback function delay time, step response time, flow as a function of position, and other parameters.

Flow tests were run on the present design of the flow control valve at the Alden Research Laboratory, Worster Polytechnic Institute, Holden, Massachusetts. These tests were run on a 20-inch valve in July 1975 and on a 24-inch valve in June 1975. The series of tests was directed and witnessed by ITT-Hammel-Dahl engineering personnel to determine the percentage of linear travel versus the valve coefficient C_v.

Tests were run at Bingham-Willamette again in the latter part of 1976 consisting of "Performance Tests" and "Operational Tests." The performance tests verified the function generator, flow vs. position, and also verified that the valve did not cavitate under operational conditions. The operational tests consisted of operating the valve for over 300 hours under full flow and temperature simulating daily load following.

G.4.1.3 Low Frequency Motor-Generator (LFMG) Set Testing

In 1976, a production LFMG set was tested in conjunction with a production recirculation pump and flow control valve. These were done with the recirculation pump in a pumping loop simulating the hydraulic characteristics of an actual plant recirculation loop and simulating the conditions (flow, pressure and temperature) under which the LFMG set will be required to power the recirculation pump. Tests were performed to determine the following:

a. recirculation pump normal coastdown time to 40%, 25%, and 15% of rated speed;

b. recirculation pump motor residual voltage decay time;

c. time for LFMG set to accelerate and be ready to accept load;

d. capability of LFMG set to accept the load when energizing the pump motor during pump coastdown at pump residual speeds of 40%, 25%, and 15% of rated speed; and

e. stability of the flow control valve system with the pump operating at 25% speed.

The results were quite satisfactory. There is a margin of approximately 3 to 1 between the time when the LFMG set must accept load and the time when the LFMG is ready to accept load. As expected, the optimum residual pump speed to perform the transfer was determined to be near 25% of rated speed. The LFMG set accepts the transfer with almost imperceptible change in MG set speed. The flow control valve is stable with 25% flow with the pump loop at normal pressure.

The production control circuit for the LFMG set was not included as part of the LFMG tests. However, pre-op tests on a production circuit have proven quite successful.

G.4.1.4 Hydraulic Power Unit and Actuator Testing

In 1974 the actuator, hydraulic power unit (HPU), and complete analog control circuit were tested to prove the design and operability of the equipment. An analog computer with simplified programs was used to simulate the master and flux feedback inputs to allow full automatic closed-loop operation of the system. Tests were done with production recirculation pump, flow control valve, HPU, actuator and controls. With the exception of velocity tests were conducted to simulated plant conditions. These tests determined the following:

a. dynamic performance for each closed loop control mode (position, flow, flux, and master),

b. stability,

c. valve position versus flow characteristics,

d. linearization of position versus flow characteristics,

e. valve deadband, and

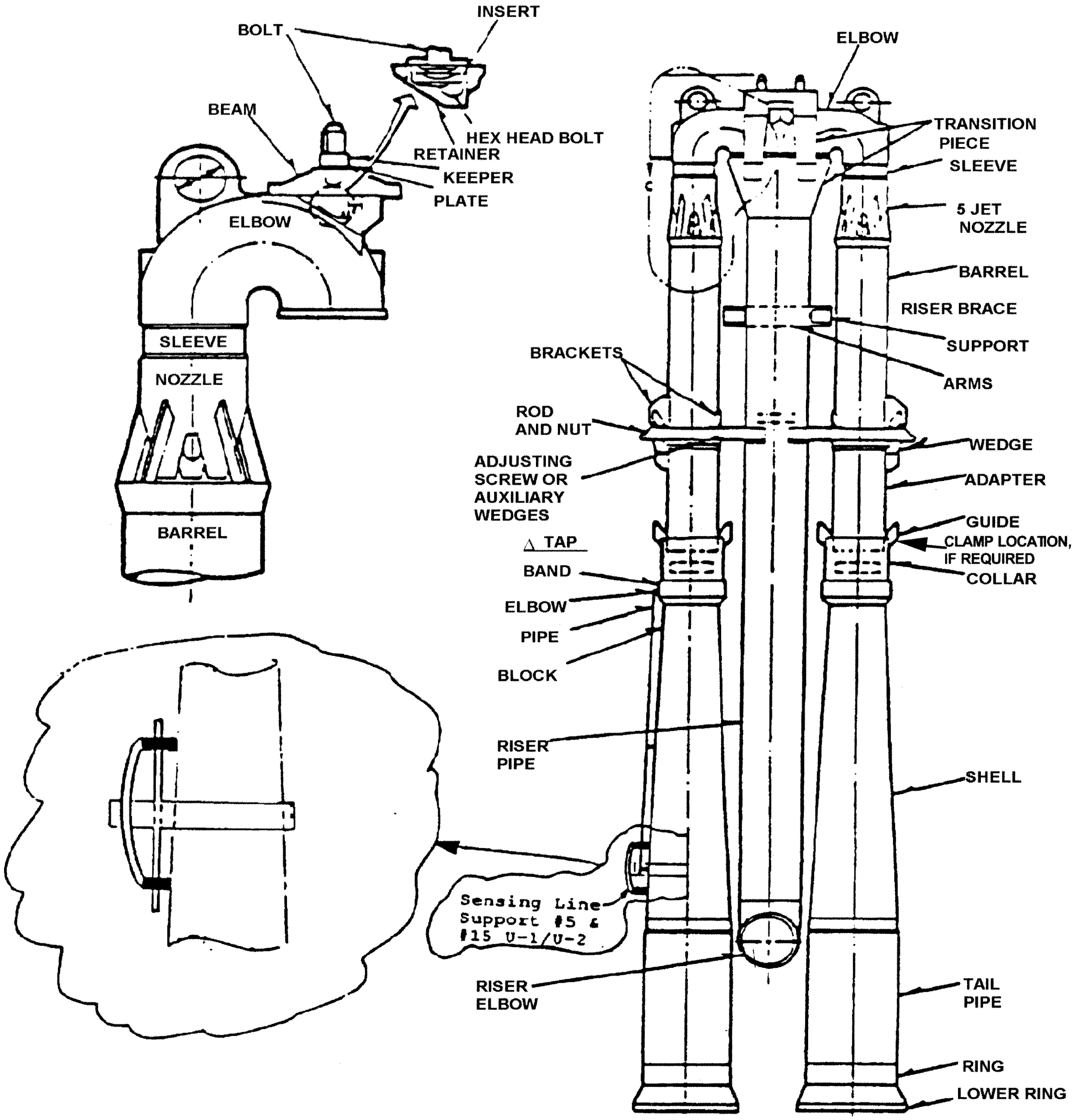
f. maximum actuator velocity under worst-case single failure.

Velocity tests were done with the hydraulic pump flow adjusted for maximum flow capability. The remainder of the system was adjusted for optimum stable operation. Velocity tests were also done with the actuator connected to the valve but without pressure in the valve and without flow through the valve. This was conservative, since actual velocities will be less than measured under conditions of flow and pressure in the valve.

Tests gave satisfactory results, and it was anticipated that performance would meet contractual requirements. Maximum average velocity measured was 21%/sec in the actuator retract (fastest) direction.

With the addition of the LFMG set, it was necessary to change the active stroke length of the flow control valve. In 1976 another series of tests was conducted. These tests were similar to those in 1974 except that closed-loop tests for the master, flux, and flow loops were not repeated, and shorter lines were used between the HPU and actuator. Length of the hydraulic lines was selected based on length of lines in an actual plant installation.

Similar results were obtained except that, as expected, dynamic performance was improved due to the shorter hydraulic lines. Also the actuator velocity was greater. Due to change in criteria, the actuator short-term peak velocity was measured as well as the average velocity. Short-term peak velocity was 27%/sec for approximately 5% of the active stroke, and average velocity was 22%/sec in the retract (fastest) direction.



LASALLE COUNTY STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE G.4.1-1 JET PUMP TERMINOLOGY

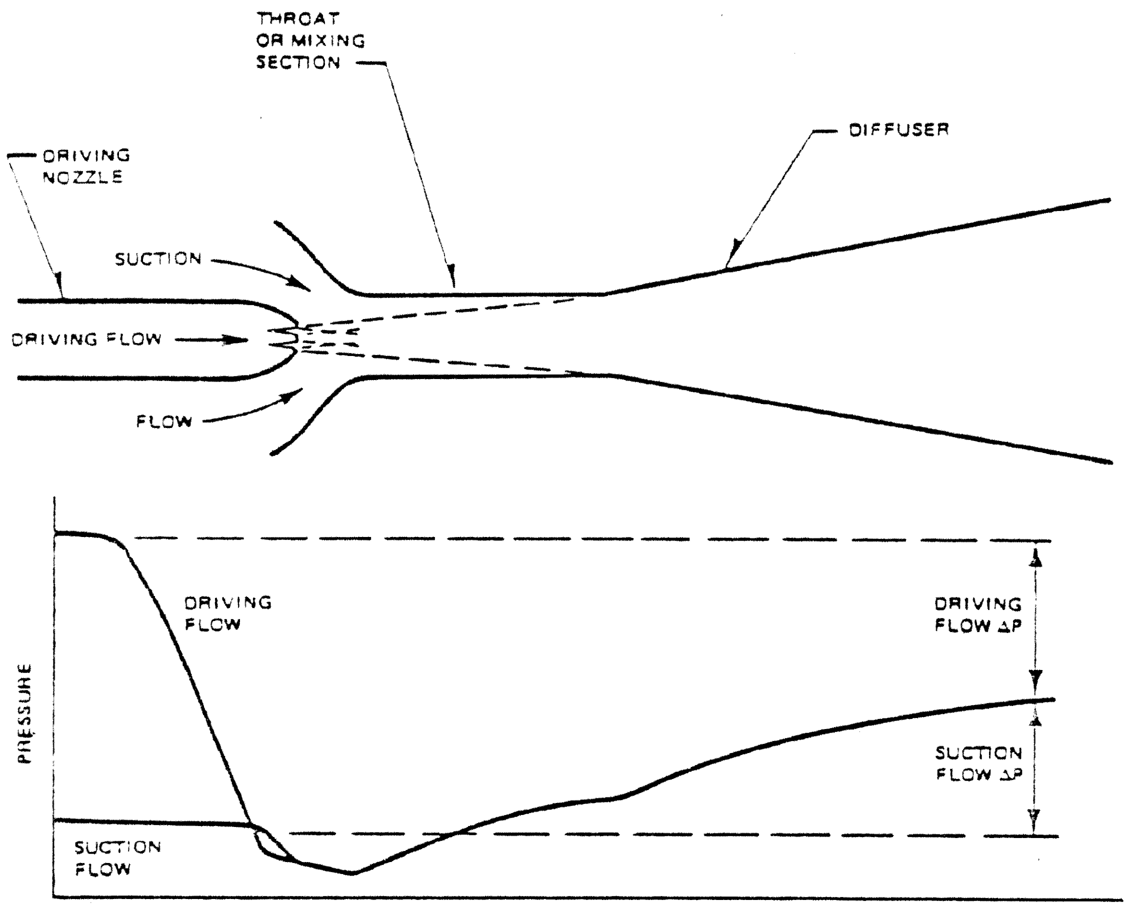


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FIGURE G.4.1-2

OPERATING PRINCIPLE OF JET PUMP

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G.4.2 Preoperational Tests

G.4.2.1 Recirculation System

Tests were performed on the recirculation system after construction and installation were complete and before fuel loading to confirm correct installation and that all process equipment was operational.

G.4.2.1.1 Test Conditions

The test conditions vary depending on the state of vessel internals assembly and the flow rate needed for the test. The vessel internals establish backpressure for the jet pumps and thereby the system operating conditions. Given the flow rate needed, the required water level, water temperature, and reactor pressure needed to prevent system cavitation were established.

G.4.2.1.2 Test Requirements

G.4.2.1.2.1 Functional Tests

Functional tests are conducted on individual components or assemblies of components such as interlocks, indicators, recorders, switches, alarms, annunciators, hydraulic power unit - actuator - flow control valve, etc., to verify that they function properly.

G.4.2.1.2.2 System Performance Tests

System performance tests are conducted in both one- and two-pump operating modes to verify that the performance is within expectations and within design criteria. Jet pump flows of at least 50% of rated are needed to verify that no construction debris is plugging the jet pump nozzles.

G.4.2.2 Vessel Internals Vibration Tests

Vibration tests are conducted to verify the structural integrity of core support structure and reactor internals in response to Regulatory Guide 1.20. The jet pumps are part of this program.

G.4.2.2.1 Vibration Test Conditions

Flow testing includes operation of the recirculation system in hot, pressurized conditions without fuel, at flows up to 100% of rated volumetric flow. The results of vibration measurements in prototype plants (Note: Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants, NEDE-24057-P, November 1977) show that this flow condition produces vibration responses in core support structures which are greater than those at normal operating conditions and are thus conservative for testing purposes. The test includes both balanced and one-loop recirculation system operation of sufficient duration to subject major components to a minimum of 10^6 cycles at their lowest dominant response frequencies and at maximum response amplitudes. Planned test times are 35 hours of two-pump operation at balanced flow, 14 hours of A-loop operation and 14 hours of B-loop operation at maximum flow condition without fuel in the reactor.

G.4.2.2.2 Inspection of Internals

The inspection of reactor internals was conducted following preoperational testing of the reactor system. The inspection was conducted prior to fuel loading and reactor criticality. This inspection has been performed previously on prototype BWR's following preoperational testing.

After completion of flow testing, the vessel head and the shroud head were removed and the vessel was drained. Access to the lower plenum is provided by opening a manhole in the shroud support plate. Reactor internal structures and components, including those in the lower plenum region, are given a close visual inspection to detect possible wear, cracking, loosening of bolts, and the presence of debris and loose parts. The inspection covered the following components:

a. Peripheral control rod drive and incore guide tubes, housing, and their lower joints.

b. Incore guide tube stabilizer connections and stabilizer bars. Plenum region for evidence of loose and/or failed parts.

c. Inside surfaces of the jet pump adapter to shroud support welds and jet pump diffuser to jet pump adapter welds. (Unit 1 only.)

d. Shroud support plate support columns. (Unit 2 only.)

e. Liquid control and delta pressure line and bracket welds.

f. The shroud-to-shroud weld.

g. Jet pump instrument lines and brackets.

h. The jet pump annulus for evidence of loose parts.

i. Jet pump beams, beam bolts, wedges, and locator screws.

j. Jet pump riser braces and welds.

k. Shroud and shroud head flange locating pins for evidence of deleterious motion marks other than those caused from normal installation.

l. Core support plate bolt keepers.

m. Steam separators and standpoints, shroud head bolt support ring brackets and supports.

n. Feedwater sparger and attachments.

o. Core spray lines, bracket, and core spray spargers.

p. Surveillance sample program specimen holder assemblies.

The La Salle preoperational test of reactor internals vibration conforms to Regulatory Guide 1.20 as a confirmatory test and inspection program. It pre-dated but fully meets the objectives of the NRC's approved generic program for BWR/4 and BWR/5 as accepted by the R. L. Tedesco letter of October 28, 1980 to G. G. Sherwood (which references NEDO-24057A).

Results of the preoperational test PT-PV-102 are included in the permanent station records.

G.4.3 Startup Tests

G.4.3.1 Recirculation System

G.4.3.1.1 General Description

Startup tests are those tests performed after fuel is loaded. The goals of the tests are as follows:

a. Verify the operational limits of the plant by establishing the power flow operating map.

b. Generate operating procedures that ensure rapid power ascension and maximize plant availability.

c. Obtain performance data needed to confirm the SAR analysis and conduct the surveillance required by the technical specifications and for detecting problems by establishing system performance bench marks where a deviation later in plant life indicates a potential problem.

d. Ensure that the flow control system performs as specified.

e. Verify that all interlocks work properly.

G.4.3.1.2 Specific Tests

The subsections which follow specify the purposes of the specific tests.

G.4.3.1.2.1 Cavitation Tests

Determine cavitation zones and establish operating procedures to preclude operation in these zones. Verify adequacy of the automatic cavitation interlocks.

G.4.3.1.2.2 Stratification Test

Determine the reactor operating modes where the stratification limits are likely to occur so that plant operating procedures can be established to avoid these modes.

G.4.3.1.2.3 Flow Increase Operation Test

Determine the operating procedures necessary to avoid a scram when transferring from 25 to 100% speed and when restarting an idle recirculation pump.

G.4.3.1.2.4 Steady-State Performance Test

Obtain operating data for future reference and for jet pump surveillance required by Technical Specifications.

G.4.3.1.2.5 Jet Pump and Recirculation Pump Flow Measurement

Calibrate the core flow measurement system and adjust the APRM flow bias instrumentation.

G.4.3.1.2.6 Recirculation Control System Test

Tune up the flow control subsystem to achieve the warranted response requirements. Measure plant response to one and two recirculation pump trips.

G.4.3.1.2.7 Recirculation Pump Trip Test

Record and verify acceptable performance of the RPT system.

G.5 RECIRCULATION PUMP TRIP AND FLOW CONTROL FEATURES

G.5.1 Recirculation Pump Trip (RPT) System

G.5.1.1 Identification

The RPT system includes power supplies, circuit breakers, trip circuitry, sensors and bypass switches to trip the recirculation pump motors from their main power supplies in response to a turbine/generator trip or load rejection event at greater than or equal to 25% power. Isolated outputs to the process computer and annunciators are provided for monitoring and are not a part of the RPT system.

The RPT is classified as Class IE, Seismic Category I, and Safety Class 3. The RPT system is designed to aid the reactor protection system (RPS) in protecting the integrity of the fuel barrier. Turbine stop valve closure or turbine control valve fast closure will initiate a scram and concurrent recirculation pump trip in order to keep the core within the thermal-hydraulic safety limits during abnormal operational transients.

G.5.1.2 Power Supplies

The RPT system utilizes two types of power from the same sources as the reactor protection system (RPS); 120-Vac from nonessential RPS motor-generators is supplied for the sensor channels and essential 125-Vdc from station batteries is supplied for the logic trip circuits for RPT.

G.5.1.3 Equipment Design

G.5.1.3.1 Circuit Description

The RPT system circuit is derived from the reactor protection trip sensor channels and combined into logic to perform the recirculation pump trip function (see Figure G.5.1-1). The fail-safe circuits are relays energized during normal operation. A failure of any one channel will not trip the recirculation pump motor circuit breakers or prevent tripping the breakers with the occurrence of an event requiring RPT.

Turbine control valve fast closure signals to the RPT system, from the reactor protection system, are derived from oil line pressure switches located on each of the four fast-acting control valve hydraulic mechanisms. When hydraulic oil line pressure is lost, the RPT system trips the recirculation pump motors from their power supplies whenever the reactor is operating at greater than or equal to 25% of rated core thermal power (see Figure G.5.1-2).

Turbine stop valve closure signals to the RPT system from the reactor protection system, are derived from valve stem position switches mounted on the four turbine stop valves. The switches open before the valve is more than 10% closed to provide the earliest positive indication of valve closure (see Figure G.5.1-3).

Turbine first-stage pressure signals to the RPT system, from the reactor protection system, are derived from four pressure switches. The pressure switches trip at a pressure setpoint corresponding to 25% of rated core thermal power to bypass the turbine stop valve and turbine control valve fast closure trips below this value (see Figure G.5.1-2).

Loss of one power supply will not cause nor prevent an RPT action.

The RPT, when initiated, completes the recirculation pumps circuit breakers trip. Restarting the pumps for normal power operation requires deliberate operator actions.

Channel and logic relays are fast-response, high-reliability relays from the reactor protection system. The relays are selected so that the continuous load will not exceed 50% of the continuous duty rating. The total response time from start of the turbine control valve fast closure signal or the turbine stop valve closure signal to complete suppression of the electric arc between the fully open contacts of the pump motor circuit breaker is less than 190 msec.

The RPT logic is illustrated in Figure G.5.1-1. The system is arranged as two separately powered trip systems. Each logic trip system has at least two channels of the monitored variable. Either of the two automatic trip systems will trip both of the two recirculation pump motors.

G.5.1.3.2 Bypasses and Interlocks

With the reactor power under 25% of rated core thermal power, the RPT is automatically bypassed by four pressure switches associated with the turbine first stage. Any one of the four channels in a bypass state initiates main control room annunciation. A manual keylocked bypass switch is provided for each of the two RPT systems, located on the relay panels, for logic testing. Placing the manual switch in bypass position initiates a control room annunciator. RPT is not inhibited, since the redundant system is operable.

The RPT system initiation logic is composed of contact from the RPS sensor channel relays. The interlock is performed using separate relay contacts so that no failure in the RPT system can prevent an RPS scram. Isolated contacts from the RPT system interlock with the recirculation pump low-frequency motor-generator set (LFMG) autostart circuit. A failure in the LFMG circuit does not affect the operation of the RPT system.

G.5.1.3.3 Redundancy

The RPT is divided into two divisions. Each division duplicates the function of the other to the extent that either system performs the pump motor power supply trip regardless of the state of operation or failure of the other trip system. The turbine stop valve closure and turbine fast valve closure signals are diverse inputs to the RPT for pump motor trip. Each pump motor is provided with two circuit breakers in series tripped from redundant division logic.

G.5.1.3.4 Testability

The RPT system logic can be tested during reactor operation without pumps trip and without inhibiting the pump-trip function by the redundant system.

A keylocked bypass switch for system circuit testing is provided for each system located at the RPS relay cabinets.

Because the RPT system logic consists of relay contacts actuated in the RPS, the RPS surveillance tests include the testing of the RPT logic system. See Section 7.2 of the UFSAR for RPS testing.

The recirculation pump motor power supply circuit breakers will be tripped by the RPT system during the reactor refueling period.

G.5.1.4 Environmental Considerations

The electrical equipment and devices of the RPT system are located in the control room and in the electrical equipment rooms. These areas have a controlled environment, isolated from accident environment, assuring reliable operation. The logic channel sensors located at the turbine-generator are discussed in Section 7.2 of the UFSAR.

G.5.1.5 Operational Considerations

The sensor channel logic relays are normally energized with contacts open in the RPT system logic. Loss of both RPS M-G buses or sensor variables out of tolerance will cause recirculation pump trip.

The RPT system has no manual trip feature. The pump motor power supply circuit breakers have their normal manual control switches at the control room panels for normal pump start and stop operation. These controls are isolated from the RPT circuits. Circuit breaker status indications for the operator consist of a green pilot light for breaker open position and a blue pilot light for breaker closed. In addition there is a red pilot light to monitor the RPT trip coil.

G.5.1.6 Analysis

Presented in the following are analyses to demonstrate RPT system conformance to functional requirements and specific regulatory guides.

G.5.1.6.1 Conformance to Functional IEEE Requirements

IEEE 279 1971 - Criteria for Protection Systems for Nuclear Power Generating Stations

a. General Functional Requirement (IEEE 279, Para 4.1). The function of the RPT system is to reduce the fuel thermal consequences during turbine or generator trip transients by tripping the recirculation pumps early in the event.

   The valve stem position of each of the four turbine stop valves is monitored by limit switches with setpoint at 10% or less valve motion from the open position. The trip channel signals to the RPT system anticipated imminent closure of the stop valves. The response time of the trip switch contacts is less than 10 msec after the valve has reached the setpoint. The logic arrangement is established to enhance frequent testing of these valves without causing an RPT for each valve test. The logic is one-out-of-two-twice stop valve closure signals from system A or B to produce RPT. The turbine control valve fast closure is monitored by pressure switches mounted at the turbine acceleration relay EH coil line pressure. The pressure switches provide RPT channel signals in less than 30 msec after decay of oil line pressure to the set point. The logic is two-out-of-two for each trip system A and system B. Testing the logic circuit is available without recirculation pump trip.

   The turbine stop valve and control valve fast-closure-trip bypass consists of sensing turbine first-stage pressure with two trips and four pressure switches so as to activate a trip bypass if the reactor is operating below 25% of rated core thermal power. This bypass is provided to permit continued reactor operation at lower power levels when the turbine valves are closed. This bypass is automatically removed as the reactor power and turbine first stage pressure increases above the setpoint value equivalent to 25% of rated core thermal power.

b. Single-Failure Criterion (IEEE 279, Para. 4.2). The RPT meets the single failure criterion. Sensors are electrically and physically separated, with conduit provided to the RPS cabinets. RPT signals to recirculation pump motor circuit breaker trip coils are separated into Division I and Division II and circuit breaker to pump motors are divisionally separated.

c. Quality and Components and Modules (IEEE 279, Para. 4.3). The quality of components and modules of the RPT system are of equal quality and high standards of the Class IE RPS as described in NEDO 10139, "Compliance of Protection Systems to Industry Criteria: GE BWR Nuclear Steam Supply System."

d. Equipment Qualification (IEEE 279, Para. 4.4). This paragraph is satisfied by qualification testing and certification of active RPT components. Complete records covering all essential components are maintained by GE.

e. Channel Integrity (IEEE 279, Para. 4.5). The RPT system logic consists of RPS channels as described in Section 7.2 of the UFSAR. Those components unique to the RPT system are located in the control room and electric equipment rooms environment.

f. Channel Independence (IEEE 279, Para. 4.6). The trip channels are physically separated and electrically isolated to meet this design requirement.

g. Control and Protection System Interaction (IEEE 279, Para. 4.7). The RPT system has no interface with control systems. Consequently, no failure or combination of failures in a control system will have any effect on the RPT system.

h. Derivation of System Inputs (IEEE 279, Para. 4.8). The RPT system is required to trip the recirculation pump motors on turbine/generator trip or load rejection. The sensing of turbine stop valve closure or turbine control valve fast closure are the actuated variables for the RPT function.

i. Capability of Sensor Checks (IEEE-279, Para. 4.9). The sensors are designed into the reactor protection system and are testable as described in Section 7.2 of the UFSAR.

j. Capability for Test and Calibration (IEEE 279, Para. 4.10). The turbine stop valves are tested individually by closing a stop valve and verifying RPT relay operation before the control room lights indicate the valve is closed. Calibration of the limit switches is possible only during shutdown and by physical observation. The turbine control valve closure pressure switches may be valved out, tested, and calibrated during periodic testing.

k. Channel Bypass or Removal from Operation (IEEE 279, Para. 4.11). The RPT system meets this design requirement as given above in Subsection G.5.1.3.2.

l. Operating Bypasses (IEEE 279, Para. 4.12). The recirculation pump motors are not required to trip below 25% of rated core thermal power. The four bypass channels comply with all requirements of IEEE 279 criteria.

m. Indication of Bypasses (IEEE 279, Para. 4.13). Automatic or manual test bypasses are annunciated in the control room.

n. Access to Means for Bypassing (IEEE 279, Para. 4.14). Automatic channels can be removed from operation, one at a time, for periodic testing under administrative control. Manual test bypass is accomplished with a keyed switch for periodic testing, under administrative control.

o. Multiple Setpoints (IEEE 279, Para. 4.15). Multiple setpoints are not a design requirement for RPT.

p. Completion of Protective Action Once it is Initiated (IEEE 279, Para. 4.16). The RPT system initiates to completion circuit breaker trips in maximum of five cycles when rated reactor power is above 40%. The trip is sealed in by switchgear action. Deliberate operator action is required to return the recirculation pumps to normal reactor operation.

q. Manual Actuation (IEEE 279, Para. 4.7). Manual initiation of RPT is not a design requirement since it is not possible to manually activate RPT in time to perform its safety function within 190 msec.

r. Access to Setpoint Adjustments, Calibration and Test Points (IEEE 279, Para. 4.10). Access to the turbine stop valve limit switches is not anticipated during reactor operation due to radiation and temperature environmental conditions. Turbine stop valve test controls are accessible. The turbine control valve fast closure pressure switches are available for periodic calibration.

s. Identification of Protective Actions (IEEE 279, Para. 4.19). RPT actions are identified to the channel level by control room annunciation.

t. <u>Information Readout (IEEE 279, Para. 4.20)</u>. Control room annunciation complies with this design requirement.

u. <u>System Repair (IEEE 279, Para. 4.21)</u>. The RPT system components are designed to facilitate maintenance, with the exception of the turbine stop valve limit switches. The redundancy of the eight switches permits plant operation with a defective switch until access can be gained to the switches for repair.

v. <u>Identification (IEEE 279, Para. 4.22)</u>. The RPT system logic relays are located in divisionally separated RPS cabinets with marker plate identification. The pump motor circuit breakers have distinctive markings and external cables are color coded. The RPT sensors are part of the reactor protection system and are discussed in Section 7.2 of the UFSAR.

IEEE 323-1971

The RPT system complies with this industry standard by type qualification testing and certification of all essential components. Complete records are maintained covering essential components.

IEEE 336-1971

The extent of conformance to this industry standard on the installation, inspection and testing requirements for instrumentation and electric equipment during construction is described in the Applicant's QA Program.

IEEE 338-1971

The RPT system is testable up to but not including actual recirculation pump trip circuit breaker during periodic testing of the sensor channels and logic systems. The pump trip circuit breaker testing is performed during a refueling outage.

IEEE 344-1971

This standard for seismic qualification is complied with for all RPT system Class 1E components.

G.5.1.6.2 Conformance to Regulatory Guides

Conformance of the RPT system to Regulatory Guide 1.22 is demonstrated under IEEE 338-1971.

Annunciation is provided in the control room to indicate a system or part of a system is inoperable or bypassed. As a result of design, preoperational testing, and startup testing, no erroneous bypass indication is anticipated. Therefore, Regulatory Guide 1.47 conformance exists.

Conformance to Regulatory Guide 1.53 for single failures is by compliance with IEEE 279-1971, Paragraph 4.2.

The requirements of the RPT system design do not include manual initiation, hence Regulatory Guide 1.62 is not applicable.

G.5.1.6.3 General Design Criteria Conformance 10 CFR 50 Appendix A

Criterion 13 - Each RPT system input is monitored and annunciated.

Criterion 19 - The RPT does not require controls or instrumentation in the control room.

Criterion 20 - The RPT system monitors the appropriate variables and initiates automatically when the variables exceed the design setpoints.

Criterion 21 - The system is designed with four independent and separated sensor channels and two independent and separated system divisions. No single failure or operator action can prevent RPT. The system can be tested during plant operation to verify its availability.

Criterion 22 - The redundant portions of RPT are separated such that no single failure or credible natural disaster can prevent RPT. Functional diversity is accomplished by sensing the turbine stop valve position for turbine trip and turbine control valve fast closure for load rejection.

Criterion 23 - The logic system is fail-safe. Loss of one of the two divisions of logic electrical power will not prevent the RPT system function because the logic will trip. The RPT system itself is energize-to-operate. Loss of one division of 125-Vdc control power will not prevent the RPT system function. Postulated adverse environments will not prevent an RPT.

Criterion 24 - The RPT has no control function. The system has interlocks to control systems through isolation devices.

Criterion 29 - No anticipated operational occurrences will prevent the RPT from accomplishing its safety function.

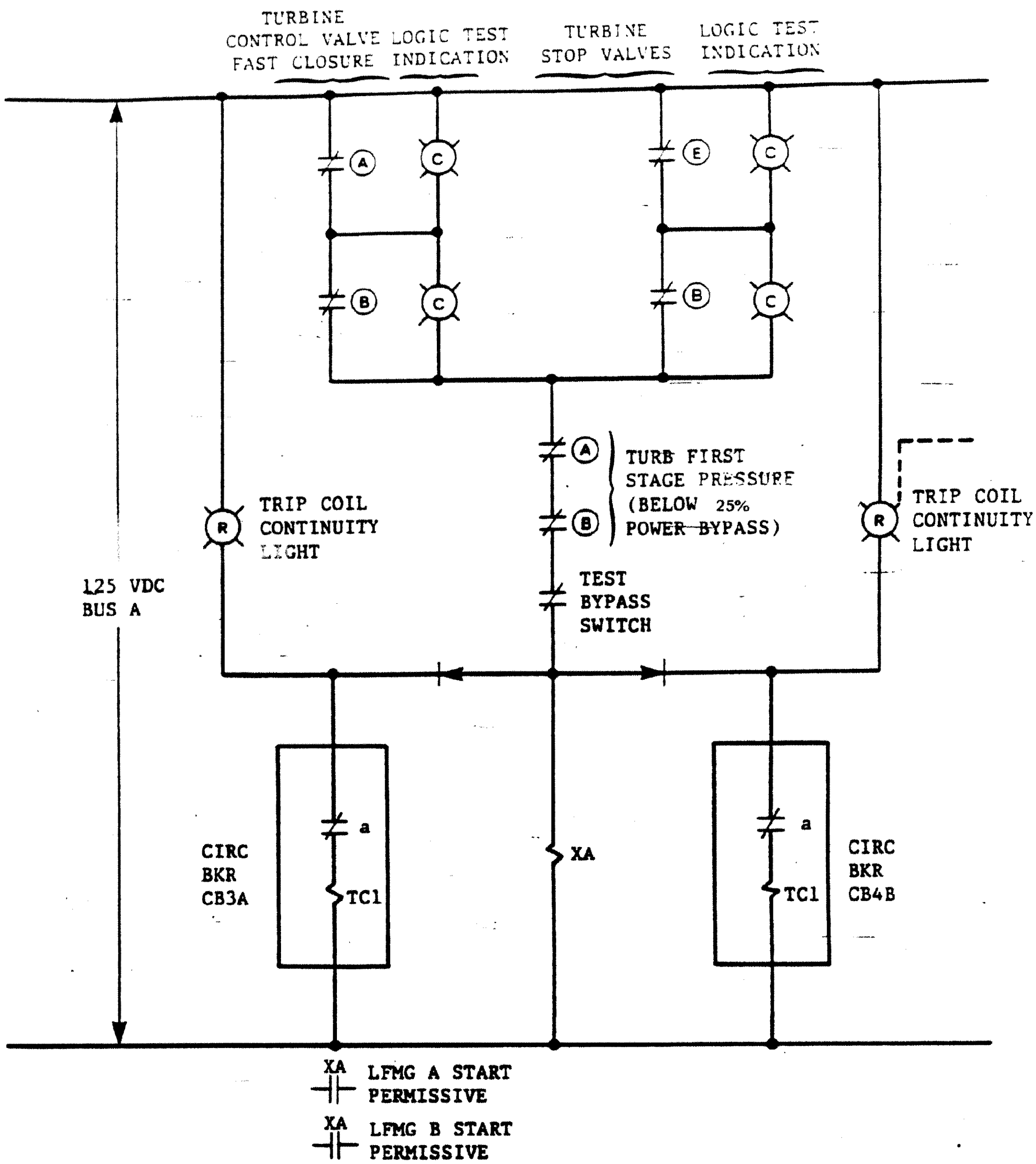

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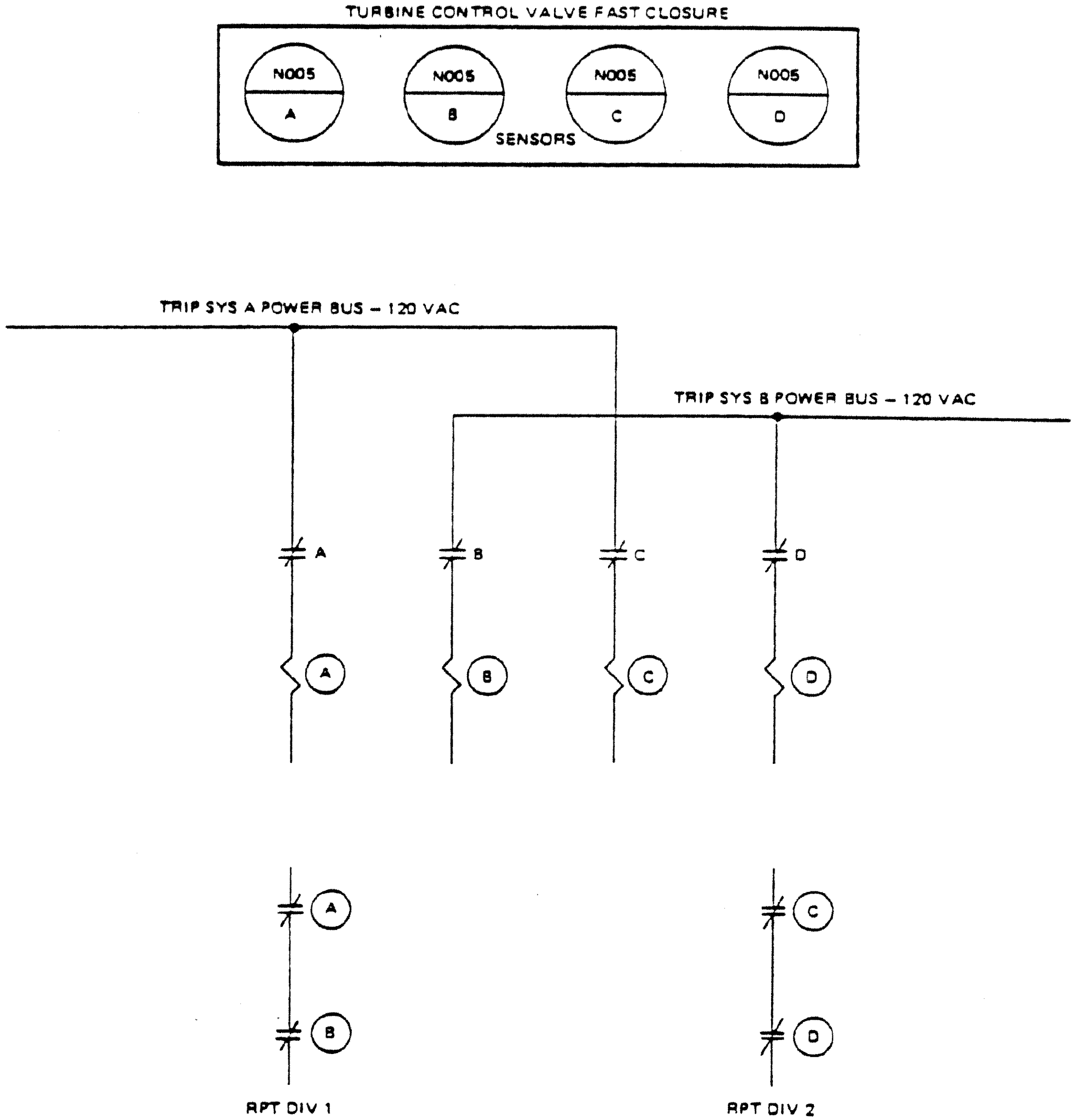
FIGURE G.5.1-1

RECIRCULATION PUMP TRIP SYSTEM A



TYPICAL FOR: TURBINE FIRST STAGE PRESSURE
SWITCHES N003A, B, C, D

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FIGURE G.5.1-2 TURBINE CONTROL VALVE FAST CLOSURE SENSORS

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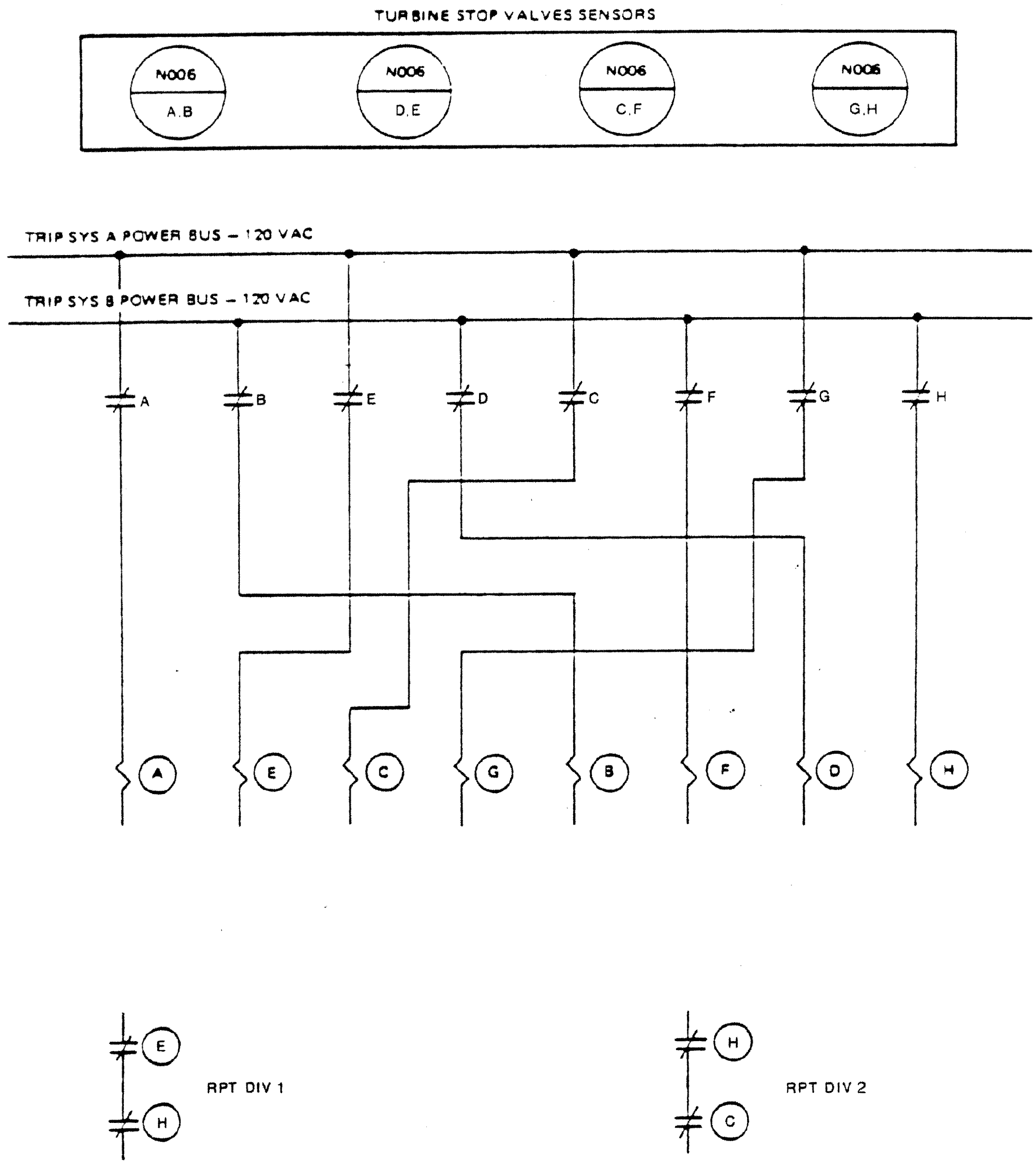

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FIGURE G.5.1-3

TURBINE STOP VALVE SENSORS

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G.5.2 Recirculation Flow Control

G.5.2.1 Identification

G.5.2.1.1 Function

The objective of the recirculation flow control function is to control reactor power level, over a limited range, by controlling the flow rate of the reactor recirculating water.

The recirculation flow control system consists of the electrical circuitry, switches, indicators, motor, and alarm devices provided for operational manipulation of the recirculation flow control valves, the LFMG set, and the surveillance of associated equipment. Recirculation flow control is either by individual manual operation or ganged manual if the power level is above 65% of rated, when the plant is operating on a rod pattern where rated power is produced with rated recirculation flow. During periods of low power level such as plant startup and shutdown, the recirculation pump and motor is powered by the LFMG set and is operated at approximately 25% rated full load speed. Refer to the cycle-specific COLR to determine if automatic flow limits are provided for a particular cycle.

G.5.2.1.2 Classification

This control function fulfills a power-generation criterion and is classified as nonessential for safety.

G.5.2.2 Power Sources

G.5.2.2.1 Control System

The recirculation flow control system instrument power is supplied by two nonessential 120-Vac instrument buses. Flow control loop A is powered from bus A and flow control loop B from bus B. That portion of the control system which is common to both loops A and B (master controller, etc.) is powered from either bus A or B. Each bus receives its normal power supply from the appropriate 480-Vac normal auxiliary power system.

Alternate

On loss of normal auxiliary power, one of the reserve auxiliary transformers provides backup power to the 480-Vac normal auxiliary power system.

G.5.2.2.2 <u>Pump Drive Motor and Controls</u>

The recirculation pump drive motor is an a-c 4-pole induction motor. The recirculation pump/motor operates from the normal plant electrical supply during normal plant power operation. At low-power levels, the recirculation pump/motor operates from the electrical output of the LFMG set. Since the LFMG set electrical output frequency is at approximately one-fourth the normal plant electrical frequency, the recirculation pump/motor is driven at approximately one-fourth of its rated speed.

The LFMG set consists of a 16-pole a-c induction motor driving a 4-pole a-c synchronous generator through a flexible coupling. This arrangement provides one-quarter normal plant frequency at the output of the generator.

The generator exciter is directly connected to the generator to provide a brushless excitation system. The voltage regulator for the excitation system is located in the auxiliary relay panel which is separate from the LFMG set.

G.5.2.3 <u>Equipment Design</u>

G.5.2.3.1 <u>General</u>

Reactor recirculation flow is varied by throttling the recirculation flow control valve. The recirculation pumps operate at constant speed, on either LFMG or normal 60-cycle power. By adjusting the position of the flow control valve, the change in recirculation flow automatically changes the reactor power level.

Control of core flow is such that, at various control rod patterns, different power level changes can be automatically accommodated. For a rod pattern where rated power accompanies 100% flow, power can be reduced to approximately 65% of full power by manual flow variation. At other rod patterns, manual power control is possible over a range of approximately 30% of the maximum operating power level for that rod pattern.

An increase in recirculation flow temporarily reduces the void content of the moderator by increasing the flow of coolant through the core. The additional neutron moderation increases the reactivity of the core, which causes reactor power level to increase. The increased steam generation rate increases the steam volume in the core with a consequent negative reactivity effect, and new steady-state power level is established. When recirculation flow is reduced, the power level is reduced in the reverse manner.

Each flow control valve has its individual manual control system and a ganged control mode which allows both valves to be manually controlled simultaneously.

In the ganged control mode, each loop has an individual flow controller that causes adjustment to valve position to meet a demanded change in valve position and, hence, core flow and core power. This process continues until both the errors existing at the inputs of the individual flow controllers are driven to zero.

The reactor power change resulting from change in recirculation flow causes the turbine EHC control system to reposition the turbine control valves.

G.5.2.3.2 LFMG Set

The LFMG set is not intended to be capable of starting the recirculation pump/motor with the pump/motor initially at zero speed. At low reactor power levels, the pump/motor start is initiated on the normal plant electrical power supply; as the pump/motor speed approaches rated full load speed, it is tripped automatically. When the coastdown pump/motor speed is about 25% of rated full load speed, the pump/motor is reenergized from the LFMG set and driven at about 25% rated full-load speed. Preceding initiation of the pump/motor, the plant operator may manually start the LFMG set. If the LFMG set is not operating when the pump/motor start is initiated, the LFMG set will be started automatically.

If pump/motor start is initiated at higher reactor power levels, the LFMG set will not start automatically, and the pump/motor will continue to operate at rated full load speed.

Certain trip functions, as defined in Table G.5.2-1, will trip the pump/motor and automatically transfer it to the LFMG set. Other trip functions will trip the pump/motor without transfer to the LFMG set.

G.5.2.3.3 Block Valves (Suction/Discharge)

The block valves are motor-operated gate valves for the recirculation pump and FCV isolation. The following interlocks are required for pump/motor startup and trip control function:

a. The power supply to the pump/motor will be tripped if either block valve is less than 90% open.

b. The pump/motor will not start unless the block valves are greater than 90% open.

G.5.2.3.4 Flow Control Valve Position Control Components

The flow control valves can be controlled individually or jointly. The control configuration is shown in Figure G.5.2-1. Each FCV has two independent controllers (AC70) and group S800 I/O modules supporting each redundant half of the hydraulic power unit (HPU) and associated process instrumentation. There is also an AC70 that controls the jet pump instrumentation flat panel display via a MODBUS connection.

Each loop flow controller M/A station communicates with both controllers corresponding to one recirculation loop. The control and logic of each HPU subloop is implemented in each individual controller without redundancy. The redundancy is achieved by the existence of two independent HPU subloops (servo valve, solenoid operated 4-way valve, hydraulic pump, etc.). In case of failure in one controller, an automatic transfer is performed to the backup controller and HPU. A failure in one of the HPU subloops will force a transfer to backup HPU subloop and controller. The controllers are balanced to allow bumpless transfer between them.

The ganged control algorithm is implemented in one of the controllers. In ganged control mode, both recirculation flow loops are controlled in parallel by a ganged position setpoint (common position setpoint M/A station).

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G.5.2.3.4.6 Manual/Automatic Transfer Stations

Setting the individual M/A station flow controllers to automatic provides "ganged" parallel manual operation of the flow control loops.

G.5.2.3.4.7 Flow Controller

The individual flow controller (one for each valve) transmits the signal that adjusts the valve position. Each flow-regulating valve can be manually positioned from individual M/A stations or in parallel from the ganged position setpoint station.

G.5.2.3.4.8 Limiter

A limiting function is provided for one feedwater pump trip flow runback. An electronic limiter with reasonable range adjustment is provided in each main flow control loop. This limiter is normally held bypassed by controller logic. When one feedwater pump trip coincides with reactor low water level alarms, the main regulating valve control signal is limited to close the valve to the desired position.

G.5.2.3.4.9 Hydraulic Power Unit

G.5.2.3.4.9.1 General Description

The hydraulic power unit is an electromechanical device that moves the actuator on the flow control valve in response to flow demand changes. Each flow control valve has a separate hydraulic power unit. Each unit has two subloops to provide redundant operation.

A failure in the "A subloop" would automatically place the "B subloop" into operation. The main components of the unit are shown on Figure G.5.2-2 (Unit 1 only).

The units are proven commercial devices rated for operation up to 3,000 psig. Actual operating pressure is 1800 psig. Acceptance of the units requires a hydrostatic pressure test at 3,000 psig.

Generally, mechanical failure in a subloop will lock up the actuator. If the redundant subloop starts and is operational, the valve will unlock in less than 1 second. Some failures that cause a transfer to the standby subloop, such as high fluid temperature, will not result in a lockup.

The hydraulic power units are located outside the drywell. Only the hydraulic lines, pilot-operated check valves and the hydraulic actuator are located inside the drywell.

G.5.2.3.4.9.2 Operation

A brief description of the operation of the hydraulic power units (HPU) is given to aid in the understanding of how the system functions. A diagram showing the main components of the HPU is given in Figure G.5.2-2.

Each subloop has a total inventory of about 75 gallons of Fyrquel EHC hydraulic fluid, 50 gallons of which is stored in a reservoir tank. A 15-hp, 1800-rpm motor is coupled to a variable displacement pump. The displacement is controlled by a pressure compensator which maintains desired hydraulic pressure. Maximum capacity of the pump is 9.7 gpm.

Actuator rod movement is accomplished by a 4-way servovalve that can supply the pump pressure to either side of the actuator piston. When the servovalve is "centered," flow to the actuator piston is blocked, which results in no movement. The servovalve movement is governed by the flow controller as described in Subsection G.5.2.3.4.7.

Downstream of the servovalve is a 4-way pilot-operated valve that isolates the standby HPU from the operating unit. This valve also serves to isolate (close) the actuator hydraulic lines for lockup purposes.

Mounted on the actuator, inside the drywell, are pilot-operated check valves. These valves can isolate the hydraulic cylinder in the event of a flow control lockup or hydraulic pressure loss. Supply pressure to both the pilot-operated check valve and the pilot-operated isolation valve is controlled by a solenoid-operated 4-way valve. This valve will cut off pressure to the pilot check valves and supply pressure to the isolation valve when actuated by a lockup signal from the flow controller. With this arrangement, actuator lockup is accomplished by both the pilot-operated check valves and the isolation valve.

A shuttle valve, located between the lockup valves of each subloop and the pilot-operated check valves, isolates the standby subloop from the operating unit.

On the discharge side of the pump is an accumulator that acts as a surge tank to improve the response time of the HPU. Inside the tank is a collapsible bladder that can be precharged with inert gas. By varying the initial precharge pressure, the surge fluid volume can be altered.

In addition to the pressure compensator on the pump, a pressure relief valve, set at 110% of operating pressure, is located near the pump discharge.

For clarity, several components were omitted from the HPU diagram. There are filters located in-line on the pump suction and discharge, on the servo return lines, and on the pump bypass drain line. Manual isolation valves are located between the pilot-operated check valves and the isolation valve, between the shuttle valve and the lockup valve, and on the pump suction. Since the reservoir tank is above the pump, the pump suction isolation valve permits maintenance on the HPU without draining the reservoir. To maintain hydraulic fluid at normal operating temperatures (below 145° F), an air-cooled heat exchanger is located on the reservoir return line.

G.5.2.3.4.9.3 <u>Protection Interlocks</u>

Various interlocks are provided to shutdown a malfunctioning subloop. The following conditions will result in an automatic transfer to the standby HPU:

a. low pump discharge pressure;

b. low reservoir tank level (near empty); and

c. high-temperature hydraulic fluid (150° F).

Other conditions which will not cause automatic transfer, but will be displayed and annunciated are as follows:

a. low reservoir level (different from Item b);

b. warm temperature hydraulic fluid;

c. discharge pump filter dirty;

d. pump bypass drain filter dirty; and

e. high actuator seal leakage.

G.5.2.3.4.9.4 Hydraulic Fluid Description

The hydraulic fluid used is Fyrquel EHC. It has an auto-ignition temperature of 1150° F. When heated to 455° F, the vapor will flash with an ignition source. If heated to 700° F, burning of the vapor lasts at least five seconds once ignited.

G.5.2.3.4.10 Hydraulic Line Design

Actual operating pressure is 1800 psig, although units are rated for service up to 3000 psig.

G.5.2.3.4.11 Actuator Design

The actuator is a double-acting cylinder rated for 3,000 psi severe operation, 5,000 psi non-shock. Actual operating pressure is 1800 psi. Position stops are located inside the cylinder to limit the travel of the piston. The stops have been set to provide the correct stroke travel for the flow control valve.

A linear velocity transformer is mounted on the actuator rod and provides the velocity feedback to the flow control system. The position feedback signal is provided by redundant transducers connected to the ball shaft of the valve.

Hydraulic sealing of the actuator rod is provided by a primary seal which provides the full pressure containment, and a secondary seal which prevents any leakage outside the unit. Between the seals is a cavity which has a drain line connected back to the supply reservoir. A flow switch mounted in this line will visually indicate and annunciate to the operator a high drain flow condition.

G.5.2.3.5 Inspection and Testing

The controller, valve position limiter, hydraulic power unit, and valve actuator are functioning during normal power operation. Any abnormal operation of these components can be detected during operation. The components that do not continually function during normal operation can be tested and inspected for calibration and operability during scheduled plant shutdown. All the recirculation flow control components are tested and inspected according to the component manufacturers' recommendations. The LFMG set can be operated, tested, and maintained during normal power operation.

G.5.2.4 Environmental Considerations

The recirculation flow control system is not required for safety purposes, nor required to operate during or after the design-basis accident. The system is required to operate in the normal plant environment for power-generation purposes only.

G.5.2.5 Operational Considerations

G.5.2.5.1 General Information

Controllers for positioning the flow control valve are located in the main control room. The individual loop controllers have provisions for either ganged or manual operation. Control switches for pump isolation valves, LFMG set and pump/motor, and interlock reset functions are also located in the main control room. Switches and indicators for control of the flow control valve hydraulic system are located on an operator station in the main control room for easy accessibility.

Except for the equipment protective interlocks, controls are manual, requiring operator action.

The LFMG set is required to supply power to the recirculation pump/motor only during plant low-power conditions. Provisions are made to allow operation of the LFMG set independent of pump/motor operation during normal plant power operations as well as during plant shutdown.

G.5.2.5.2 Reactor Operator Information

Indications, trends and alarms are provided to keep the operator informed of the status of system and equipment and to quickly determine location of malfunctioning equipment.

Visual display consists of loop flow, recirculation drive flow, valve position, and controller output and velocity feedback meters. Alarms are provided to alert the operator of malfunctioning control signals, inability of the valve position control loop, condition of hydraulic system, pump motor, and temperatures of cooling water. In most cases, alarms are supplemented by alarm indication provided by operator station indicators and messages to more closely define the problem area.

Indicating lights are provided to indicate the status of the LFMG set and the pump/motor control breakers. A pump/motor speed indicator is provided to indicate (in addition to the breaker indicating lights) to the operator which power supply is driving the pump/motor. Alarms are provided to alert the operator of automatic trips and transfers of the pump/motor, malfunctions, and availability of automatic control circuitry.

G.5.2.6 General Functional and Specific Regulatory Requirements Conformance

General Functional Requirements Conformance

The recirculation control system is designed so that no recirculation failure caused by the recirculation control system failure would result in violation of MCPR safety limit. The detailed description and analysis of the recirculation system failure events are in Section G.6.

The valve actuator has an inherent rate-limiting feature that will keep the resulting rate of change of core flow and power to within safety limit in the event of an upscale or downscale failure of the valve position or velocity controller.

Specific Regulatory Requirements Conformance

The interface signal and components that are used for both safety protection function and control function are designed in compliance with the following general design criteria (GDC) of Appendix A to 10 CFR 50, and IEEE 279, Section 4.7.

a. GDC 13 for Instrumentation and Control Requirements

   Conformance to this requirement is shown in the process instrument diagram figure. The recirculation system is capable of monitoring and controlling all the important processes and control variables over their anticipated range for normal operation and for anticipated operational occurrences.

b. GDC 24 for Separation of Protection and Control System

   The end-of-cycle recirculation pump trip (RPT) is the only safety-related control signal in the recirculation system. Failure of the recirculation control system or any component thereof would not affect the capacity of the protection system to perform its safety function.

c. IEEE-279, Section 4.7 for Control and Protection System Interface

   Any failure in the recirculation control system will not prevent the protection system from performing its intended protective function.

TABLE G. 5.2-1

REACTOR RECIRCULATION PUMP TRIP FUNCTIONS

ACTION					
EVENT	**A**	**B**	**C**	**D**	**E**
1	X		X		X
2	X		X		X
3		X		X	
4					X
5				X	
6	X			X	
7	X			X	
8	X		X		X
9	X			X	
10			X		

Events

1. Suction or discharge block valves less than 90% open.
2. Vessel high pressure (ATWS).
3. Turbine trip or generator load rejection.
4. 100% speed power supply trip of one of two operating recirculation pumps.
5. 100% speed power supply trip of all operating recirculation pumps.
6. Total feedwater flow less than approximately 20% nuclear boiler rated.
7. Temperature difference between the vessel water saturation temperature and recirculation pump suction temperature is less than the minimum permissible value.
8. Pump motor or LFMG set electrical protection logic is activitated.
9. Vessel low level (Level 3).
10. Vessel low-low level (Level 2).

Actions

A. Normal control trip of 100% speed power supply.
B. RPT engineered Safety Class 3 trip of 100% speed power supply.
C. Trip of 25% speed power supply.
D. Automatic start of LFMG set during coastdown following a 100% speed trip.
E. No automatic start of LFMG set during coastdown following a 100% speed trip.

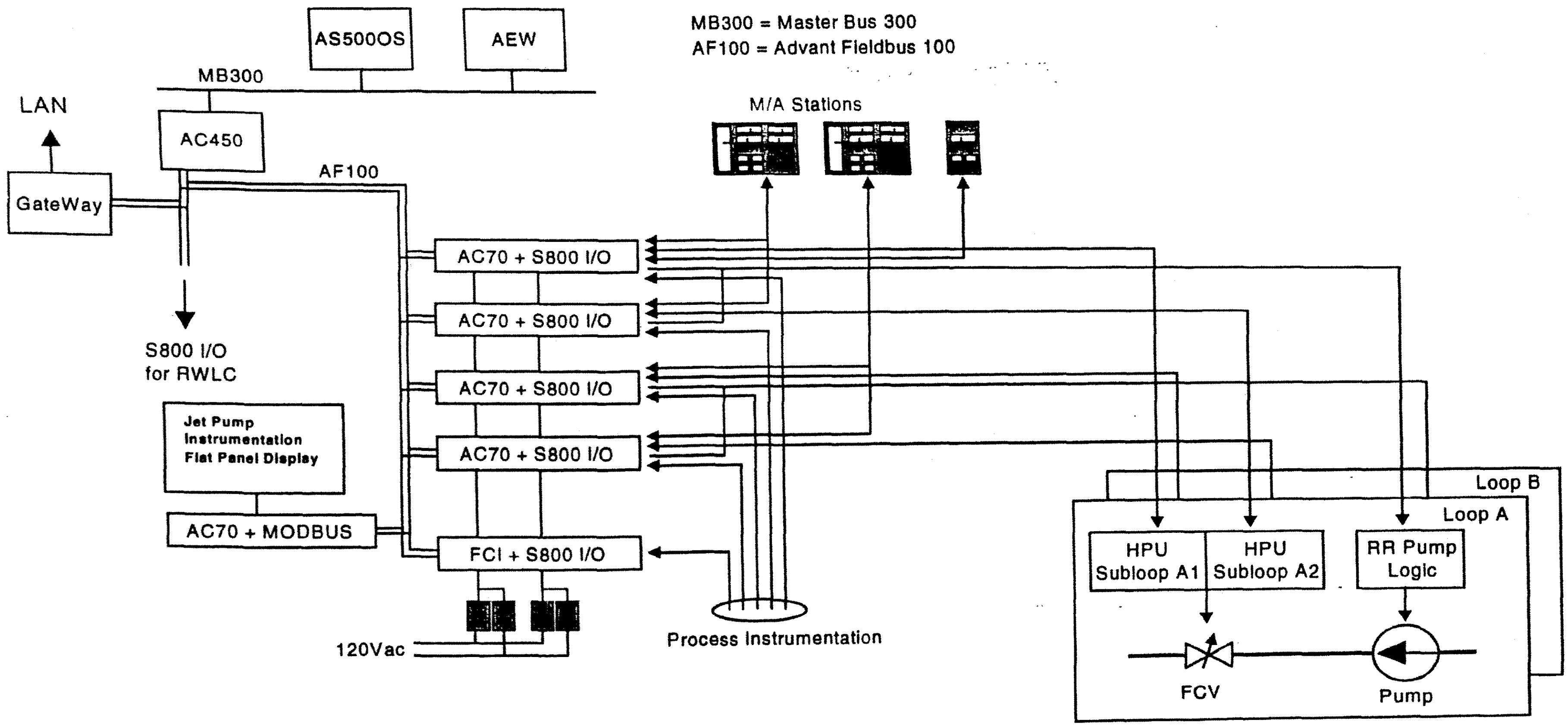

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FIGURE G.5.2-1

CONFIGURATION OF THE RRFC SYSTEM

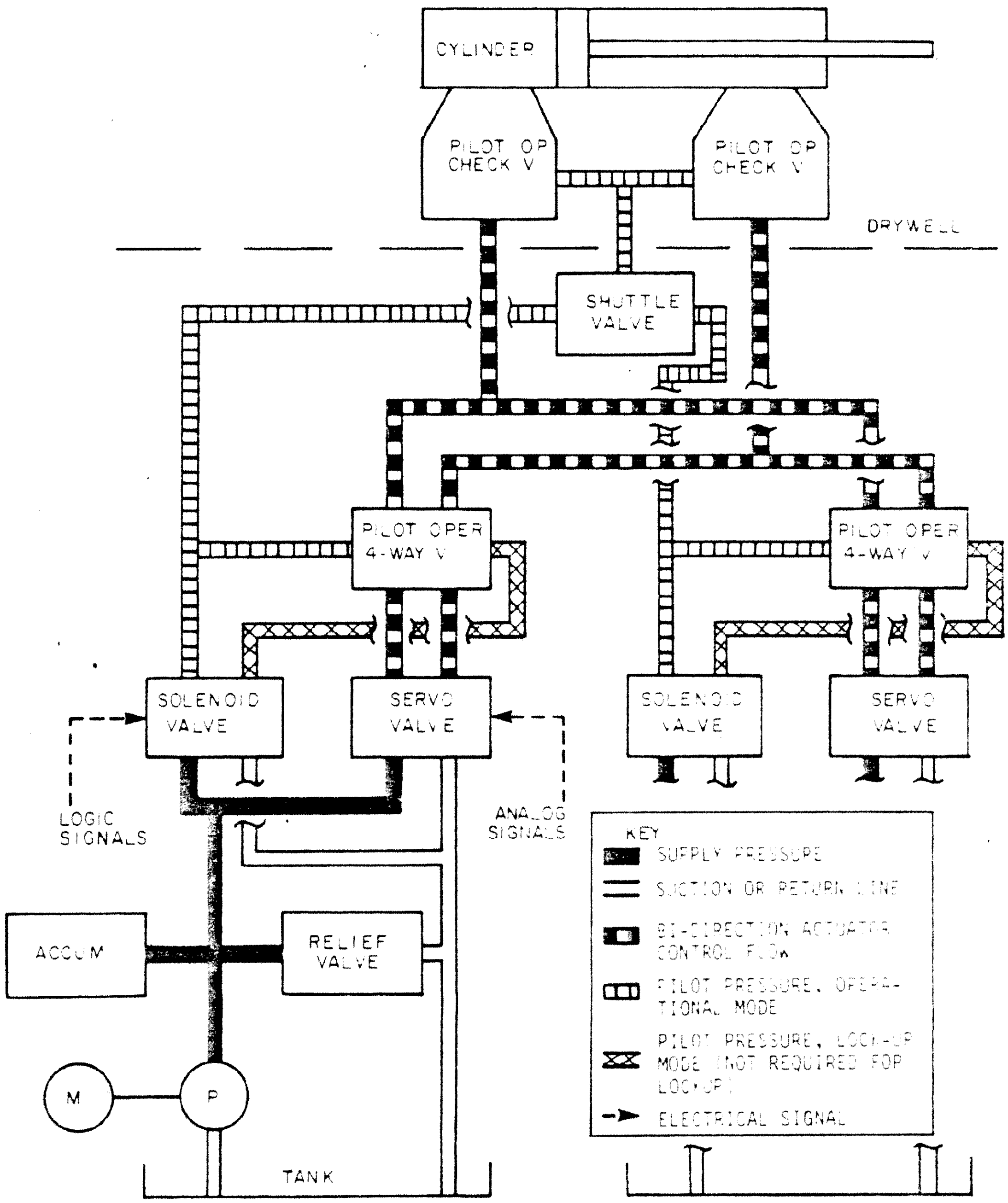

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FIGURE G.5.2-2

VALVE ACTUATOR HYDRAULIC POWER UNIT

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G.6 TRANSIENT ANALYSIS

Refer to Section 15.3.

G.7 REFERENCES

1. R. B. Linford, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor" (NEDO 10802), February 1973.

2. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50 Appendix K" (NEDO 20566). Flows - vessel water saturation temperature and recirculation pump inlet is less than the setpoint value. A High accuracy RTD is provided to measure recirculation pump inlet temperature. A pressure transmitter measuring vessel pressure in conjunction with a function generator is provided to measure vessel water saturation temperature.

3. GE Document, "SAFER/GESTR-LOCA, Loss-of-Coolant Accident Analysis, LaSalle County Station Units 1 & 2," NEDC-32258P, October 1993.

4. GE Document, "Evaluation of Lower Recirculation Flow Control Valve Cavitation Interlock Setpoint for LaSalle County Station Unit 1 and 2", GE-NE-AOO-05525-01, May 1994.

5. GE Document, "LaSalle 1 and 2 Jet Pump Cavitation Protection Logic," GE-NE-B3500504-01 DRF B35-00504, Class 3 June 1996.

6. “Exxon Nuclear Methodology For Boiling Water Reactors: EXEM BWR ECCS Evaluation Model,” XN-NF-80-19(P)(A) Volumes 2, 2A, 2B, 2C Exxon Nuclear Company, Inc., September 1982.

7. LaSalle LOCA-ECCS Analysis MAPLHGR Limits For Atrium-9B Fuel,” EMF-2175(P), Siemens Power Corporation, August 1999.

8. Westinghouse Report PNTC 00-261, LaSalle Unit 1 – Reactor Recirculation Flow Control, RRFC Functional Description (Proprietary)

9. Westinghouse Report PNTC 00-129, LaSalle Unit 2 – Reactor Recirculation Flow Control, RRFC Functional Description (Proprietary)

10. “LaSalle Units 1 and 2 LOCA-ECCS Analysis MAPLHGR Limit for ATRIUM-10 Fuel”, EMF-2641(P), Revision 0, Framatome ANP, November 2001.

ATTACHMENT G.A

REACTOR RECIRCULATION SYSTEM

DESIGN DRAWINGS

The following drawings are included by Reference Only in this attachment. Copies of drawings are located in the UFSAR Volumes entitled “Drawings”.

1. Piping and Instrumentation Drawings (P&ID)

   M-93, Unit 1
   M-139, Unit 2

2. Electrical Schematic Diagrams

   1E-1-4205AA-AZ, Unit 1
   1E-1-4205BA-BX, Unit 1
   1E-1-4205ZA-ZC, Unit 1
   1E-1-4205CA-CR, Unit 1

   1E-2-4205AA-AZ, Unit 2
   1E-2-4205BA-BZ, Unit 2
   1E-2-4205ZA-ZC, Unit 2
   1E-2-4205CA-CK, Unit 2

3. Control and Instrumentation Drawings

   M-2093, Unit 1
   M-2139, Unit 2

ATTACHMENT G.B

FAILURE MODES AND EFFECTS ANALYSES

ATTACHMENT G.B

FAILURE MODES AND EFFECTS ANALYSES

TABLE OF CONTENTS

ATTACHMENT G.B

FAILURE MODES AND EFFECTS ANALYSES

LIST OF TABLES

ATTACHMENT G.B

FAILURE MODES AND EFFECTS ANALYSIS

LIST OF FIGURES

G.B.1 Suction and Discharge Block Valves Failure Modes and Effects Analysis

TABLE G.B.1-1
(SHEET 1 OF 2)

SUCTION AND DISCHARGE BLOCK VALVES FAILURE MODES AND EFFECTS ANALYSIS

NO	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTOMS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
	Suction & Discharge Block Valve 24 Motor Operated Gate Valve. Provides Recirc Pump & Flow Control Valve Isolation During Maintenance	Leakage	Corrosion Thermal Shock High Stress Due to SSE Improper Installation Maintenance	Depending on the Leakage Rate, a High Drywell Pressure Trip May Occur, Causing Reactor Scram and Initiation of HPCS. Repair or Replace Seal Packing.	High Drywell Pressure Alarm Drywell Sump Alarms Cooler Condensate Alarms Drywell Rad Monitors Drywell Area Temperatures	Emergency Make-up Systems Activated	May Cause Reactor Scram	
		Rupture LOCA	Corrosion, Loads SSE, Weld Defect	High Drywell Pressure Will Cause Reactor Scram and Initiation of HPCS. Coincident With Above, Low Reactor Water Level (3) Will Cause Trip of Recirc Pumps To Low Speed. At Level (2) Pumps Will Trip Off. Repair Valve and Requalify.	Alarm On High Dry-well, Low RX Water Level	Emergency Make-up Systems Activated	Reactor Scram LOCA Event	
		Fail Close	Logic Error Motor Failure Improper ASM Maintenance Shaft Seizure	Unable to Restart Recirc Loop Loop or Plant Shutdown to Repair. No Effect on Safety as Plant Should be in Hot Standby or Cold Start Mode.	Close Position Switch On Valve		Loop Unavailable	
		Fail Open	Logic Error, Motor Failure, Improper ASM Maintenance, Shaft Seizure	Inability to Isolate Recirc Pump and FCV. Loop or Plant Shutdown to Repair Inability to use shutdown cooling through loop since cannot prevent backflow through the recirc pump.	Open Position Switch-On Valve Open Position Switch-On Valve Recirc Pump Flow and Head Indication Due to Backflow Low Jet Pump Flow Readings	None Shutdown Cooling through Other Loop or through LPCS Lineup	None None	 Both Valves have to fail in a Loop to Prevent RHR Operation.

TABLE G.B.1-1
(SHEET 2 OF 2)

NO	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTOMS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
	Suction & Discharge Block Valve (Cont)	Inadvertent Closure	Logic Error Operator Error	At Low Speed, Pump Will Trip At 90% Open Limit Switch On Valve.	Position Limit Switch On Valve		Loop or Plant Shutdown if Scram has Occurred.	Motor Operator Two Minutes to close
				At High Speed Pump Will Trip At 90% Open Limit Switch On Valve.	Position Limit Switch On Valve		Loop or Plant Shutdown if Scram has Occurred	

G.B.2 Flow Control Valve, Actuator and Hydraulic Power Unit Failure Modes and Effects Analysis

G.B.2.1 Flow Control Valve Failure Modes and Effects Analysis

G.B.2.2 Flow Control Valve Actuator and Hydraulic Power Unit Failure Modes and Effects Analysis (FMEA)

G.B.2.2.1 Purpose

The purpose of this FMEA is to present the consequences of malfunctions within the hydraulic equipment which controls the recirculation flow control valve in response to signals from the associated electronic equipment.

G.B.2.2.2 Discussion

G.B.2.2.2.1 General

The attached tabulation presents all conceivable single failures irrespective of their probability of occurrence. In addition, the "Remarks and Other Effects" column presents second, third, etc., failures which could increase the severity of the single failure. A diagram showing the main components of the hydraulic power unit (HPU) is given in Figure G.B.2-1. The Roman numerals used correspond to the equipment numbers in the FMEA.

G.B.2.2.2.2 "Name" and "Failure Mode" Columns

The "Name" column lists the various components by name and, where necessary for identification, the function. The "Failure Mode" column lists, for each entry in the name column, all failure modes which have unique effects. The exception to this is that failure modes which are common to all components (example: leakage) are listed only for the last entry, "ENTIRE CIRCUIT", in the "Name" column.

G.B.2.2.2.3 "Cause" Column

The most likely cause(s) for each failure is listed in this column. In addition to those causes listed, nearly all failure modes could be the result of high earthquake loads.

G.B.2.2.2.4 "Symptoms and Local Effects" Column

Effects in terms of variations in hydraulic parameters (temperature, pressure, flow, etc.), operation of other components, reliability, etc., are listed in this column. Unless specified otherwise, this column has been based on the assumption that the failure has occurred while the equipment is OPERATIONAL (i.e., controlling the flow control valve).

G.B.2.2.2.5 "Method of Detection" Column

The following symbols and definitions are applicable:

A = Condition is annunciated in the control room.

V = Condition may be detected by visual inspection of electrical control panels and/or hydraulic power units (Note: Inspection herein is defined as visual check of indicators, listening for excessive noise, checking valve handles for proper position, etc.).

U = Condition is undetectable by above methods.

G.B.2.2.2.6 "Inherent Compensating Provision" Column

Interlocks and other circuit features which reduce the severity of failure effects (local or system) are listed in this column. Interlocks which might not be tripped or which have no effect are not listed (for example, since a "jammed" actuator cannot move anyway, an interlock which shuts down the equipment when this occurs would not be listed).

G.B.2.2.2.7 "Effect Upon System" Column

This column presents the most critical effect on control of the flow control valve for each failure listed. The following symbols and phrases are used:

D = Degradation of transient performance (slow delay times, large overshoots, etc.).

L = Flow control valve position fixed due to initiation of hydraulic "lockup" circuits.

N = No effect on performance or "lockup" capability.

T = Flow control valve position fixed momentarily while operation is transferred to redundant circuits.

"Actuator will not respond....." = Flow control valve fixed in position even though "lockup" circuits are not engaged.

"Oscillation" = High frequency oscillation of flow control valve position.

"Excess Velocity" = Flow control valve position changes at rates exceeding normal limit of electronic controls (11%).

= Indicates sequence of effects. Examples:

T

L

Indicates that operation will be transferred to redundant circuits but a "lockup" will occur shortly after.

Only qualitative effects are listed herein. Quantitative results are presented in Section G.B.3.3.3.

G.B.2.2.2.8 "Remarks" Column

Numbers in column refer to list of remarks at end of tabulation.

Key to "Remarks" Column

1. If redundant circuits are disabled, effect will be "L".

2. If listed interlock(s) are disabled, flow control valve will not respond to changes in control signal and transfer will be initiated by servo error interlock.

3. If degradation of performance is severe enough, servo error interlock will initiate a transfer to redundant circuits.

4. If both relief valve and pump compensator have failed (double failure) pressure increases or oscillation may be sufficient to cause oscillation of flow control valve.

5. Excess temperature may also occur, in which case, temperature interlock would initiate transfer to redundant circuits.

6. If pressure interlock disabled, degradation of performance will be affected and Comment 3 is applicable.

7. This is an extremely unlikely type of failure due to the construction of this valve. (In order for contaminants to prevent opening, they must first pass through one or more small orifices which, if blocked, cause the valve to fail open.)

8. The only single failure which can totally disable the lockup circuit is a loss of pressure integrity between the actuator piston and the pilot-operated check valve. Since the flow control valve cannot drive itself, additional failures must occur to allow the hydraulic circuit to drive the actuator. One example of simultaneous failures which will do this follows:

a. Electrical controls fail to stop pump.

b. Solenoid valve fails in "A" position or electrical controls fail to de-energize solenoid A.

c. Servovalve fails off center or electronic controls provide erroneous control signal to servovalve.

Other combinations will accomplish the same thing but all require a minimum of three simultaneous failures, at least one of which must be in the electrical control circuits.

9. If servo-error interlock does not function, transfer will not occur.

10. This is an extremely unlikely mode of failure for a check valve.

11. Shuttle valve construction is such that it is virtually impossible for shuttle valve to block pilot line from both subloops simultaneously. (Note: Even if it did, pressure in pilot line would leak off through pilot-operated check valves to allow them to close).

12. If pressure gauge was damaged during initial calibration of equipment or during an overhaul, all components in the affected subloop could be set incorrectly. This would be detectable, however, when transient performance was checked by a pronounced difference in the performance of the two subloops. If, on the other hand, damage occurred during operation, the fact that <u>all</u> pressure devices (pressure switch, relief valve, and pump compensator) appeared to be out of adjustment in the same direction would provide indication of gauge damage.

13. Based on the design of the servovalve, this is the least likely mode of servovalve failure. This equipment utilizes a jet-pipe pilot stage which, if plugged by contamination, causes the valve to center. (Conventional flapper-nozzle pilot stages used on most equipment tend to fail "hard-over" as a result of contamination). Servo "hard-over" can result from a flow controller failure, e.g., short.

14. A servovalve failed "hard-over" is the <u>only</u> single failure which can cause a sustained, excess velocity. This failure is discussed in Section G.B.3.3.3. A number of multiple failure cases exist which may make the consequences more severe. (Without the servovalve failure to initiate and maintain the velocity, other combinations of malfunctions will not initiate or maintain the excess velocities). Other coincident failures which may increase the severity (i.e., result in higher velocities than would result if only the servovalve failed) are as follows:

a. compensator failure - high;

b. relief valve failure - high and compensator failure;

c. excess pump flow capacity;

d. accumulator precharge low; and

e. loss of pressure integrity.

15. Failure of the servovalve to center and simultaneous failure of the pilot-operated valve to center when operation is transferred to the alternate subloop will result in flow loss from the operational (alternate) subloop and a degradation of transient performance.

16. Leakage will usually be slow enough to allow air to bleed to tank through internal leakage before sufficient amounts accumulate to cause oscillation.

17. If temperature interlock does not function, pump and/or servovalve may be damaged. Also life of all seals and actuator may be reduced.

18. If low-level interlock does not function, pressure will be lost and pressure interlock will cause transfer or lockup. Some pump seal damage may occur, however.

TABLE G.B.2-1
(SHEET 1 OF 2)

FLOW CONTROL VAVLE FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
	Flow Control Valve 24 Ball Type Allows Power Changes by Varying Recirc Loop Flow	Leakage	Weld Defect Corrosion Seal Defect	No Loss of Flow Control. Need to Shut One Loop Down, Isolate with Block Valves and Repair.	High Drywell Pressure, High Level in Drywell Sump	Emergency Make-up Systems Activated.	Loop Un-available	
		Rupture (LOCA)	Earthquake SSE High Thermal Stress.	Bounded by LOCA Event. Plant Unavailable.	High Drywell Pressure, Low Reactor Water Level	Emergency Make-up Systems Activated.	Plant Shutdown	See Discussion in Section G.3.3.3
		Opening or Closure Rate >11%/Sec but Less than Effective 30% Sec (see Section G.3.3.3)	Logic Error Hydraulic Actuator Failure.	Reactor Protection System Activated. MCPR Above Safety Limit.	High Flux Transient. Reactor Scram.	Most Flow Control Single Failures Result in Valve Lockup.	Plant Un-available	
		Opening or Closure Rate >30%/ Sec Effective (see Section G.3.3.3)	Multiple Hydraulic Power Unit (HPU) Failures.	Accident Event. Multiple Failures Required in Hydraulic Power Unit.	High Flux Transient. Reactor Scram.		Plant Un-available	Multiple HPU Failures Can Result in Instantaneous Velocities Exceeding 60%/Sec.
		Lockup	Logic Error Hydaulic Power Unit Failure.	Valve Locked in Position. Repair Can Be Made While Plant is Operating.	Lockup Annunciated and Displayed.	If Fault Lies in Hydraulic Power Unit, Standby Unit, Will Start.	Power Change By Control Rod Movement and Flow Control Using Remaining Recirculation Loop.	Valve Will Automatically Lockup on Abnormal Control Signal and HPU Malfunction

TABLE G.B.2-1
(SHEET 2 OF 2)

FLOW CONTROL VAVLE FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
		Seizure	Wear Corrosion	Seizure Could Result in High Actuator Loads Leading to 1) Link Failure 2) Level Shaft Key Failure 3) Shaft-Ball Shrink Fit Failure	Loss of Flow Control. Displayed and Annunciated if Flow Controller Demands Movement.		After Detection, Required to Shut Loop HPU Down	Flow Control with One Loop Remaining

TABLE G.B.2-2
(SHEET 1 OF 16)

FLOW CONTROL VALVE ACTUATOR AND HYDRAULIC POWER

UNIT FAILURES MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
I	Electric Motor	Will Not Run	Power Loss, Electrical Fault, Mechanical Fault	Loss of Pressure	V, A	Overload/Undervolt and Pressure Interlocks	T	1, 2
II	Coupling	Decouples	Mechanical Fault	Loss of Pressure	V, A	Pressure Interlock	T	1, 2
III	Pump	A. Pressure Control Set High or Compensator Jammed at Maximum Position	Incorrectly Adjusted, Loss of Adjustment Contamination	High Pressure	V	Relief Valve Limits Pressure to 110% Normal	D	3, 4, 5, 14
		B. Pressure Control Set Low, But Greater Than Interlock Set Point	Incorrectly Adjusted, Loss of Adjustment	Low Pressure	V	None	D	3
		C. Pressure Control Set Less Than Interlock Set Point	Same as III.B	Low Pressure	V, A	Pressure Interlock	T	1, 6

TABLE G.B.2-2
(SHEET 2 OF 16)

FLOW CONTROL VALVE ACTUATOR AND HYDRAULIC POWER

UNIT FAILURES MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
		D. Pressure Compensator Sticking or Fully Jammed in Mid Position	Contamination and/or Wear	Pressure Oscillations, or High Pressure with FCV Stationary, Low with FCV in Motion	V	Relief Valve Limits Peak Pressures, Pressure Interlock Will Trip if Oscillations Are Severe Enough	D	3, 9, 14, 5
		E. Excess Flow Capacity	Incorrectly Adjusted or Mechanical Fault or Loss of Adjustment	None	U	Compensator Reduces Flow to Required Value (i.e., Value Demanded by Servo Valve)	N	14
		F. Inadequate Flow Capacity Sufficient to Maintain Pressure	Incorrectly Adjusted, Loss of Adjustment, Internal Leakage Due to Wear	None While Actuator is Stationary, More Extreme Pressure Decrease Transients When Actuator is in Motion	U	None	D	3
		G. No Flow	Mechanical Failure	Pressure Loss	V, A	Pressure Interlock	T	1, 2
IV	Pilot Operated Check Valve (Either)	A. Fails Closed	Mechanical Fault or Contamination	Fluid Trapped in Actuator	A	None	Actuator Will Not Respond to Changes in Position Demand Signal (One Direction Only) T L	

TABLE G.B.2-2
(SHEET 3 OF 16)

FLOW CONTROL VALVE ACTUATOR AND HYDRAULIC POWER

UNIT FAILURES MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
		B. Fails Open	Mechanical Fault, Contamination	None During Normal Operation. Fluid Not Trapped in Actuator During Lockup	U	Pilot Operated 4-Way Valves Trap Fluid During Lockup. Also, Pumps Stop During Lockup and FCV Cannot Drive Itself	N	8
V	Relief Valve	A. Failed Closed	Contamination	Increased Pressure Spikes During Transients, Reduced Reliability Due to Pressure Spikes	U	Steady State Pressure Maintained at Normal Value by Pump Compensator	N	4, 7, 14
		B. High Setting	Improperly Adjusted; Loss of Adjustment	Same as V.A., But Less Severe	U	Same as V. A.	N	4, 14
		C. Low Setting But Greater Than Interlock Setpoint	Incorrectly Adjusted; Loss of Adjustment	Low Pressure	V	None	D	3, 5
		D. Low Setting and Less Than Interlock Set-Point	Incorrectly Adjusted; Loss of Adjustment	Low Pressure	V, A	Pressure Interlock	T	1, 6, 5
		E. Failed Open	Contamination, Mechanical Fault	Loss of Pressure	V, A	Pressure Interlock	T	2, 1

TABLE G.B.2-2
(SHEET 4 OF 16)

FLOW CONTROL VALVE ACTUATOR AND HYDRAULIC POWER

UNIT FAILURES MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
VI	4-Way Valve, Sol. Op.	A. Fails Centered or at "B" Position	Loss of Control Signal, Incorrect Signal, Coil Failure, or Contamination (Start-up Only for Last Cause)	Pilot-Operated Check Valves and Pilot Operated 4-Way Valves Close, Locking Actuator	A	Servo Error Interlock	Actuator Does not Respond to Control Signals T	1, 9
		B. Fails at "A" Position	Incorrect Control Signals or Contamination	None While Subloop is Operational. Delays Speed of Subloop Isolation During Transfer to Alternate Subloop, Delays Actuator Lockup During Shutdown	U	During Transfer or Shutdown, Subloop's Pump is Stopped. Pilot Operated Check Valves and Pilot Operated 4-Way Valve Close Upon Pressure Loss Providing Lockup	N	8
VII	Pilot Operated 4-Way Valve	A. Fails Centered (Closed)	Contamination Loss of Adjustment, or Improper Adjustment	Servo Valve Isolated From Actuator	A	Servo Error Interlock	Actuator Does Not Respond to Control Signals T	
		B. Does Not Center (i.e., Remains in "X" Position or Travels Past Center to "Y" Position).	Contamination Loss of Adjustment, or Improper Adjustment.	No Effect when Subloop is Operational. Servo Valve Not Isolated from Interconnecting Piping when subloop is Not Operational.	U	Servo Valve is Centered (Closed) When Subloop is Not Operational and Thus Acts to Isolate Remainder of Subloop. Pilot Operated Check Valves Still Provide Actuator Lockup Function.	N	8, 15

TABLE G.B.2-2
(SHEET 5 OF 16)

FLOW CONTROL VALVE ACTUATOR AND HYDRAULIC POWER

UNIT FAILURES MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
VIII	Check Valve, Pressure Line	A. Fails Closed	Contamination	Loss of Pressure	V, A	Pressure Interlock	T	1, 2, 10
		B. Fails Open	Contamination	No Effect When Subloop is Operational. Allows Potentially Damaging Backflow from Accumulator to Pass Through Filter and Pump Whenever Pump Stops.	V (During Shut Down)	None	N	
IX	Servo Vale	A. Reduced Gain	Open Circuit in One Coil, Contamination	Decreased Flow Through Servo Valve for any Specific Control Signal	U	None	D	3
		B. Fails Centered	Contamination Loss of Cont. Signal, Short in Coil, or Internal Mechanical Fault.	Servo Valve Flow does not Respond to Control Signals.	A	Servo Error Interlock	Actuator Fails to Respond to Control Signals Actuator May “Drift” Slowly From Desired Position. T	9, 1
		C. Excessive Null Shift	Loss of Adjustment, Improperly Adjusted Internal Mechanical Fault.	Servo Valve Partially Open With No Signal, Responds More to Signals of One Polarity than the Other.	U	None	D	3, 8, 15
		D. Oscillates	Loose or Broken Internal Parts.	Pressure Oscillations, Noise, Vibration.	V	None	Oscillation	

TABLE G.B.2-2
(SHEET 6 OF 16)

FLOW CONTROL VALVE ACTUATOR AND HYDRAULIC POWER

UNIT FAILURES MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
		E. Fails Hard Over	Contamination Internal Fault, Incorrect Control Signal.	Full Pump and Accumulator Flow Directed to Actuator.	A	Servo Error Interlock. Also, sizing and Adjustment of Other Components Limits Velocity.	Excess Actuator Velocity T	9, 13, 14 8, 15
		F. Degradation of Flow Characteristics and or Excess Null Leakage.	Erosion Or Scoring of Spool and Sleeve.	Increased Non-Linearities in Flow Gain.	U	None	D	3
X	Shut-Off Valve (Manual), Servo Valve Isolation	A. Failed Open	Contaminants Stem Broken	No Effect During Operation, Subloop Should not be Maintained While Other Subloop is Operating.	U (While Operating)	None	N	
		B. Failed Closed	Inadvertently Operated.	Subloop Isolated From Portion of Field Piping	V	Servo Error Interlock	Actuator Will not Respond to Control Signals. T	9, 1
XI	Check Valve, Servo Valve, Drain	A. Failed Open	Contamination Broken Spring	No Effect During Operation. Fluid Loss from Reservoir During Servo Valve Replacement.	U	Line is Small and Easily Plugged for Maintenance.	N	
		B. Failed Closed	Contamination	Servo Valve will Fail Centered and Possibly Be Permanently Damaged. Possible Loss of Pressure Integrity in Low-Pressure Circuits	A	Servo Error Interlock	Actuator Fails to Respond to Control Signals. T	9, 10, 1

TABLE G.B.2-2
(SHEET 7 OF 16)

FLOW CONTROL VALVE ACTUATOR AND HYDRAULIC POWER

UNIT FAILURES MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
XI	Check Valve, Servo Valve Drain	A. Failed Open	Contamination Broken Spring.	No Effect During Operation. Fluid Loss from Reservoir During Servo Valve Replacement.	U	Line is Small and Easily Plugged for Maintenance.	N	
		B. Failed Closed	Contamination	Servo Valve will Fail Centered and Possibly Be Permanently Damaged. Possible Loss of Pressure Integrity in Low-Pressure Circuits	A	Servo Error Interlock	Actuator Fails to Respond to Control Signals. T	9, 10, 1
XII	Check Valve, Pump Drain Filter Bypass	A. Fails Open	Contamination Broken Spring.	Pump Drain Flow Bypass Fuller's Earth Filter. Reliability Decreased by Loss of Filtration.	U	Protection Against Particulate Contamination Provided by Other Filters. Introduction of Water and Chloride is Minimal if Good Operational Practices are followed.	N	
		B. Failed Closed	Contamination	Pump Drain Flow Cannot Bypass Fuller's Earth Filter. May Result in Excess Pressure Damaging Pump Shaft Seal and/or Fuller's Earth Filter.	U	High Pressure Across Element is Alarmed to Protect Element of Filter From Damage.	N	10
XIII	Check Valve, Heat Exchange Bypass	A. Failed Open	Contamination Broken Spring.	Flow Bypasses Heat Exchanger, May Result in Excess Temperature. Will also Result in Reservoir Fluid Loss During Maintenance of Return Line Components.	V, A	Temperature Interlock	T	1, 17
		B. Failed Closed	Contamination	Excessive Pressure at Inlet to Temperature Control Valve. Potentially Damaging to Temperature Control Valve, Return and Drain Filters, and Pump Shaft Seal. Possible Loss of Pressure Integrity in Low Pressure Circuits.	A	Will Result in Drain Filter Alarm, However; Damage Could Occur Too Rapidly for Alarm to be of Use.	N	10

TABLE G.B.2-2
(SHEET 8 OF 16)

FLOW CONTROL VALVE ACTUATOR AND HYDRAULIC POWER

UNIT FAILURES MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
XIV	Check Valve, Heat Exchanger Outlet	A. Failed Open	Contamination Broken Springs	No Effect During Operation, Loss of Fluid From Reservoir During Maintenance of Return Circuit Components.	U	None	N	
		B. Failed Closed	Contamination	Flow Bypasses Heat Exchanger. May Cause Excessive Temperature	V, A	Temperature Interlock	T	10, 1, 17
XV	Shuttle Valve	A. Fails to Shuttle (Can Only Occur on Startup or Transfer) Fully or Partially in One Direction Only.	Contamination, Mechanical Fault	Pressure from Operational Subloop Blocked or Partially Blocked from Pilot Line Vented to Tank Via Alternate Subloop Resulting in Closure of Pilot Operated Check Valves.	A	Servo Error Interlock	Actuator Fails to Respond to Control Signals T	9, 11, 1
		B. Fails to Shuttle Fully in Either Direction (Can Only Occur on Startup or Transfer).	Contamination, Mechanical Fault	Same as XI.A.	A	Does Not Prevent Lockup Since Pilot Line is Vented.	Actuator Fails to Respond to Control Signals. T L	11

TABLE G.B.2-2
(SHEET 9 OF 16)

FLOW CONTROL VALVE ACTUATOR AND HYDRAULIC POWER

UNIT FAILURES MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
XVI	Needle Valve Used as Bypass Valve	A. Failed Closed	Handle Broken Off.	No Effect During Operation. Maintainability Degraded.	U During Operation	None	N	
		B. Failed Open	Contamination, Improperly Adjusted, Loss of Adjustment	Pump Flow Bypassed to Return	V	None	D	3
XVIa	Needle Valve Used as Gauge Shut Off Valve	A. Open Too Far.	Loss of Adjustment, Improperly Adjusted.	Excessive Shock and Vibration at Pressure Gauge. Reliability of Gauge Degraded.	V	None	N	
		B. Closed	Loss of Adjustment, Improperly Adjusted	Pressure Gauge Isolated from Circuit	V	None	N	
XVIb	Needle Valve Used as Pressure Switch Shutoff Valve.	A. Failed Open.	Contamination	No Effect During Operation. Maintenance of Pressure Switch Less Convenient.	V	None	N	
		B. Failed Closed Before Subloop Pressurized	Improperly Adjusted	Pressure Interlock will be Tripped.	A	None	T	1
		C. Failed Closed After Subloop Pressurized	Loss of Adjustment	Pressure Interlock Disabled.	V	None	N	
XVII	Ball Valve Used as Suction Line Shutoff Valve	A. Failed Closed	Inadvertent Operation. Improper Setting.	Pump Suction Isolated From Reservoir. Loss of Pressure. Possible Pump Damage.	A, V	Pressure Interlock	T	1, 2
		B. Failed Open	Contamination, Stem Broken	No Effect During Operation. Reservoir Must be Drained to Service Pump.	V	None	N	

TABLE G.B.2-2
(SHEET 10 OF 16)

FLOW CONTROL VALVE ACTUATOR AND HYDRAULIC POWER

UNIT FAILURES MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
XVIIa	Ball Valve Used as Drain Shutoff Valve	A. Failed Closed	Inadvertent Operation. Improper Setting	Subloop Reservoirs Isolated from Each Other and All Actuator Drain Flow Directed to other Reservoir. Eventual Low Fluid Level in Reservoir.	V, A	Low Level Interlock. Also, Reservoirs Sized such that Interlock will be Tripped Prior to Over Filling of Alternate Reservoir.	T	1, 18
XVIII	Temperature Control Valve	A. Fails Open	Contamination, Broken Spring	Return Fluid Always Flows Through Heat Exchanger. Low Temperature	V	None	D	3
		B. Fails Closed	Leakage Vapor in Capillary, Pinched Capillary.	Flow Blocked from Heat Exchanger. High Temperature	V, A	Temperature Interlock	T	1, 17
XIX	Heat Exchanger	A. Does Not Cool Fluid.	Fouling Air or Fluid Flow Blocked.	High Temperature	V, A	Temperature Interlock	T	1, 17
XX	Accumulator	A. Pre-Charge High	Improper Adjustment	Less Fluid Stored in Accumulator. More Severe Pressure Spikes During Transients, However, Negligible Effect on Life or Reliability.	U	None	D	3, 14
		B. Pre-Charge Low	Improper Adjustment or Leakage	More Fluid Stored in Accumulator.	U	None	D	3
		C. Leakage of Pre-Charge to System Fluid (Hydraulic).	Faulty Bladder	Oil Column Stiffness Reduced, Pressure Oscillations, Noise	V	Equipment Design is such that Small Amounts of Air Will be Bled Back to Tank Before They Can Initiate Oscillation..	Oscillation	16

TABLE G.B.2-2
(SHEET 11 OF 16)

FLOW CONTROL VALVE ACTUATOR AND HYDRAULIC POWER

UNIT FAILURES MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
XXI	Filter, Pressure Line	Bypassing Fluid	Faulty Bypass Valve, Torn Element, Dirty Element.	Reduced Reliability Due to Lack of Filtration. (Primarily Effects Servo Valve.)	A (If Due to Dirty Element or Faulty Bypass) None (If Due to Torn Element)	Return and Fuller's Earth Filters Provide Some Protection (Sump Strainer Also).	N	
XXII	Strainer	A. Blocked	Dirty	Loss of Pump Output Pressure or Erratic Output Pressure If Blockage is Extreme. Otherwise, Excessive Noise from Pump.	V, A, If Extreme	Pressure Interlock (If extreme)	T	1, 6
		B. By-passing Fluid	Torn	Reduce Protection for Pump, Thus, Reduced Pump Reliability.	None	Primary Control of Contamination Provided by 10µ Filters.	N	
XXIII	Filter, Fuller's Earth, Pump Drain Line	Blocked (Causes Fluid to Bypass Via Check Valve)	Dirty Element, Contamination in Orifice.	Reduced Reliability Due to Loss of Protection Against Chlorides, Water, Etc.	A, V if Element is Dirty, N One if Orifice is Blocked	Protection Against Particulate Contamination Provided by Other Filters. Introduction of Water and Chlorides is Minimal if Good Operational Practices Followed.	N	
XXIV	Pressure Switch, Pressure Line	A. Fails Open	Electrical Fault, Broken, Spring, Jammed Actuator, Loss of Adjustment	Faulty Indication of Pressure Loss and Initiation of Interlock Action.	V, A	None	T	1
		B. Fails Closed	Electrical Fault, Jammed Actuator, Loss of Adjustment.	Disables Pressure Interlock and Alarm.	U, V (When Subloop is Not Operational).	None	N	

TABLE G.B.2-2
(SHEET 12 OF 16)

FLOW CONTROL VALVE ACTUATOR AND HYDRAULIC POWER

UNIT FAILURES MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
XXV	Pressure Switch, Drain Filter	A. Fails Open	Electrical or Mechanical Fault.	"Drain Filter Dirty" Alarm Disabled Thus Reduced Reliability.	V (When Filter is Dirty)	Bypass Check Valve Protects Filter From Damage.	N	
		B. Fails Closed	Electrical or Mechanical Fault.	Trips "Filter Dirty" Alarm.	V	None	N	
XXVI	Temperature Switch	A. Fails Open	Electrical or Mechanical Fault.	Disables Temperature Alarm and/or Interlock.	V if Temperature is High Otherwise, U.	None	N	17
		B. Fails Closed	Electrical or Mechanical Fault.	Trips Temperature Alarm and/or Interlock	V	None	T	1
XXVII	Level Switch	A. Fails Open	Electrical or Mechanical Fault.	Disables Fluid Low Level Alarm and/or Interlock	V if Level is Low, Otherwise, U.	None	N	18
		B. Fails Closed	Electrical Fault or Hole in Float.	Trips Low Level Interlock and/or alarm	V, A	None	T	1
XXVIII	Flow Switch	A. Fails Open	Electrical or Mechanical Fault.	Actuator Drain Flow Alarm Disabled.	U	None	N	
		B. Fails Closed	Electrical or Mechanical Fault.	Actuator Drain Flow Alarm Tripped.	U	None	N	
XXIX	Sight Glass	A. Glass Broken	Someone Hit It.	Fluid Lost from Reservoir	V, A	Sight Glass Located High Enough to Limit Fluid Loss	N	
XXIXa	Thermometer	B. Incorrect Reading		Incorrect Visual Reading, No Effect Since Device is Not Used to Adjust Components	V if Error Significant	None	N	

TABLE G.B.2-2
(SHEET 13 OF 16)

FLOW CONTROL VALVE ACTUATOR AND HYDRAULIC POWER

UNIT FAILURES MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
XXX	Filler	A. Leaks	Loose	Increased Chance for Contamination	U	Filters Protect Against Contamination.	N	
XXXI	Pressure Gauge	Incorrect Reading	Zero Shift, Failed Movement, Deformed Bourden Tube.	No Effect Unless Components Are Readjusted Without Checking Gauge Calibration.	See Remarks	See Remarks	N	12
XXXII	4-Way Valve, Manual	A. Failed in Center or Unused Position	Inadvertent Operation of Valve.	Vents Pilot Line Causing Pilot Operated Check Valves to Close.	V, A	Servo Error Interlock	Actuator Will Not Respond to Control Signal. T	1, 9
		B. Failed in Operational Position	Contamination Broken Stem.	No Effect During Operation. Equipment Should not be Maintained While Alternate Subloops Operational.	V	None	N	
XXXIII	Filter	A. Bypassing Dirty Fluid	Faulty Bypass, Torn or Dirty Element.	Reduced Reliability Due to Lack of Filtration.	V	Other Filters Provide Sufficient Protection, Thus, Reliability is minimal.	N	
XXXIV	Filter Reservoir	Saturated for Damaged Element.	Poor Maintenance	Allows water Vapor and/or Particulate Contamination In Reservoir, Reducing Reliability.	U	System Filters Will Protect Critical Components.	N	

TABLE G.B.2-2
(SHEET 14 OF 16)

FLOW CONTROL VALVE ACTUATOR AND HYDRAULIC POWER

UNIT FAILURES MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
XXXV	Hydraulic Cylinder	A. Excess Friction or "Stiction".	Worn Bearings, Contamination	Greater Differential Pressure Required to Drive Actuator.	None	None	D	3
		B. Actuator Totally Jammed	Worn Bearings Seized	Actuator Will Not Move	A	None	Actuator Will Not Respond to Changes in Control Signals. T L	
XXXVI	Entire Circuit	A. Internal Leakage Between High- and Low-Pressure Circuits or Internal Leakage Between Actuator Extend and Retract Lines.	1. Worn Valve, Worn Seal, Etc.	Decrease in Flow Available to Drive Actuator.	U	None	D	3

TABLE G.B.2-2
(SHEET 15 OF 16)

FLOW CONTROL VALVE ACTUATOR AND HYDRAULIC POWER

UNIT FAILURES MODES AND EFFECTS ANALYSIS

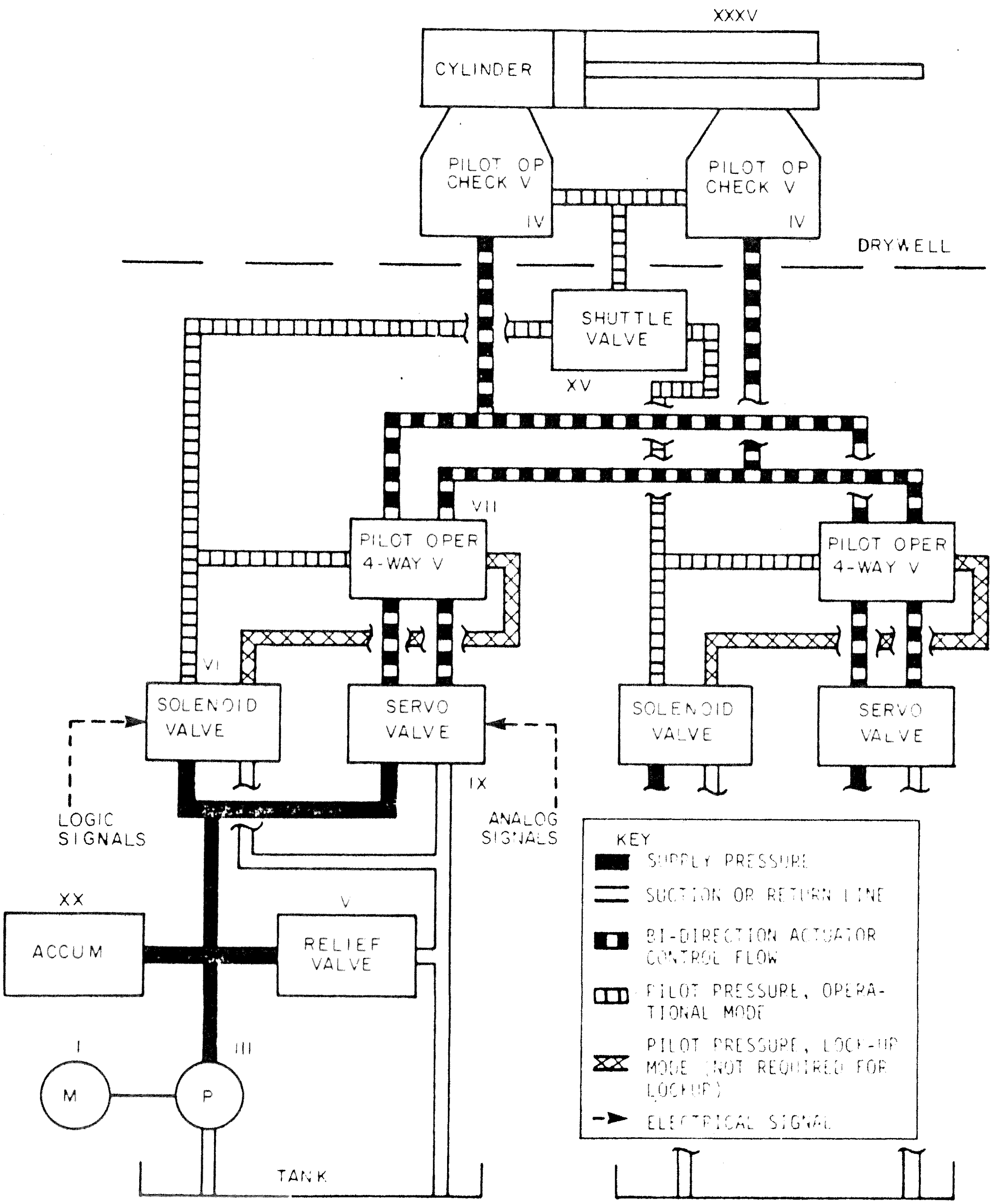
NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
		B. Restriction Between Pump and Servo Valve, Servo Valve and Actuator, or Servo Valve and Tank.	Partially Closed Valve.	Decrease in Differential Pressure Available to Drive Actuator	U	None	D	3
		C. Restriction Between Pump and Pilot Circuits.	Partially Closed Valve.	Increase in Time Required to Unlock Actuator. No Effect During Operation But May Cause a Transfer or Shutdown During Startup.	U (If Operating) V, A (If Startup Affected).	None	N (If Operating L (During Startup).	
		D. Restriction Between Pilot Circuits and Tank.	Partially Closed Valve.	Increased Time Required to "Lock" Actuator.	U	Pump (s) Are Stopped Whenever Lockup is Initiated. After Pump is Stopped, Normal Internal Leakage Will Cause Pressure to Decay Allowing Springs to Close Pilot op. Check Valves and 4-Way Valves.	N	8
		E. External Leakage From Redundant Circuits	Seal Worn, Loose Fitting, Etc. Earthquake	Low Fluid Level	V, A	Low Level Interlock	T	1, 18

TABLE G.B.2-2
(SHEET 16 OF 16)

FLOW CONTROL VALVE ACTUATOR AND HYDRAULIC POWER

UNIT FAILURES MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
		F. External Leakage From Common Circuits.	Seal Worn, Loose Fitting, Etc. Earthquake	Low Fluid Level.	V, A	None	T L	
		G. Loss of Pressure Integrity in Redundant Circuits.	Rupture of Component or Piping. Earthquake	Low Fluid Level, Possibly Low Pressure Also, Flow and/or Pressure May be Unavailable to Drive Actuator or Maintain Pilot Valves in Open Position.	V, A	Low Level Interlock, Low Pressure Interlock, Servo Error Interlock.	Actuator May Not Respond to Changes in Control Signal. T	1, 18, 14
		H. Loss of Pressure Integrity in Common Circuits.	Rupture of Component or Piping Earthquake	Low Fluid Level, Possibly Low Pressure Also, Flow and/or Pressure May be Unavailable to Drive Actuator or Maintain Pilot Valves in Open Position. If Rupture Occurs in Open and Closed Lines, Less Restriction in Circuit.	V, A	None	Actuator May Not Respond to Changes in Control Signal. T L	8, 14
		I. Air Leakage Into Circuit	Loose Suction Line Fitting, Worn Pump Shaft Seal, Etc. Earthquake	Loss of Fluid Stiffness, Noise, Pressure Fluctuations.	V	Circuit Design is Such that Pump Suction is Usually Flooded (No Vacuum) to Prevent Occurrence. Also, Small Volumes of Air Will be Bled Back to Tank Before Problems are Caused.	Oscillation	16



LA SALLE COUNTY STATION

FINAL SAFETY ANALYSIS REPORT

FIGURE G.B.2-1

VALVE ACTUATOR HYDRAULIC POWER UNIT

G.B.3 Flow Control Instrumentation and Controls Failure Modes and Effects Analysis

The Failure Modes and Effects Analysis (FMEA) and reliability analysis of the digital Reactor Recirculation Flow Control (RRFC) system is provided in Reference G.B.7-1.

NOTE 1

No input signal failure or internal component failure can have an effect more severe than sudden failure of the controller output to an extreme high- or low-value. Failure in a flow controller affects only one recirculation loop in the same way that a flux controller failure affects both simultaneously, thus a flow controller failure is less severe than a flux controller failure.

NOTE 2

No input signal failure or internal component failure can have an effect more severe than sudden failure of the hydraulic servovalve to admit full hydraulic fluid flow to open or close the recirculation flow control valve. The servovalve "hard-over" failure effect is discussed in Subsection G.3.3.3. A failure in the opening direction may cause neutron flux to peak above the scram setpoint and trip the reactor; otherwise, flux will peak and settle to a normal maximum flow operating condition. A failure in the closing direction will cause flow to decrease to a minimum flow operating condition.

TABLE G.B.3-1

FLOW CONTROL INSTRUMENTATION AND CONTROLS FAILURE

MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTOMS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
1	Flow Control (A or B Loop) (Ganged Control)	Output Signal High/Low	Electrical/ Electronic Failure	See Note 1	Recirc System Instrumentation	11% Rate Limiter by Valve Positioning Electronics	Flow Increase or Decrease	
2	Valve Positioning Electronics	Drive Hydraulic Servo Valve to Extreme Position	Electrical/ Electronic Failure	See Note 2	Recirc System Instrumentation	None	Flow Increase, Decrease, or Lockup	

G.B.4 Recirculation Pump Trip (RPT) Failure Modes and Effects Analysis

TABLE G.B.4-1
(SHEET 1 OF 4)

RECIRCULATION PUMP TRIP (RPT) FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
1	Sensor Fuse F8A (Typ for F8B, C or D)	Fail Open	Broken Wire, Loose Connection, Etc.	Fuse Open Without Short Circuit or Overcurrent. Relay K8A Deenergized. Failure in Trip (Safe) Direction. RPT is 1:1 Instead of 2:2 for Turbine Control Valve Fast Closure Trip.	Annunc & Comp	Fail-safe Logic Design	Negligible	
2	Turbine Control Valve Fast Closure Sensor C71-N005A (Typ for N005C, N005B, N005D)	Fail Closed	Mechanical Failure, Contact Shorted or Fused, Etc.	Relay K8A Engergized Contact 11-12 Open, Turbine Control Valve Trip is Inhibited.	Periodic Test	Redundant RPT System B Trip is	Minor, Pump Trip is 1:1	
		Fail Open	Broken Wire, Loose Connection, Etc.	Relay K8A Deenergized Fail-Safe, Contact 11-12 Closed. RPT is 1:1 Instead of 2:2 for Turbine Control Valve Fast Closure Trip	Annunc & Comp	Fail-Safe Logic Design	Negligible	
3	Sensor Fuse F9A (Typ for F9B, C, or D)	Fail Open	Broken Wire, Loose Connection, Etc.	Fuse Open Without Short Circuit or Overcurrent Relay K9A Deenergized RPT Bypass is 1:1 Instead of 2:2 for Reactor Power Above 30% Trip Permissive	Annunc & Comp	Fail-Safe	Negligible	
4	Reactor Power Sensor C71-N003A, (Typ for N003B, C, or D)	Fail Closed	Mechanical Failure, Contact Shorted or Fused, Etc.	Relay K9A Energized, Contact 11-12 Open, Reactor Power Trip Permissive is Inhibited RPT is 1:2 Instead of 2:2	Periodic Test is Available	Redundant RPT System. B Trip is Available	Minor, Pump Trip is 1:1	
		Fail Open	Broken Wire, Loose Connection, Etc.	Relay K9A Deenergized, Contact 11-12 Closed, RPT Bypass Failure in Safe Direction	Annunc & Comp	None	Negligible	
5	Sensor Fuse F10A (Typ for F10E, C, G, B, F, D, or H)	Fail Open	Broken Wire, Loose Connection, Etc.	Fuse Open Without Short Circuit or Overcurrent, Relay K10A Deenergized RPT is 1:2 Instead of 1:2 Twice for Turbine Stop Valve Trip; Failure in Safe Direction	Annunc & Comp	Fail-Safe Logic	Negligible	

TABLE G.B.4-1
(SHEET 2 OF 4)

RECIRCULATION PUMP TRIP (RPT) FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
6	Turbine Stop Valve Sensor (C71-N006A) (Typ for N006E, C, G, B, F, D, or H)	Fail Closed	Mechanical Failure, Contact Shorted or Fused Etc.	Relay K10A Energized, Contact 9-10 Open, Turbine Stop Valve Trip is 1:1 and 1:2 Instead of 1:2 Twice	Periodic Test	Redundant RPT System B Operational	Negligible	
		Fail Open	Broken Wire, Loose Connection, Etc.	Relay K10A Deenergized Contact 9-10 Closed, Stop Valve Trip is 1:2 and 2:2 Instead of 1:2 Twice; Fail-Safe.	Annunc & Comp	Fail-Safe Logic Design	Negligible	
7	Logic Fuse F31A, (Typ for F31B, F32A, F32B)	Fail Open	Broken Wire, Loose Connection, Etc.	Fuse Open Without Short Circuit or Overcurrent RPT is 1:1 Instead of 1:2	BKR Trip Coil Status Light is Not on	Bus B Trip is Available	Minor Pump Trip is 1:1	
8	Test Bypass Switch S13A (Typ for S13B)	Fail Closed In Bypass Position	Mechanical Failure, Contact Shorted or Fused, Etc.	Normally Open Contacts Closed in ‘Bypass’ Position	Periodic Test	Logic Design	Both Recirc Pumps Might Trip Surveillance Test of Logic	
		Fail Open in ‘Normal’ Position	Mechanical Failure, Loose Connection, Etc.	Contacts Open, RPT System A is Disabled	Annunc	Redundant RPT System B Trip is Available	Minor, Pump Trip is 1:1	
9	Diode CR3A (Typ For CR3B, CR4A, CR4B)	Fail Shortened	Mechanical Failure, Fused, Etc.	No Symptoms	Periodic Test		No Safety Impact – If Both Diodes Short, Will Not Be Able To Ascertain Individual Trip Coil Continuity. RPT Still Operative.	
		Fail Open	Broken Wire, Loose Connection, Etc.	Diode Open, Pump BKR CB3A Cannot Trip, System A is Disabled	Periodic Test (Shutdown)	Redundant RPT System B Trip is Available	Minor, Pump Trip is 1:1	

TABLE G.B.4-1
(SHEET 3 OF 4)

RECIRCULATION PUMP TRIP (RPT) FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
10	RPT Aux Relay K30A (Typ for K30B) Contacts	Fail Closed	Mechanical Failure, Contacts Fused, Etc	Contacts Closed, Isolated Signals for Start Permissive to LFMG Set Start Circuit	Computer	None	None	
		Fail Open	Broken Wire, Loose Connection, Etc..	Contacts Open, LFMG Auto Start Circuit Inhibited	Periodic Test	None	None	
	RPT Relay K30A Coil	Fail Short		On Trip, Fuse Will Blow, Disabling System A.	Shutdown Test Only	Redundant RPT System B Operative		
		Fail Open		No Annunciation, Computer Indication.	Shutdown Test	Redundant RPT System B Operative		
11	Circuit Breaker Status Indicating Lamp DS27A (Typ for DS27B, DS28A, DS228B)	Fail Shorted	Mechanical Failure, Contacts Shorted, Etc.	Circuit Breaker CB3A Trips, Reactor on One Loop Flow	Speed Indication	None	None	Extremely Remote
					Decreased Reactor Power			Postulated Failure
		Fail Open	Mechanical Failure, Loose Connection, Etc.	Loss of CB3A Indication Status of RPT Trip Coil	Redundant Indicating Light	None	Negligible	
12	Circuit Test Lamp DS25A (Typ for DS25B, C, D, DS26A, B, C, D)	Fail Closed	--	--	--	Redundant Trip System B Operative	--	
		Fail Open	Mechanical Failure, Loose Connection, Etc.	Loss of Periodic Logic Test Indication	Periodic Test	None	None	

TABLE G.B.4-1
(SHEET 4 OF 4)

RECIRCULATION PUMP TRIP (RPT) FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
13	RPT Circuit BKR CB3A (Typ for CB3B, CB4A, CB4B)	Fail Closed	Mechanical Failure, Loose Connection, Etc.	CB3A Does Not Trip Open. RPT System A is Defeated.	Indicating Lights	Redundant System B Trip is Available	Minor, Pump Trip is 1:1	
		Fail Open	Mechanical Failure Hot Short To Trip Coil	CB3A Trips Without Trip Signal, Reactor on One Loop Flow	Speed Indication. Decreased Reactor Power	None	None	
14	Trip Channel A1, A2, Sensor 120 VAC Power Supply (Typ for Channel B1, B2)	Fail Deenergized	Feeder Breaker Open, Loose Connection, Etc.	Turb Cont Valve Trip is 1:2 Instead of 2:2 for BUS A&B: Turb Stop Valve Trip is 1:2 Instead of 1:2 Twice for BUS A&B; 30% Power Bypass is 1:1 Instead of 2:2 for BUS A&B.	Annunc & Comp	None	Logic BUS is in Half Trip for BUS A&B	

G.B.5 Control Interlocks Failure Modes and Effects Analysis

TABLE G.B.5-1
(SHEET 1 OF 7)

CONTROL INTERLOCKS FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
1	A Logic Circuit Reactor Low-Power LMFG Auto Transfer Sequence (Typical for B Logic Circuit)	A. F067A (F023A Limit Switch Stack 10-10C Fails Closed and 11-11C Fails Open with Valve Closed	Limit Switch Stack Mechanical	Allows Recirc Pump Start with Discharge (Suction) Valve Closed Pump Energy Will Heat Up Dead Headed Recirc Water	Flow Indicator	If 10-10C Only Fails, Pump Will Trip on Lockout When Breaker Closes	Possible Pump Damage	
		F067A (F023A) Limit Switch Stack 10-10C Fails Open with Valve Open	Stack Mechanical Failure, Open Circuit	Prevents Recirc Pump Start or Transfer to High Speed	Failure to Start as Seen on Ind Lights Speed Ind		Limits Capacity	
		F067A (F023A) Limit Switch Stack 11-11C Fails Closed with Valve Open	Stack Mechanical Failure, Short Circuit	Recirc Pump Breaker CB3A Trips	Alarm, Recirc Pump Motor-A Trip		Limits Capacity	
		B. FCV F060A Min Position Limit Switch Fails Closed with Valve Full Open	Short Circuit, Mechanical Failure	Allows Recirc Pump to be Started to 25% Speed or 100% Speed with the Flow Control Valve Fully Open	Valve Position Indicator, Loop Flow Ind.	Operating Procedures Require Closing Valve Before Initiating Start Sequence	Possible Time Delay Overcurrent Trip	Reactor Subjected to Flow Increase Transient
		FCV F60A Min Position Switch Fails open with Valve Closed to Min Position	Open Circuit, Mechanical Failure	Prevents Recirc Pump Start or Transfer to High Speed	Speed Indication, Ind. Lights		Limits Capacity	

TABLE G.B.5-1
(SHEET 2 OF 7)

CONTROL INTERLOCKS FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
1		C. K103 Relay Fails Closed When Control Power is not Available to CB3A	Short Circuit on Contacts	Recirc Pump Gets Start Signal, LFMG Set Starts But Main Breaker Fails to Close. During Low to High Speed Transfer, LFMG Set Trips But Main Breaker Fails to Close	Ind. Lights Speed Ind. Alarm – Incomplete Sequence		Limits Capacity	This Occurrence During Low to High Speed Transfer Results in Recirc Pump Trip from 25% Speed
		K103 Relay Fails Open When Control Power is Available to CB3A	Open Circuit to Coil or Contacts	Prevents Recirc Pump Start Transfer to High Speed	Ind. Lights, Speed Ind. Alarm - Circuit Not Available		Limits Capacity	
		D. Contacts K138A, 4-8 Fail to Open on Incomplete Sequence	Short Circuit	Allows Operator to Initiate Start/Transfer Sequence Without First Resetting Incomplete Sequence Logic After an Incomplete Sequence Start		Breaker Protective Relays, Operator Action	Possible Pump Motor Over-heating from Excessive Starts	Operator Would Have to Ignore Incomplete Sequence Alarm
		Contacts K138A, 4-8 Fail Open	Open Circuit	Prevents Recirc Pump Start or Transfer to High Speed	Ind. Lights Speed Indicator		Limits Capacity	
		E. 1, 1T Contacts of S101A Control Switch Fail Closed	Short Circuit	Recirc Pump Starts or Transfer from Low to High Speed Occurs Providing all Other Interlocks are Satisfied			Unexpected Pump Start or Low High Speed Transfer	Operating Procedures Required Operator to Trip Pump if Loop Flow Imbalance Occurs.
		1, 1T Contacts of S101A Control Switch Fail Open	Open Circuit	Prevents Recirc Pump Start or Transfer to High Speed				

TABLE G.B.5-1
(SHEET 3 OF 7)

CONTROL INTERLOCKS FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
1		F. K111A Remains Energized When Control Power is not Available to LFMG Set 1) Motor Breaker 2) Gen Breaker	Short Circuit	Attempted Start of Recirc Pump Results in Pump Accelerating to 100% Speed as LFMG Set Fails to 1) Start 2) Close in at 25% Speed	Speed Ind. Ind. Lights Alarm – Recirc Pump Motor A Auto Transfer Incomplete Alarm – Circuit Not Available			Minor Flow Increase/Decrease Transient <30%/Second

TABLE G.B.5-1
(SHEET 4 OF 7)

CONTROL INTERLOCKS FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
2	A Logic Circuit Reactor Normal Power Level Low Speed to High Speed Transfer (Typical for B Logic Circuit)	A. Contacts 3-4 of K113A Circuit Fail Closed	Short Circuit	This Seal-Out Contact Prevents Sustained High Speed Operation When Recirc Pump is Started to 25% Speed and is Redundant to the digital Reactor Water Level Control System		The Digital Reactor Water Level Control System	Loss of Redundancy	
		B. Contacts 3-4 of K128A Circuit Fail Closed	Short Circuit	Allows Attempted High Speed Transfer with Low Feedwater Flow. Trips LFMG Set and Pump Speed to Zero. Operating Procedures Prevent Transfer	Low FW Flow Indicating Light. Alarm Auto Trip or Transfer to Low Speed	5-6 Contacts of K128A Will Trip Pump on Low FW Flow if Operator Error Initiates Transfer		Failure of K128A to Energize on Low FW Flow May Result in Cavitation Without Pump Trip
		C. Contacts 3-4 of K129A	Short Circuit	Allows Attempted High Speed Transfer With Vessel Low Level. Trips LFMG Set and Pump Speed to zero. Operating Procedures Prevent Transfer	Low Level Indicating Light Alarm – Auto Trip or Transfer to Low Speed	5-6 Contacts of K129A Will Trip Pump on Low Level if Operator Error Initiates Transfer		Failure of K129A to Energize on Low Level May Results in Cavitation Without Pump Trip
		D. Contacts 3-4 of K130A	Short Circuit	Allows Attempted High Speed Transfer With Low Steamline to Recirc Loop ΔT. Trips LFMG Set and Pump Speed to Zero. Operating Procedures Prevent Transfer	Low ΔT Indicating Light Alarm – Auto Trip or Transfer to Low Speed	5-6 Contacts of K130A Will Trip Pump on Low ΔT if Operator Error Initiates Transfer		Failure of K130A to Energize on Low ΔT May Result in Cavitation Without Pump Trip
		E. Contacts 3-4 of K132A	Short Circuit	Allows Attempted High Speed Transfer Following RPT Signal. Trips LFMG Set and Pump Speed to Zero. Pump Cannot Start Due to Trip Signal Directly from RPS System	Alarm – Auto Trip or Transfer to Low Speed	3-4 Contacts of K132A Will Trip Pump on RPT	None	Low Reactor Power Level and Low FW Flow Also Prevent High Speed Operation Following RPT/SCRAM Event

TABLE G.B.5-1
(SHEET 5 OF 7)

CONTROL INTERLOCKS FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
3	A Logic Circuit High Speed to Low Speed Auto Transfer Sequence (Typical for B Logic Circuit)	A. Contacts 3-3T of S101A Fail Closed	Short Circuit	Placing Control Switch S101B in Transfer M/G Will Not Change Status of Pump or LFMG Set	Ind. Lights		None	
		B. Contacts 8-8T of S101A Fail Closed	Short Circuit	Placing Control Switch S101A in Transfer MG Will Not Change Status of Pump or LFMG Set	Ind. Lights			
		C. Contacts 3-3T of S101A Fail Open	Open Circuit	Placing Control Switches S101A and S101B in Transfer M/G Will Result in Recirc Pump 'B' Transferring from High to Low Speed and Recirc Pump A Tripping to Zero Speed	Speed Ind.		Must Manually Restart Recirc Pump A to 25% Speed	
		D. Contacts 3-3T of S101B Fail Open	Open Circuit	Same as 3.C Above Except Pump 'A' Transfers to Low Speed and Pump 'B' Trips to Zero Speed	Same as 3.C Above		Same as 3.C Above	
		E. Contacts 1-2 of K132A Fail Open	Open Circuit	LFMG Set Will Fail to Close in at 25% Speed Following RPT	Speed Indication After RPT	Alarm – Auto Trip or Transfer to Low Speed	Must Manually Restart Recirc Pump to 25% Speed	
		F. Contacts 1-2 of K128A Circuit Fail Open	Open Circuit	LFMG Set Will Fail to Close in at 25% Speed on Low Total Feedwater Flow	Speed Indication	Same as 3.E above	Must Manually Restart Recirc Pump to 25% Speed	
		G. Contacts 1-2 of K129A Circuit Fail Open	Open Circuit	LFMG Set Will Fail to Close in at 25% Speed on Low Vessel Water Level	Speed Indication	Same as 3.E Above	Must Manually Restart Recirc Pump to 25% Speed	

TABLE G.B.5-1
(SHEET 6 OF 7)

CONTROL INTERLOCKS FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
3		H. Contacts 1-2 of K130A Fail Open	Open Circuit	LFMG Set Will Fail to Close in at 25% Speed on Low Steamline to Recirc Loop ΔT	Speed Indication	Same as 3.E Above	Must Manually Restart Recirc Pump to 25% Speed	
		I. Contacts 1-2 of K132A or 1-2 of K128A or 1-2 of K129A or 1-2 of K130A Fail Closed	Open Circuit	No Transfer of Pump to Low Speed will Result.				
		J. Contacts 5-6 of K125A Fail Open	Open Circuit	Failure of LFMG Set to Close in at 25% Speed Following RPT, Low FW Flow, Low Level or Low ΔT	Speed Ind. Alarm – Trip or Transfer to Low Speed		Must Manually Restart Recirc Pump to 25% Speed	
		K. Contact of Breaker 3B Fails Closed With Breaker Closed	Short Circuit	Initiation of High Speed to Low Speed Transfer Sequence will Result in Normal Transfer of Both Pumps to Low Speed				
		L. Contact of Breaker 3B Fails Open with Breaker Open	Open Circuit	Initiation of High-Low Speed Transfer Sequence Will Result in 'A' Pump Running at 25% Speed and 'B' Pump Trip to Zero Speed	Speed Ind.		Must Manually Restart Pump to 25% Speed	
		M. Contacts Open 3-4 of K111A Fail Open	Open Circuit	Initiation of High-Low Speed Transfer Sequence Will Result in 'B' Pump Running at 25% Speed and 'A' Pump Trip to Zero Speed	Speed Ind.		Must Manually Restart Recirc Pump to 25% Speed	

TABLE G.B.5-1
(SHEET 7 OF 7)

CONTROL INTERLOCKS FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
3		N. Contact 3-4 of K111A Fail Closed With no Control Power to LFMG Set 1) Motor Bkr 2) Gen Bkr	Short Circuit	Initiation of High-Low Speed Transfer Sequence Will Result in 'B' Pump Running at 25% Speed and 'A' Pump Trip to Zero Speed	Alarm – Low Speed Circuit not Available		Must Manually Restart Recirc Pump Once Breaker Control Power is Restored	

G.B.6 Power Supplies Failure Modes and Effects Analysis

TABLE G.B.6-1
(SHEET 1 OF 15)

POWER SUPPLIES FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
1	LFMG set motor Breaker 1A (Typical for Bkr 1B*)	A. Breaker Trips	Overcurrent short circuit, relay failure	Opening Breaker 1A* causes Breaker 2A* to rip if initially closed. Recirc pump trips if initially powered by LFMG set. LFMG set not available to run recirc at 25% speed.	Ind. Lights, alarm, “Recirc MG Set-A Protective Relay Trip”		Loss of ability to start recirc pump at low reactor power level and to transfer from high speed to low speed.	Recirc pump trip from 25% speed is less severe than trip from 100% speed.
		B. Breaker will not close	Blown fuse or open circuits	Cannot start MG set.	Ind. Lights alarm, “Low Speed Auto Transfer Circuit” not available		Loss of ability to start recirc pump at low reactor power level and to transfer from high speed to low speed.	
		C. Breaker will not trip	Blown fuse or open circuit	Cannot stop MG set	Ind. Lights			LFMG set interlocks with main power source are through gen output breaker

*GE Designation

TABLE G.B.6-1
(SHEET 2 OF 15)

POWER SUPPLIES FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
1								System reacts normally in a low speed to high speed transfer but MG set continues to run.
2	LFMG set gen Breaker 2A* (Typical for Bkr 2B*)	A. Closes when Bkr 3A* and Bkr 4A* are closed and Bkr 1A* is open	11, 12 contacts of K113A or 9, 10 Contacts of K121A fail-closed plus K126A energizes, plus K122A energizes, plus K134A energizes plus 1E005/b deenergizes plus 1A021/ a energizes <u>OR</u> Jumper installed Contact 11 of K113A to Contact 7 of K134A in Panel	Bkr 2A* trips from Bkr 1A*/b contact in trip circuit	Breaker indi- Bkr 1A*/b cating lights contact green to red		Momentary in-rush of currents to generator. Possible rotation of LFMG set, possible over heating of generator windings.	

*GE Designation

TABLE G.B.6-1
(SHEET 3 OF 15)

POWER SUPPLIES FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
2			P001A plus jumper of Bkr 3A*/b in swgr 151 plus jumper of Bkr 1A*/a in swgr 14ly OR 125-Vdc hot short to closing coil of Bkr 2A* OR Mechanical failure of closing mechanism in Bkr 2A*					
		B. Same as 2.A plus Bkr 2A* fails to trip from Bkr 1A*/b contact in trip circuit	Same as 2.A plus open circuit on Bkr 1A*/b contact	Bkr 2A* senses overcurrent and trips on lockout.	Alarm – Recirc MG set “A” protective relay trip, breaker ind. Lights	86G lockout relay	Momentary overcurent, not visible effect.	

*GE Designation

TABLE G.B.6-1
(SHEET 4 OF 15)

POWER SUPPLIES FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
2		C. Same as 2.C Plus Bkr 2A* fails to trip on overcurrent	Same as 2.B plus simultaneous failure of instantaneous overcurrent relays on ϕA and ϕC or failure of 86G to energize.	Bkr 3A* senses overcurrent and trips Bkr 3A* and Bkr 2A* on lockout.	Alarm – recirc pump motor "A" trip, breaker ind. Lights	86 lockout relay	Momentary overcurrent, no visible effect.	Reactor under goes recirc pump trip transient.
		D. Same as 2.C plus Bkr 3A* fails to trip on overcurrent	Same as 2.C plus simultaneous failures of instantaneous overcurrent relays on ϕA and ϕC or failure of 86 to energize.	Generator subjected to 25-50 times normal in-rush current, tends to overspeed to 4 times normal speed, torque values 625 times normal, excessive winding heating, winding de-flection forces 625 times normal, overvoltage – 10 times normal. Estimate motor and generator wave. 50 percent probability of sustaining overspeed. Exciter would not sustain overspeed. Above conditions would exist until LFMG set	Same as 2.A, B and/or plus speed indication	Manual trips available. LFMG set positioned such that high velocity projectiles emanating from the LFMG set will not damage plant equipment.	Major damage to LFMG set.	Reactor under goes recirc pump trip transient with coastdown time within the limits of normal trip and pump seizure.

*GE Designation

TABLE G.B.6-1
(SHEET 5 OF 15)

POWER SUPPLIES FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
2		D. (Cont'd)		electrically or mechanically separated itself from main power source.				
		E. Closes when Bkr 3A* and Bkr 4A* are closed and Bkr 1A* is closed	11, 12 Contacts of K113A or 9, 10 Contacts of K121A fail closed plus K126A energizes plus K122A energizes plus Bkr 3A*/b deenergizes OR Jumper installed Contact 11 of K113A to Contact 7 of K134A in Panel P001A Plus jumper at Bkr 3A*/b in Swgr 151 OR	Bkr 2A* senses overcurrent tripping Bkr 1A* and Bkr 2A* on lockout.	Alarm - recirc MG set "A" Protective relay trip, breaker ind. Lights	86G lockout relay	Momentary overcurrent, no visible effect	

*GE Designation

TABLE G.B.6-1
(SHEET 6 OF 15)

POWER SUPPLIES FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
			125-Vdc hot short to closing coil to Bkr 2A* OR Mechanical Failure of closing mechanism in Bkr 2A*					
		F. Same as 2.E plus Bkr 2A* fails to trip on over current	Same as 2.E plus simultaneous failure of instantaneous overcurrent relays on ϕA and ϕC or failure of 86G to energize	Bkr 3A* senses overcurrent and trips Bkr 3A*, Bkr 1A* and Bkr 2A* on lockout.	Alarm – Recirc pump motor “A” trip, breaker ind. Lights	86 lockout relay	Momentarily overcurrent. No visible effect	Reactor under goes recirc pump trip transient
		G. Same as 2.F plus Bkr 3A* fails to trip on overcurrent	Same as 2.F plus simultaneous failure of instantaneous over-current	LFMG set subjected to conditions described in 2.D but as motor begins to overspeed, Bkr 1A* senses overcurrent, Bkr 1A* and Bkr 2A* trip on lockout	Alarm - Recirc MG Set “A” protective relay trip, breaker ind. Lights	86 lockout relay	Momentarily LFMG set overspeed and overcurrent	

*GE Designation

TABLE G.B.6-1
(SHEET 7 OF 15)

POWER SUPPLIES FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
		G. (Cont'd)	Relays on ϕA and ϕC or failure of 86 to energize					
		H. Same as 2.G plus Bkr 2A* and Bkr 1A* fail to trip on 86G lockout	Same as 2.E plus simultaneous failure of instantaneous overcurrent relays on ϕA and ϕC or failure of 86G to energize	LFMG set subjected to conditions as described in 2.D until the LFMG set electrically or mechanically separates itself from the 6.9-kV power source and the 4160V power source.		Same as 2D		
		I. Trips	Overcurrent, or relay failure or short circuit	Recirc pump trips, if initially powered by LFMG set. LFMG set not available to run recirc pump at 25% speed	Ind. Lights alarm – MG set a protective relay trip		Loss of ability to start recirc pump at low reactor power level and to transfer from high speed to low speed	Recirc pump trip from 25% speed is less severe than a trip from 100% speed

*GE Designation

TABLE G.B.6-1
(SHEET 8 OF 15)

POWER SUPPLIES FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
		J. Will not close	Blown fuse, or open circuit	Cannot start recirc pump at low reactor power or transfer from high speed to low speed operation	Ind. Lights, alarm – "Recirc Pump A Low Speed Auto Transfer Circuit Not Available"		Prolonged condition limits reactor operating power level	
		K. Will not trip	Blown fuse, or open circuit	Cannot stop recirc pump when powered from LFMG set, cannot transfer low speed to high speed, pump can be tripped from LMFG Breaker 2A*	Ind. Lights, alarm – "Recirc Pump A motor auto transfer incomplete		Prolonged condition limits reactor operating power level	
3	Recirc pump motor Breaker 3A* (typical for Breaker 3B*)	A. Closes when Bkrs 2A, 4A and 1A are closed (recirc pump motor powered from LFMG set)	2, 8 Contacts of K-116A or 1, 2 Contacts of K133A and 3, 4 Contacts of K123A and 6, 6T Contacts of S101A fail closed and Contacts of Bkr 2A*/b fail closed	BKR 2A* senses overcurrent and trips Bkr 1A* and Bkr A* on lockout	Alarm – Recirc MG set "A" protective relay trip, breaker ind. Lights	86G lockout erlay	Recirc pump speed increases from 25% to 100%	Reactor undergoes flow increase transient 30%/sec, possible scram

*GE Designation

TABLE G.B.6-1
(SHEET 9 OF 15)

POWER SUPPLIES FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
			OR Jumper installed 2 to 8 on K116A in Panel P001 plus jumper on Bkr 2A*/b in swgr 151-1. OR Hot short 125-Vdc to 1A aux relay or closing coil in Bkr 3A* OR Mechanical failure of closing mechanism in Bkr 3A*					
		B. Same as 3.A plus Bkr 2A*	Same as 3.A plus simultaneous	Bkr 3A* senses overcurrent and trips Bkr 1A*, 2A* and 3A* on lockout	Alarm – Recirc pump motor "A"	86 lockout relays	Momentarily overcurrent	None

*GE Designation

TABLE G.B.6-1
(SHEET 10 OF 15)

POWER SUPPLIES FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
3		Fails to trip on overcurrent	Failures of instantaneous over current relays on ϕA and ϕC OR Failure of 86G to energize		Trip, breaker ind. lights		rent. No visible effect	
		C. Same as 3.B plus 3A* fails to trip on over-current	Same as 3.B plus simultaneous failures of instantaneous over-current relays on ϕA and ϕC OR Failure of 86 to energize	LMFG set subjected to Conditions described in 2.G but as motor overspeeds Bkr 1A* senses overcurrent and trips Bkr 1A* and Bkr 2A* on lockout	Alarm – recirc MG set "A" protective relay trip, breaker ind. lights	86G lockout	Recirc pump speed increases from 25% to 100%	Reactor under goes flow increase transient <30%/ sec possible scram
		D. Same as 3.C plus Bkr 2A* and Bkr 1A* fail	Same as 3.C plus simultaneous failures of instantaneous	Same as 2.H Plus recirc pump speed increases from 25% to 100T	Same as 2.G	Same as 2.G	Same as 2.G	Reactor under goes flow increase transient <30%/ sec followed by recirc pump trip from 100% or less

*GE Designation

TABLE G.B.6-1
(SHEET 11 OF 15)

POWER SUPPLIES FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
		to trip on 86C lockout	overcurrent relays on ϕA and ϕC OR Failure of 86C to energize					speed, coastdown time within the limits of normal trip and pump seizure
		E. Breaker Trips	Overcurrent, relay failure, short	Recirc pump trips from 100% to 0% speed	Alarm – “Recirc Pump A motor Trip” ind. lights, loop flow			Reactor subjected to flow decrease transient
		F. Breaker will not close	Blown fuse, open circuit	Cannot start recirc pump, cannot transfer from low speed to high speed operation	Alarm - “Recirc Pump A” low speed auto transfer circuit not available		Prolonged conditions limits reactor operating power level	
		G. Breaker will not trip from Coil No. 1 (TC-1)	Blown fuse, open circuit	TC-1 will not trip breaker from turbine trip or load rejection. TC-2 will trip breaker with an additional time delay of a few milliseconds	Indicating lights	Bkr 4A* (TC-1) will trip recirc pump	1/2 trip logic becomes 1/1	

*GE Designation

TABLE G.B.6-1
(SHEET 12 OF 15)

POWER SUPPLIES FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
		H. Breaker will not trip from trip coil No. 2 (TC-2)	Blown fuse, open circuit	Recirc pump will not auto trip from logic interlocks or from fault protection relays	Surveillance test	Manual trip from Bkr 4A* is available	Loss of Recirc loop as operational interlocks and functions. Safety systems not jeopardized.	
4	Breaker 4A* (Typical for Bkr 4B*)	A. Breaker trips	Relay failure short circuit	Recirc pump trips from 100% to 0% speed, Bkr 3A* trips	Alarm “Recirc Pump Motor – 1A Trip” ind. Lights, loop flow			Reactor subjected to flow decrease transient
		B. Breaker will not close	Blown fuse, Open Circuit	Cannot start recirc pump, cannot transfer from low speed to high speed operation	Ind. Lights		Prolonged condition limits reactor operating power level	
		C. Breaker will not trip from trip coil No. 1 (TC-1)	Blown fuse, open circuit	TC-1 will not trip breaker from turbine trip or load rejection	Indicating lights	Bkr 3A* (TC-1) will trip recirc pump	1/2 trip logic becomes 1/1	

*GE Designation

TABLE G.B.6-1
(SHEET 13 OF 15)

POWER SUPPLIES FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
		D. Breaker will not trip from trip coil No. 2 (TC-2)	Blown fuse, open circuit	Breaker will not open from control switch.	Indicating lights	Bkr 3A* not Bkr 4A* is normally used to trip the recirc pump from the main power source	Somewhat reduced operating/maintenance flexibility until fault repaired	
5	240-Vac 1ϕ supply to gen. voltage regulator (Typical for loop B)	A. Loss of power	Feeder breaker trip	Gen field voltage decays to zero. Recirc pump motor decays to zero % speed	Speed ind. Flow ind.		Loss of ability to start recirc pump at low reactor power level and to transfer from high speed to low speed	
		B. Over-current	Short ground	Overload trips 86G relay, locking out LFMG set. LFMG set not available to power recirc pump, recirc pump motor trips from 25% to 0% speed	Alarm, “Recirc MG Set”. A protective relay trip, speed inc.		Same as I	

*GE Designation

TABLE G.B.6-1
(SHEET 14 OF 15)

POWER SUPPLIES FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
6	120-Vac dist. Panels at MCC 131A-2 or 132B-1, Redundant (Non-Div) (Typical for 120-Vac Dist. PNL at MCC 132B-1 (Non-Div)	Loss of Power	Feeder breaker trip	None	Alarm – "Control Signal Abnormal"	Loss of Redundancy	Unable to change recirculation flow with FCV	
7	125-Vdc dis. Panels 111X (Non-Div) and 11Y (Div 1) (Typical for 125-Vdc dist panels 112X (Non-Div) and 112Y (Div 2)	Loss of Power	Main breaker or selected output breakers trip	Loss of control powers to Bkrs 1A*, 2A*, 3A* and 4A*. These breakers will not close or trip if required to do so manually or in response to automatic logic.	Alarms – "Feed Breaker Trip" "Switchgear DC Control Power Failure"	Recirc pump trip safety function maintained by redundant trip coil in Bkr 4A* which is powered by division II control power	Normal recirc breaker operations inhibited pump start, transfer high speed, low speed to high speed. Pump trip on RPT signal not inhibited but LFMG set will not start in this case.	
8	6.9-kV swgr 151 (Typical for 6.9-kV swgr 152)	Loss of Power	Overcurrent feeder breaker faults	Bus undervoltage relay trips Bkr 3A*.	Alarm – "6.9-kV Swgr 151. Voltage Low," "Rx, Recirc Pump 1A Auto Trip."		Recirc pump trips from 100% to 0% speed	Reactor subjected to recircc flow decrease transient.

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TABLE G.B.6-1
(SHEET 15 OF 15)

POWER SUPPLIES FAILURE MODES AND EFFECTS ANALYSIS

NO.	NAME AND FUNCTION	FAILURE MODE	CAUSE	SYMPTONS AND LOCAL EFFECTS INCLUDING DEPENDENT FAILURES	METHOD OF DETECTION	INHERENT COMPENSATING PROVISION	EFFECT UPON SYSTEM	REMARKS AND OTHER EFFECTS
9	4.6-kV swgr (Div 1) (Typical for 4.16-kV swgr 142Y (Div II).	Loss of Power	Overcurrent Simultaneous feeder breaker faults	LFMG set lockout trips recirc pump is powered from LFMG at 25% speed. Operating hydraulic pump trips. Bus undervoltage relay trips Bkr 1A*	Alarms – "4.16-kV Swgr Voltage Low, " – "LFMG set 1A Bkr 1A auto Trip"	FCV lockup circuit	Unable to change recirc flow	

*GE Designation

G.B.7 References

1. Westinghouse Report PNTC 01-031, Reactor Recirculation Flow Control (RRFC) System Reliability Analysis

APPENDIX H – FIRE HAZARDS ANALYSIS

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The entire contents of UFSAR Appendix H – Fire Hazards Analysis, are relocated to the LaSalle County Station Fire Protection Report (FPR). References to UFSAR Appendix H remain valid.

APPENDIX I
INFORMATION RELEVANT TO KEEPING LEVELS OF RADIOACTIVITY IN EFFLUENTS TO UNRESTRICTED AREAS AS LOW AS REASONABLY ACHIEVABLE

Programs and requirements for control of radioactive effluents, calculation of offsite doses, and radiological environmental monitoring are included in the Offsite Dose Calculation Manual.

APPENDIX J

ANALYSIS OF THE EFFECTS OF MODERATE ENERGY LINE THROUGH WALL LEAKAGE CRACKS OUTSIDE PRIMARY CONTAINMENT

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APPENDIX J

ANALYSIS OF THE EFFECTS OF MODERATE ENERGY

LINE THROUGH WALL LEAKAGE CRACKS OUTSIDE PRIMARY CONTAINMENT

LIST OF TABLES

DRAWINGS CITED IN THIS APPENDIX*

DRAWING*	SUBJECT
M-69	Service Water, Unit 1
M-71	Fire Protection
M-72	Fire Protection Sheets (1, 2, and 5)
M-72	Fire Protection Control Room Breathing Air (Sheet 3)
M-72	Fire Protection High Pressure Breathing Air (Sheet 4)
M-74	Cycled Condensate Storage, Unit 1
M-75	Clean Condensate Storage
M-78	Hydrogen & Carbon Dioxide System (Sheet 1)
M-78	Stator Cooling System (Sheets 2 and 3)
M-78	HWC System (Sheet 5)
M-85	Diesel Oil System, Unit 1 (Sheet 1)
M-85	Security Diesel Generator Building (Sheet 2)
M-86	Primary Containment Chilled Water Coolers, Unit 1
M-87	Core Standby Cooling System Equipment Cooling Water System, Unit 1
M-90	Reactor Building Closed Cooling Water, Unit 1
M-94	Low Pressure Core Spray (LPCS), Unit 1
M-95	High Pressure Core Spray (HPCS), Unit 1
M-96	Residual Heat Removal System (RHRS), Unit 1 (Sheets 1-4)
M-96	Suppression Pool Clean Up and Transfer System (Sheet 5)
M-101	Reactor Core Isolation Cooling System, Unit 1
M-102	Station Heat Recovery
M-210	Reactor Building Piping Plan El. 710'-6" Area 3, Unit 1
M-214	Reactor Building Piping Plan El. 710'-6" Area 3, Unit 2
M-219	Reactor Building Piping Plan El. 694'-6" Area 4, Unit 1
M-223	Reactor Building Piping Plan El. 694'-6" Area 4, Unit 2
M-225	Reactor Building Piping Plan El. 673'-4" Area 2, Unit 1
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M-230	Reactor Building Piping Plan El. 673'-4" Area 3, Unit 2
M-414	Diesel Generator Room Piping Plan Basement Floor Area 5, Unit 1
M-415	Diesel Generator Room Piping Plan Basement Floor Area 1, Unit 2
M-419	Diesel Generator Room Piping Plan Basement Floor Area 5, Unit 2

* The listed drawings are included as "General References" only; i.e., refer to the drawings to obtain additional detail or to obtain background information. These drawings are not part of the UFSAR. They are controlled by the Controlled Documents Program.

APPENDIX J - ANALYSIS OF THE EFFECTS OF MODERATE ENERGY LINE THROUGH WALL LEAKAGE CRACKS OUTSIDE PRIMARY CONTAINMENT

J.0 INTRODUCTION

This Appendix summarizes the Moderate Energy Line Break (MELB) Analysis, which was performed in accordance with the guidance provided in:

- NUREG 0800(Standard Review plan), Section 3.6.1
- Branch Technical Position (BTP) ASB-3-1 (provided as Attachment B to NUREG 0800, Section 3.6.1).
- J. F. O'Leary Letter of July 12, 1973 (provided as Attachment C to NUREG 0800, Section 3.6.1).
- Branch Technical Position (BTP) MEB-3-1.

The MELB analysis demonstrates the capability to shutdown the plant and maintain in a cold shutdown condition assuming a MELB (i.e., through wall crack in a moderate energy line) and a single active component failure. The focus of the MELB analysis is the protection of safe shutdown equipment from spray and flooding resulting from a postulated MELB. In accordance with the guidance of Branch Technical Position (BTP) ASB 3-1, the analysis is based upon:

- The postulated MELB occuring during normal plant conditions. It does not occur concurrently with other design basis accidents or transients.
- Offsite power being available. A Loss-of Offsite Power (LOOP) concurrent with the postulated MELB is not considered. The analysis demonstrates that a MELB in a safety-related building will not result a trip of the RPS system or the FW/turbine trip system. A crack in a moderate energy line in the turbine building could possibly result in a trip of the FW/Turbine. However, such a crack would not affect equipment required to achieve cold shutdown.

J.1 MODERATE ENERGY LINES

Moderate energy fluid systems are those that meet the following criteria during normal plant conditions:

a) Maximum operating temperature or pressure is:
    - Maximum operating temperature less than or equal to 200 °F, and
    - Maximum operating pressure less than or equal to 275 psig.

b) High energy piping systems whose pressure exceeds 275 psig or temperature exceeds 200 °F for short periods only (1% of the normal operating life span of the plant, or 2% of the time period required to accomplish its system design function).

c) Nominal pipe size is greater than 1 inch.

d) Minimum pressure is greater than 10 psig.

e) Non-seismic piping whose minimum pressure is ≤ 10 psig.

Moderate energy lines specifically excluded from MELB consideration consist of:

- Seismic Category I piping whose minimum pressure is ≤ 10 psig. Based upon Branch Technical Position (BTP), MEB 3-1, Paragraph B.2.c.3, non-seismic piping whose pressure is ≤ 10 psig is not excluded from MELB consideration.
- Those whose stress has to be demonstrated by analysis to meet the exclusion criteria of MEB 3-1, paragraph B.2.c.1. This is where the stress is less than or equal to:

    1.2 Sm for Class 1 piping

    $0.4(1.2\ S_h + S_A)$ for Class 2, 3, and B31.1 piping

Criterion d was chosen on the following basis:

a) Since the flow rate and the maximum spray distance from a postulated through –wall crack are related to pressure by a square root relationship, the 10 psig is very small.

b) The pipe wall thickness is many times the required minimum wall thickness for such low pressures and therefore, the probability of a rupture occurring is acceptably small.

The moderate energy systems consist of:

- Clean Condensate (MC)
- Cycled Condensate storage (CY)
- Core Standby Cooling System (DG, FC, HP/E22, RH/E12)
- Diesel Oil System (DO)
- Fire Protection (FP)
- Hydrogen and Carbon Dioxide System (HY, CO)
- High Pressure Core Spray (HP/E22)
- Low Pressure Core Spray (LP/E21)
- Reactor Building Equipment Drains (RE)
- Residual Heat Removal (RH/E12)

- Reactor Core Isolation Cooling (RI/E51)
- Station Heat Recovery (SH)
- Primary Containment Chiller Water Coolers (VP)
- Reactor Building Closed Cooling Water (WR)
- Service Water (WS)

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FIGURE J-1.0

ASSOCIATED CONTROLLED DRAWINGS*

FOR J-1 FIGURES

J.2. AFFECTED SAFETY-RELATED EQUIPMENT

Table J.2 identifies each area of the plant that could potentially be affected by a MELB. For each of these areas the Table then identifies whether a MELB is postulated. If yes, the safe shutdown equipment located in the area, if no, the basis for excluding the area. For each area that a MELB is postulated, the Table identifies whether there is any safe shutdown equipment, which requires spray protection. If spray protection is not required, the basis for this conclusion is identified.

Table J.2

Plant Areas and Equipment Affected by MELB

Sheet 1 of 10

Bldg.	Elev.	Area Description	MELB Postulated in Area	Safe shutdown equipment requiring protection	Exclusion Justification
Primary Containment	N/A	Drywell	No	N/A	Excluded per BTP ASB 3-1, paragraph B.1
Primary Containment	N/A	Wetwell	No	N/A	Excluded per BTP ASB 3-1, paragraph B.1
Reactor Building	673'-4"	Raceway	No	N/A	The Raceway at Elev. 673'-4" and 694'-0" represents a common area. Grating at Elev. 694'-0" separates the two elevations. Therefore, environmentally they are one area. This area contains high energy RI steam lines.
Reactor Building	673'-4"	NW corner room (A-RHR)	Yes	No	The NW corner rooms at Elev. 673'-4" and 694'-0" and the RHR heat exchanger room at Elev. 710'-6" comprise a single area with multiple entry points. Therefore, environmentally they are one area. At this elev, the safe shutdown equipment located in this area is the "A" RHR pump 1(2)E12-C002A. During conditions when a MELB is postulated to occur (i.e., normal operating conditions) the RHR equipment functions to support decay heat removal and vessel inventory control. For vessel inventory control refer to discussion under the HPCS pump (Reactor building Elev. 673'-4", SW corner room). For decay heat removal, there are multiple methods available to accomplish this function. They are Div 1 RHR, Div 2 RHR, and several alternate decay heat removal methods. Since the equipment associated with these methods is physically separated, should spray from a moderate energy line crack disable one Division of RHR, and a single active equipment failure disable other Division of RHR, an alternate decay heat removal method would be available.
Reactor Building		NE corner room (LPCS & RCIC)	No	N/A	The NE corner rooms at Elev. 673'-4" and 694'-0" are connected via the stairwell. Therefore, environmentally they are one area. This area contains high energy RI steam lines.

Table J.2

Plant Areas and Equipment Affected by MELB

Sheet 2 of 10

Bldg.	Elev.	Area Description	MELB Postulated in Area	Safe shutdown equipment requiring protection	Exclusion Justification
Reactor Building	673'-4"	SE corner room (B & C RHR)	Yes	No	The SE corner rooms at Elev. 673'-4" and 694'-0" and the RHR heat exchanger room at Elev. 710'-6" comprise a single area with multiple entry points. Therefore, environmentally they are one area. At this elev., the safe shutdown equipment located in this area is the "B" RHR pump 1(2)E12-C002B, and the RHR water leg pump 1(2)E12-C003. During conditions when a MELB is postulated to occur (i.e., normal operating conditions) the RHR equipment functions to support decay heat removal and vessel inventory control. For vessel inventory control refer to discussion under the HPCS pump (Reactor building Elev. 673'-4", SW corner room). For decay heat removal, there are multiple methods available to accomplish this function. They are Div 1 RHR, Div 2 RHR, and several alternate decay heat removal methods. Since the equipment associated with these methods is physically separated, should spray from a moderate energy line crack disable one Division of RHR, and a single active equipment failure disable other Division of RHR, an alternate decay heat removal method would be available.
Reactor Building	673'-4"	SW corner room (HPCS)	Yes	No	The SW corner rooms at Elev. 673'-4" and 694'-0" are connected via the stairwell. Therefore, environmentally they are one area. At this elev., the safe shutdown equipment in this area is associated with the HPCS system (i.e., HPCS pump 1(2)E22-C001, HPCS water leg pump) 1(2)E22-C003, and HPCS instrument panel 1(2)H22-P024). PER BTP ASB 3-1 a MELB is only postulated to occur during normal operating conditions. Since a crack in a moderate energy line is not postulated to occur concurrent with an accident, the accident mitigation function of HPCS is not required. However, HPCS is one of the systems that can be used to provide water inventory to the core. Should HPCS become disabled, other systems are available to provide water to the core. These other systems are RCIC & ADS/LPCS, ADS/RHR-A, ADS/RHR-B & C. Since the components associated with these methods are physically separated, if spray from a moderate energy line crack disabled HPCS and a single active equipment failure disables another method, at least one means would remain available.

Table J.2

Plant Areas and Equipment Affected by MELB

Sheet 3 of 10

Bldg.	Elev.	Area Description	MELB Postulated in Area	Safe shutdown equipment requiring protection	Exclusion Justification
Reactor Building	673'-4"	Rx Bldg drain tank room	Yes	No	Area has no SR equipment
Reactor Building	673'-4"	Raceway	No	N/A	The Raceway at Elev. 673'-4" and 694'-0" represents a common area. Grating at Elev. 694'-0" separates the two elevations. Therefore, environmentally they are one area. This area contains high-energy RI steam lines.
Reactor Building	694'-0"	NW corner room	Yes	No	The NW corner rooms at Elev. 673'-4", 694'-0", and the RHR Heat Exchanger room at Elev. 710'-6" comprise a single common area with multiple entry points. Therefore, environmentally they are one area. At this elev, the only safe shutdown equipment are the RHR Heat Ex and MOVs which are excluded
Reactor Building	694'-0"	NE corner room	No	N/A	The NE corner rooms at Elev. 673'-4" and 694'-0" are connected via the stairwell. Therefore, environmentally they are one area with high energy RI lines.
Reactor Building	694'-0"	SE corner room	Yes	No	The SE corner rooms at Elev. 673'-4", 694'-0", and the RHR Heat Exchanger room at Elev. 710'-6" comprise a single common area with multiple entry points. Therefore, environmentally they are one area At this elev., the only safe shutdown equipment are the RHR Heat Ex and MOVs which are excluded
Reactor Building	694'-0"	SW corner room	Yes	No	The SW corner rooms at Elev. 673'-4" and 694'-0" are connected via the stairwell. Therefore, environmentally they are one area. The area at this Elev. contains no safe shutdown equipment
Reactor Building	694'-0"	Main Steam Tunnel	No	N/A	Area contains high energy FW & MS lines.
Reactor Building	710'-6"	Drywell equip drain room	Yes	No	Area contains no safe shutdown equipment

Table J.2

Plant Areas and Equipment Affected by MELB

Sheet 4 of 10

Bldg.	Elev.	Area Description	MELB Postulated in Area	Safe shutdown equipment requiring protection	Exclusion Justification
Reactor Building	710'-6"	"A" RHR heat ex room	Yes	No	The RHR Heat Exchanger room at Elev. 710'-6" and NW corner rooms at Elev. 673'-4", 694'-0", comprise a single common area with multiple entry points. Therefore, environmentally they are one area. At this elev., the only safe shutdown equipment are the RHR Heat Ex and MOVs which are excluded
Reactor Building	710'-6"	"B" RHR heat ex room	Yes	No	The RHR Heat Exchanger room at Elev. 710'-6" and SE corner rooms at Elev. 673'-4", 694'-0", comprise a single common area with multiple entry points. Therefore, environmentally they are one area. At this elev., the only safe shutdown equipment are the RHR Heat Ex and MOVs which are excluded.
Reactor Building	710'-6"	General area	Yes	No	Only safe shutdown equipment are MCC 480V 1(2)35Y-2 (1(2)AP76E) MCC 480V 1(2)36Y-1 (1(2)AP82E) These MCCs provide electrical power to Div 1 and Div 2 respectively. The specific components being fed are valves associated with the HG, LPCS, RHR, RHR-WS, and MSIV-LCS; water leg pumps associated with LPCS & RI; DO fuel oil transfer pump; Cubicle coolers associated with the RHR, LPCS, RHR-WS; and the MSIV-LCS heaters. As discussed previously, the only components that are required during conditions in which a MELB is postulated to occur are those associated with the decay heat removal and vessel inventory control. For vessel inventory control refer to discussion under the HPCS pump (Reactor building, Elev. 673'-4", SW corner room). For decay heat removal, there are multiple methods available to accomplish this function. They are Div 1 RHR, Div 2 RHR, and several alternate decay heat removal methods. Since the equipment associated with these methods is physically separated, should spray from a moderate energy line crack disable one MCC, and a single active equipment failure disable other MCC, an alternate decay heat removal method would be available.
Reactor Building	740'-0"	General area	No	N/A	Area contains high energy RI steam lines.

Table J.2

Plant Areas and Equipment Affected by MELB

Sheet 5 of 10

Bldg.	Elev.	Area Description	MELB Postulated in Area	Safe shutdown equipment requiring protection	Exclusion Justification
Reactor Building	761'-0"	General area	No	N/A	Area contains high energy RD lines.
Reactor Building	786'-6"	General area	No	N/A	Area contains high energy RT lines.
Reactor Building	807'-0"	RWCU valve room	No	N/A	Area contains high energy RT lines.
Reactor Building	820'-6"	RWCU area	No	N/A	Area contains high energy RT lines.
Reactor Building	820'-6"	SGTS Area	Yes	No	The safe shutdown equipment in this area is 480V MCC 136X-1 (Unit 1), 236X-1 (Unit 2) [1(2)AP78E]. The important loads during postulated MELB conditions are associated with the VC and VX systems. Should spray impair this MCC, sufficient redundancy exists to accomplish the function of the VC equipment even if a single active equipment failure occurs. Procedurally controlled compensatory actions ensure that the function of the VX system can be accomplished should VX equipment be impaired by a MELB.
Reactor Building	832'-0"	Instrument Storage Room	Yes	No	Area has no SR equip
Reactor Building	843'-6"	Refuel Floor	Yes	No	Area has no shutdown equip

Bldg.	Elev.	Area Description	MELB Postulated in Area	Safe shutdown equipment requiring protection	Exclusion Justification
Diesel Gen. Building	674'-0"	Div 1 diesel fuel tank room	Yes	No	A Loss-of Offsite Power (LOOP) is not assumed. Therefore the emergency diesel generators are not required for safe shutdown.
Diesel Gen. Building	674'-0"	Div 2 diesel fuel tank room	Yes	No	A Loss-of Offsite Power (LOOP) is not assumed. Therefore the emergency diesel generators are not required for safe shutdown.
Diesel Gen. Building	674'-0"	HPCS diesel fuel tank room	Yes	No	A Loss-of Offsite Power (LOOP) is not assumed. Therefore the emergency diesel generators are not required for safe shutdown.
Diesel Gen. Building	674'-0"	Div 1 RHR-WS pump room	Yes	No	The safe shutdown equipment is the Div 1 CSCS equipment. Specifically the RHR-WS pumps (1(2)E12-C300A,B), Div 1 DG cooling water pump (0DG01P). Since neither an accident or a LOOP is not postulated concurrent with a crack in a moderate energy line, only the RHR-WS pumps which support decay heat removal, are required to achieve and maintain cold shutdown. There are multiple methods available to accomplish this function of decay heat removal. They are Div 1 RHR & CSCS, Div 2 RHR & CSCS, and several alternate decay heat removal methods. Since the equipment associated with these methods is physically separated, should spray from a moderate energy line crack disable Div 1 of CSCS, and a single active equipment failure disabled other Division of CSCS, an alternate decay heat removal method would be available.

Table J.2

Plant Areas and Equipment Affected by MELB

Sheet 7 of 10

Bldg.	Elev.	Area Description	MELB Postulated in Area	Safe shutdown equipment requiring protection	Exclusion Justification
Diesel Gen. Building	674'-0"	Div 2 RHR-WS pump room	Yes	No	The safe shutdown equipment is the Div 2 CSCS equipment. Specifically the RHR-WS pumps (1(2)E12-C300C,D, Div 2 DG cooling water pump (1(2)DG01P). Since neither an accident nor a LOOP is not postulated concurrent with a crack in a moderate energy line, only the RHR-WS pumps which support decay heat removal, are required to achieve and maintain cold shutdown. There are multiple methods available to accomplish this function of decay heat removal. They are Div 1 RHR & CSCS, Div 2 RHR & CSCS, and several alternate decay heat removal methods. Since the equipment associated with these methods is physically separated, should spray from a moderate energy line crack disable Div 2 of CSCS, and a single active equipment failure disabled other Division of CSCS, an alternate decay heat removal method would be available.
Diesel Gen. Building	674'-0"	HPCS cooling water pump room	Yes	No	The only equipment in this area that is potentially required for safe shutdown is the HPCS diesel generator cooling water pump 1(2)E22-C002. As discussed previously, HPCS is one of the systems that can be used to provide vessel inventory control. Should HPCS become disabled, sufficient other systems are available to provide water to the core. These other systems are RCIC & ADS/LPCS, ADS/RHR-A, ADS/RHR-B & C. Since the components associated with these methods are physically separated, if spray from a moderate energy line crack disabled HPCS and a single active equipment failure be postulated to occur, at least one means would be available.
Diesel Gen. Building	710'-6"	Div 1 DG room	Yes	No	A Loss-of Offsite Power (LOOP) is not assumed. Therefore the emergency diesel generators are not required for safe shutdown.
Diesel Gen. Building	710'-6"	Div 1 diesel day tank room	Yes	No	A Loss-of Offsite Power (LOOP) is not assumed. Therefore the emergency diesel generators are not required for safe shutdown.
Diesel Gen. Building	710'-6"	Div 2 DG room	Yes	No	A Loss-of Offsite Power (LOOP) is not assumed. Therefore the emergency diesel generators are not required for safe shutdown.

Table J.2

Plant Areas and Equipment Affected by MELB

Sheet 8 of 10

Bldg.	Elev.	Area Description	MELB Postulated in Area	Safe shutdown equipment requiring protection	Exclusion Justification
Diesel Gen. Building	710'-6"	Div 2 diesel day tank room	Yes	No	A Loss-of Offsite Power (LOOP) is not assumed. Therefore the emergency diesel generators are not required for safe shutdown.
Diesel Gen. Building	710'-6"	HPCS DG room	Yes	No	A Loss-of Offsite Power (LOOP) is not assumed. Therefore the emergency diesel generators are not required for safe shutdown.
Diesel Gen. Building	710'-6"	HPCS diesel day tank room	Yes	No	A Loss-of Offsite Power (LOOP) is not assumed. Therefore the emergency diesel generators are not required for safe shutdown.
Diesel Gen. Building	736'-6"	Div 1 diesel vent room	Yes	No	A Loss-of Offsite Power (LOOP) is not assumed. Therefore the emergency diesel generators are not required for safe shutdown.
Diesel Gen. Building	736'-6"	Div 2 diesel vent room	Yes	No	A Loss-of Offsite Power (LOOP) is not assumed. Therefore the emergency diesel generators are not required for safe shutdown.
Diesel Gen. Building	736'-6"	HPCS diesel vent room	Yes	No	A Loss-of Offsite Power (LOOP) is not assumed. Therefore the emergency diesel generators are not required for safe shutdown.
Auxiliary Building	687'-0"	Div 3 Essential Swgr room	Yes	No	In Unit 1, area contains no moderate energy lines Unit 2 contains moderate energy line 2RTA1A-3" The only equipment in this area that is potentially required for safe shutdown is 4.16KV switchgear 143 (2AP07E), 480V MCC 143-1 (2AP79E), HPCS Battery (2DC18E), HPCS battery Charger (2DC19E), HPCS relay panel (2H13-P625). This equipment controls HPCS components. As discussed previously, HPCS is one of the systems that can be used to provide water inventory to the core. Should HPCS become disabled, other systems are available to provide water to the core. These other systems are RCIC & ADS/LPCS, ADS/RHR-A, ADS/RHR-B & C. Since the components associated with these methods are physically separated, if spray from a moderate energy line crack disabled HPCS and a single active equipment failure be postulated to occur, at least one means would be available.

Table J.2

Plant Areas and Equipment Affected by MELB

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Bldg.	Elev.	Area Description	MELB Postulated in Area	Safe shutdown equipment requiring protection	Exclusion Justification
Auxiliary Building	710'-6"	Div 1 Essential Swgr room	No	N/A	Area contains no moderate energy lines
Auxiliary Building	710'-6"	Labs & Offices	Yes	No	Area contains no safe shutdown equipment
Auxiliary Building	731'-0"	Div 2 Essential Swgr room	No	N/A	Area contains no moderate energy lines
Auxiliary Building	731'-0"	AEER	No	N/A	Area contains no moderate energy lines
Auxiliary Building	749'-0"	Cable spreading room	Yes	No	Area contains no safe shutdown equipment
Auxiliary Building	749'-0"	Elec equip room	Yes	No	Area contains no safe shutdown equipment
Auxiliary Building	768'-0"	Main control room	No	N/A	Area contains no moderate energy lines
Auxiliary Building	768'-0"	Computer room	Yes	No	Area contains no safe shutdown equipment
Auxiliary Building	768'-0"	Security control center	Yes	No	Area contains no safe shutdown equipment
Auxiliary Building	768'-0"	Offices	Yes	No	Area contains no safe shutdown equipment

Table J.2

Plant Areas and Equipment Affected by MELB

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Bldg.	Elev.	Area Description	MELB Postulated in Area	Safe shutdown equipment requiring protection	Exclusion Justification
Auxiliary Building	786'-6" and 802'-0"	Lower vent equip floor	Yes	No	The safe shutdown equip is: 0PL15J; 0PL16J; 0PL42J; 0PL43J; 0VC02AA,AB; 0VC01CA,CB; 0VC02CA,CB; 0VC04CA,CB; 0VC05CA,CB; 0VE01AA,AB; 0VE01CA,CB; 0VE02CA,CB; 0VE03CA,CB; 0VE04CA,CB; Various dampers. The fans are vaneaxial fans where the motor is located internal to the fan housing and therefore is protected from spray. Correspondingly the damper actuators are enclosed, protecting electrical items from spray. The cooling coils have no electrical items. Therefore, the safe shutdown components that could potentially be affected by spray from a crack in a moderate energy line are the control panels and the motor associated with the refrigeration compressor. Since a design basis accident is not postulated concurrent with a crack in a moderate energy line, this equipment is not required for to maintain the habitability of the control room envelope (control room and AEER). The equipment is however required to maintain the temperature in the control room envelope to the design temperature for the equipment. The plant may be shutdown from the control room or from the AEER via the remote shutdown panel. There are redundant systems to provide cooling to the control room (Unit 1 VC system and Unit 2 VC system) and to the cool the AEER (Unit 1 VE system and Unit 2 VE system). Therefore, a total of four systems are available to accomplish the function of maintaining a suitable environment for equipment in the control room envelope that is used to safely shut down the plant and maintain it in a safe shutdown condition. Due to physical separation of the equipment, spray from a moderate energy line may affect one or two of these systems, but the other systems will remain available. Therefore, safe shutdown can be achieved, assuming a single active equipment failure.
Auxiliary Building	815'-0"	Upper vent equip floor	Yes	No	Area contains no safe shutdown equipment

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FIGURE J-2.0 ASSOCIATED CONTROLLED DRAWINGS* FOR J-2 FIGURES

J.3 SPRAY SHIELDS

Safe shutdown can be achieved without spray shields being required to protect equipment from the spray from postulated cracks in moderate energy lines. As discussed in Section J.2, the basis for this is that sufficient separation and redundancy exists.

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FIGURE J-3.1
TYPICAL CROSS-SECTION DETAIL
OF SPRAY SHIELD

J.4 EFFECTS OF FLOODING

Due to the presence of floor drains and stairwells, the safe shutdown equipment that could potentially affected by postulated flooding resulting from a crack in a moderate energy line is that located in lowest elevations. This is the RHR pump motors and the RHR service water pump motors.

The Division 1 and Division 2 RHR pumps are located in separate watertight rooms with bulkhead doors that open to the raceway. These doors protect these rooms from the postulated failure of the suppression pool. Should the postulated crack occur outside these rooms, the safe shutdown equipment in the room is protected. Should the postulated crack occur inside one of these rooms, any flooding would be limited to that room. As discussed in the above table, if such spray disabled the equipment, sufficient redundancy exists, to ensure the capability to accomplish the safe shutdown function assuming a single active equipment in a redundant division or system. Each of these rooms have sump pumps that alarm upon high water level. Thus the operator would receive indication of a leak and take action to isolate it.

The Division 1 and Division 2 RHR service water pumps are also located in separate watertight rooms with bulkhead doors. These doors protect any flooding from propagating to the adjacent rooms. Only the Division 2 room has a door that communicates with the Turbine Building. This is acceptable because no credible flood source has been identified in the Turbine Building. UFSAR Section 3.4.1.4 (Flood Protection Measures) identified these flood sources in the Turbine Building and explains why they are not credible. Should the postulated crack occur inside one of these rooms, any flooding would be limited to that room. As discussed in the above table, if such spray disabled the equipment, sufficient redundancy exists, to ensure the capability to accomplish the safe shutdown function assuming a single active equipment in a redundant division or system. Each of these rooms have sump pumps that alarm upon high water level. Thus the operator would receive indication of a leak and take action to isolate it.

J.5 CONCLUSION

The MELB analysis postulates that a moderate energy line outside of the primary containment develops a through wall crack which results in spray. This crack is postulated to occur during normal operating conditions. The capability to shut the plant down and maintain it in a cold shutdown condition has been evaluated. This evaluation concluded that sufficient separation and redundancy exists such that if the resultant spray were to disable equipment and a single active equipment were to occur, the affected Unit would still be able to achieve and maintain cold shutdown.

APPENDIX K

INTERIM OFFSITE POWER ARRANGEMENT

Because of transmission right-of-way problems during construction of the station, a temporary, interim supply of 138 KV had to be brought into LaSalle during construction of the station. This was in lieu of one of the dual, 345 KV lines that would permanently tie into the Edison grid from each generator unit at LaSalle. A description of this interim configuration was provided in Appendix K of the FSAR. Section 8.2 of the UFSAR describes the offsite power system currently connected to the LaSalle station.

The connection between the 345 kV and 138 kV systems at LaSalle was made permanent after the second pair of 345 kV lines was installed to support the 138 kV transmission system.

APPENDIX L

REQUIREMENTS RESULTING FROM TMI-2 ACCIDENT

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L.0 INTRODUCTION

The full text of CECo responses to NUREG-0737 action items for LaSalle County Station are included in Appendix L of the FSAR. Where applicable, the effects of evaluations, analyses, or modifications completed in response to specific action items have been incorporated in the text of the UFSAR. This appendix lists all action items and provides a brief discussion of the current status of each item along with references to UFSAR sections and Technical Specification sections where the effects of applicable responses have been included.

L.1 Shift Technical Advisor (I.A.1.1)

All NRC commitments on this topic have been met. Responsibilities and qualifications of the shift technical advisor are given in the Technical Specifications.

L.2 Shift Supervisor Administrative Duties (I.A.1.2)

All NRC commitments on this topic have been met. Responsibilities of the shift supervisor (Shift Manager at LaSalle County Station) are delineated in Section 13.1.2.2 of the UFSAR and the Technical Specifications.

L.3 Shift Manning (I.A.1.3)

All NRC commitments on this topic have been met. The Technical Specifications reflect current NRC-approved LSCS requirements for shift manning and administrative procedures on working hours.

L.4 Immediate Upgrading of Operator and Senior Operator Training and Qualification (I.A.2.1)

All NRC commitments on this topic have been met. Additional information on training is provided in Section 13.2 of the UFSAR.

L.5 Administration of Training Programs for Licensed Operators (I.A.2.3)

All NRC commitments on this topic have been met. Additional information on the training program is provided in Section 13.2 of the UFSAR.

L.6 Scope and Criteria for Licensing Examinations (I.A.3.1)

All NRC commitments on this topic have been met. Additional information on the training program is provided in Section 13.2 of the UFSAR.

L.7 Evaluation of Organization and Management Improvements of Near-Term Operating License Applicants (I.B.1.2)

All NRC Commitments on this topic have been met. The current LSCS organization is reflected in the Quality Assurance Topical Report.

L.8 Short-Term Accident Analysis and Procedure Revision (I.C.1)

All NRC Commitments on this topic have been met. Emergency procedures are discussed in Section 13.5.2 of the UFSAR. Emergency planning is discussed in Section 13.3 of the UFSAR.

L.9 Shift Relief and Turnover Procedures (I.C.2)

All NRC Commitments on this topic have been met. Section 13.5.1.3.6 of the UFSAR provides procedures for log book usage and control.

L.10 Shift Supervisor Responsibilities (I.C.3)

All NRC Commitments on this topic have been met. Directives and procedures have been prepared and implemented to meet the requirements of this action item. Shift supervisor (Shift Manager at LaSalle County Station) responsibilities are discussed in Sections 13.1.2.2 and 13.5.1.3.1 of the UFSAR.

L.11 Control Room Access (I.C.4)

All NRC Commitments on this topic have been met. Procedures have been prepared and implemented.

L.12 Procedures for Feedback of Operating Experience to Plant Staff (I.C.5)

All NRC Commitments on this topic have been met. The site follows procedures to ensure operating experience information continues to be appropriately reviewed and disseminated to plant staff.

L.13 Guidance on Procedures for Verifying Correct Performance of Operating Activities (I.C.6)

All NRC Commitments on this topic have been met. Equipment control procedures are described in Section 13.5.1.3.3 of the UFSAR.

L.14 NSSS Vendor Review of Procedures (I.C.7)

Requirement has been satisfied. Procedures have been reviewed and accepted by General Electric.

L.15 Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applications (I.C.8)

All NRC commitments on this topic have been met. Emergency planning is discussed in Section 13.3 of the UFSAR. Emergency procedures are discussed in section 13.5.2 of the UFSAR.

L.16 Control Room Design (I.D.1)

All NRC commitments on this topic have been met.

L.17 Plant Safety Parameter Display Console (I.D.2)

All NRC commitments on this topic have been met. Design information is included in Section 7.8.2 of the UFSAR.

L.18 Training During Low-Power Testing (I.G.1)

All NRC commitments on this topic have been met. The low power training and testing programs provided sufficient operator training and accumulation of sufficient technical data during both normal and off-normal plant operation.

L.19 Reactor Coolant System Vents (II.B.1)

All NRC commitments on this topic have been met. The existing reactor coolant system vent pathways are adequate as originally designed.

The elimination of the Steam Condensing Mode of Residual Heat Removal (RHR) operation has eliminated the low flow rate mode of RHR during post-LOCA conditions. This action has eliminated the possibility of having non-condensible gasses come out of solution in the top of the RHR heat exchangers, thereby, eliminating the requirement to maintain an available vent path via the E12-F073 and E12-F074 valves. Refer to UFSAR Section 5.4.7.2.2.3.

L.20 Plant Shielding (II.B.2)

All NRC commitments on this topic have been met. The shielding design review results and associated modifications are described in Section 12.3.2.5 of the UFSAR.

L.21 Post-Accident Sampling (II.B.3)

All NRC commitments on this topic have been met. Descriptions of the LSCS high-radiation sampling system (HRSS) are presented in Section 11.5.5 of the UFSAR.

L.22 Training for Mitigating Core Damage (II.B.4)

All NRC commitments on this topic have been met. The training program is discussed in Section 13.2 of the UFSAR.

L.23 Relief and Safety Valve Test Requirements (II.D.1)

All NRC commitments on this topic have been met. Additional information is provided in Section 5.2.2.10 of the UFSAR.

L.24 Relief and Safety Valve Position Indication (II.D.3)

All NRC commitments on this topic have been met. Details of implementation plan are incorporated in Section 7.7.15.2.6 of the UFSAR.

L.25 Auxiliary Feedwater System Feasibility Evaluation (II.E.1.1)

Applicable to PWR designs only.

L.26 Auxiliary Feedwater Initiation and Indication (II.E.1.2)

Applicable to PWR design only.

L.27 Emergency Power for Pressurized Heaters (II.E.3.1)

Applicable to PWR designs only.

L.28 Containment Dedicated Penetrations (II.E.4.1)

All NRC commitments on this topic have been met. Further discussions of the combustible gas control system are provided in section 6.2.5 of the UFSAR. Containment penetrations are listed in Table 6.2-21 of the UFSAR.

L.29 Containment Isolation Dependability (II.E.4.2)

All NRC commitments on this topic have been met. The containment isolation systems are discussed in Section 6.2.4 of the UFSAR.

L.30 Additional Accident - Monitoring Instrumentation (II.F.1)

All NRC commitments on this topic have been met. Discussions are incorporated in Sections 11.5.2.2 and 7.5.2 of the UFSAR.

L.31 Inadequate Core Cooling Instruments (II.F.2)

All NRC commitments on this topic have been met. Based on the BWR Owners Group evaluation, no other water level instruments provide as accurate and ready indication of adequate cooling as those installed. Additionally, in-core thermocouples could not provide unambiguous indication of inadequate core cooling.

L.32 Emergency Power for Pressurized Equipment (II.G.1)

Applies to PWR designs only.

L.33 Assurance of Proper ESF Functioning (II.K.1.5)

All NRC commitments on this topic have been met. Valve position requirements, positive controls, and test and maintenance procedures associated with ESF systems were reviewed to assure proper ESF functioning.

L.34 Safety-Related System Operability Status (II.K.1.10)

All NRC commitments on this topic have been met. Prerequisites for removing safety-related equipment from service are controlled by the Technical Specifications.

L.35 Pressurizer Low-Level Coincident Signal Bistables (II.K.1.17)

Applies to PWR designs only.

L.36 Operator Training for Prompt Manual Trip (II.K.1.20)

Applies to PWR designs only.

L.37 Automatic Safety Grade Anticipatory Trip (II.K.1.21)

Applies to PWR designs only.

L.38 Proper Functioning of Heat Removal Systems (II.K.1.22)

All NRC commitments on this topic have been met. Manual actions can be taken to mitigate the consequences of a loss of feedwater; however no manual actions are required. Sufficient systems exist to automatically mitigate these consequences.

L.39 Reactor Vessel Level Instrumentation (II.K.1.23)

All NRC commitments on this topic have been met. Reactor vessel level instrumentation is discussed in Section 7.7.1 of the UFSAR.

L.40 Control of Auxiliary Feedwater Independent of the Integrated Control System (II.K.2.2)

Applies to PWR designs only.

L.41 Failure Mode Effects Analysis on the Integrated Control System (II.K.2.9)

Applies to PWR designs only.

L.42 Safety-Grade Anticipatory Reactor Trip (II.K.2.10)

Applies to PWR designs only.

L.43 Thermal Mechanical Report--Effect of High-Pressure Injection on Vessel Integrity for Small-Break Loss-of-Coolant Accident with no Auxiliary Feedwater (II.K.2.13)

Applies to PWR designs only.

L.44 Power Operated Pressurizer Relief Valves (II.K.2.14)

Applies to PWR designs only.

L.45 Effects of Slug Flow on Steam Generator Tubes (II.K.2.15)

Applies to PWR designs only.

L.46 Reactor Coolant Pump Seal Damage (II.K.2.16)

Applies to PWR designs only.

L.47 Potential for Voiding in the Reactor Coolant System During Transients (II.K.2.17)

Applies to PWR designs only.

L.48 Sequential Auxiliary Feedwater Flow Analysis (II.K.2.19)

Applies to PWR designs only.

L.49 Installation and Testing of Automatic Power-Operated Relief Valve Isolation System (II.K.3.1)

Applies to PWR designs only.

L.50 Report on Overall Safety Effect of Power-Operated Relief Valve Isolation System (II.K.3.2)

Applies to PWR designs only.

L.51 Failure of PORV or Safety Valve to Close (II.K.3.3)

All NRC commitments on this topic have been met. Equipment failures are reported in accordance with 10CFR50.72 and 10CFR50.73.

L.52 Automatic Trip of Reactor Coolant Pumps During Loss-of-Coolant Accident (II.K.3.5)

Applies to PWR designs only.

L.53 Evaluation of Power-Operated Relief Valve Opening Probability During Overpressure Transient (II.K.3.7)

Applies to PWR designs only.

L.54 Proportional Integral Derivative Controller Modification (II.K.3.9)

Applies to PWR designs only.

L.55 Proposed Anticipatory Trip Modification (II.K.3.11)

Applies to PWR designs only.

L.56 Power-Operated Valve Failure Rate (II.K.3.11)

Applies to PWR designs only.

L.57 Confirm Existence of Anticipatory Reactor Trip Upon Turbine Trip (II.K.3.12)

Applies to PWR designs only.

L.58 Separation of HPCI and RCIC System Initiation Levels (II.K.3.13)

All NRC commitments on this topic have been met. At LSCS both the HPCS and RCIC are initiated on low water level. Based on an evaluation performed by General Electric, the proposed separation of RCIC and HPCI initiation is unnecessary for safety considerations.

The RCIC system automatically shuts off on high reactor vessel level. In response to this item, the RCIC system logic was revised to allow RCIC to restart on subsequent low water level as discussed in Section 7.4.1.2.3.1 of the UFSAR.

L.59 Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC (II.K.3.15)

All NRC commitments on this topic have been met. No modification is required for HPCS. Modification of RCIC logic has been implemented and is incorporated in Section 5.4.6.2.3 of the UFSAR.

L.60 Reduction of Challenges and Failures of Relief Valves--Feasibility Study and System Modification (II.K.3.16)

All NRC commitments on this topic have been met. Feasibility study and evaluation were completed. No modifications are required for LSCS.

L.61 Report on Outages of ECCS (II.K.3.17)

All NRC commitments on this topic have been met. Reporting of plant operational history is performed per Technical Specification 5.6.

L.62 Modification of ADS Logic--Feasibility for Increased Diversity for Some Event Sequences (II.K.3.18)

All NRC commitments on this topic have been met. ADS logic is discussed in Sections 7.3.1.2.2 and 6.3.5.2 of the UFSAR.

L.63 Restart of Core Spray and LPCI System (II.K.3.21)

All NRC commitments on this topic have been met. Modification has been implemented. Section 7.3.1. of the UFSAR was revised to reflect the modification.

L.64 Automatic Switchover of RCIC System Suction--Verify Procedures and Modify Design (II.K.3.22)

All NRC commitments on this topic have been met. Modification has been implemented. Section 5.4.6.3 of the UFSAR was revised to reflect the modification.

L.65 Confirm Adequacy of Space Cooling for HPCI and RCIC Systems (II.K.3.24)

All NRC commitments on this topic have been met. An area cooling system that is designated as Engineering Safety Feature (ESF) cools each ECCS cubicle. ECCS area cooling is discussed in Section 9.4.5 of the UFSAR.

L.66 Effect of Loss of AC Power on Pump Seals (II.K.3.25)

All NRC commitments on this topic have been met. Total loss of recirculation pump seals due to a loss of cooling will only result in a minimal loss of water that can be made up by normal or emergency water level controls.

L.67 Provide Common Reference Level for Vessel Level Instrumentation (II.K.3.27)

All NRC commitments on this topic have been met. A common reference level is used.

L.68 Verify Qualification of Accumulators on ADS Valves (II.K.3.28)

All NRC commitments on this topic have been met. ADS logic is discussed in Section 7.3.1.2.2 of the UFSAR and the accumulators are discussed in Section 9.3.1.2.2 of the UFSAR.

L.69 Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K (II.K.3.30)

All NRC commitments on this topic have been met. Additional discussion of LOCA analysis is presented in section 6.3 of the UFSAR.

L.70 Plant-Specific Calculations to Show Compliance with 10 CFR 50.46 (II.K.3.31)

All NRC commitments on this topic have been met. Results of LSCS-specific small-break LOCA calculations are provided in Section 6.3.3.7 and Table 6.3-8 of the UFSAR.

L.71 Evaluation of Anticipated Transients with Single Failure to Verify No Fuel Failure (II.K.3.44)

All NRC commitments on this topic have been met. Detailed study and evaluation concluded that for anticipated transients combined with the worst single failure, the core remains covered.

L.72 Evaluation of Depressurization with Other than ADS (II.K.3.45)

All NRC commitments on this topic have been met. BWR Owners Group analysis determined that no appreciable improvement is gained by a slower depressurization rate.

L.73 Response to List of Concerns from ACRS Consultant (Michelson Concerns) (II.K.3.46)

All NRC commitments on this topic have been met.

L.74 Emergency Preparedness (III.A.1.1)

All NRC commitments on this topic have been met. The current revision of the Emergency Plan is discussed in Section 13.3 of the UFSAR.

L.75 Emergency Support Facilities (III.A.1.2)

All NRC commitments on this topic have been met. Technical Support Center (TSC), Operational Support Center, and Emergency Operations Facility have been established as described in the Emergency Plan.

Note: The TSC no longer has controlled humidity or access to steam humidifiers.

L.76 Improving Licensee Emergency Preparedness--Long Term (III.A.2)

All NRC commitments on this topic have been met. The current revision of the Emergency Plan is discussed in Section 13.3 of the UFSAR.

L.77 Primary Coolant Sources Outside Containment (III.D.1.1)

All NRC commitments on this topic have been met. Discussion of the Primary Coolant Sources Outside Containment Program is included in Section 5.5.2 of the Technical Specifications.

L.78 In-Plant Radiation Monitoring (III.D.3.3)

All NRC commitments on this topic have been met. Discussion of radiation monitoring is discussed in Section 12.5.3.3.1 of the UFSAR and Section 5.5.3 of the Technical Specifications.

L.79 Control Room Habitability (III.D.3.4)

All NRC commitments on this topic have been met. Discussion of Control Room Habitability is included in Sections 6.4 and 2.2.3 of the UFSAR.

APPENDIX M

ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

Appendix M has been incorporated
into Section 3.11 of the UFSAR.

APPENDIX N

INTERIM SECONDARY CONTAINMENT

During operation of Unit 1 while Unit 2 was still under construction, temporary structures and procedures were employed to prevent an uncontrolled release path to the environment. These temporary measures were necessary because the Unit 1 and Unit 2 reactor buildings have interconnecting air paths. These measures constituted an "interim" secondary containment. Current details of the design and performance of the secondary containment are presented in Section 6.2.3 of the UFSAR.

APPENDIX O

CONTROL OF HEAVY LOADS

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DRAWINGS CITED IN THIS APPENDIX

DRAWING	SUBJECT
SK-5A	General Arrangement Reactor Building Floor Plans Load Paths

APPENDIX O - CONTROL OF HEAVY LOADS

O.0 Introduction/Licensing Background

As a result of NRC staff's Generic Task A-26, "Control of Heavy Loads Near Spent Fuel, NUREG-0612, "Control of Heavy Loads at Nuclear Plants" was developed. Following the issuance of NUREG-0612, a letter dated December 22, 1980 (later identified as GL 80-113 and corrected by GL 81-07), was sent to all operating plants, applicants for operating licenses and holders of construction permits requesting that responses be prepared to indicate the degree of compliance with the guidelines of NUREG-0612. The responses were to be made in two stages; Phase I (Section 5.1.1 of NUREG-0612) and Phase II. The Phase I response (Reference 6) was to identify the load handling equipment within the scope of NUREG-0612 and to describe the associated general load handling operations such as safe load paths, procedures, operator training, special and general purpose lifting devices, the maintenance, testing and repair of equipment and the handling equipment specifications. The Phase II response (Reference 7) was intended to show that either single-failure proof handling equipment was not needed or that single-failure proof equipment had been provided.

GL 85-11 reviewed the Phase II responses and determined that the cost/benefit analysis for upgrading cranes to single-failure proof was not sufficient due to the benefits realized by the implementation of the Phase I requirements. In NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," dated April 11, 1996, the staff addressed concerns on specific instances of heavy load handling and requested that licensees provide information documenting their compliance with these guidelines and their licensing bases. To investigate the need for additional regulation or guidance to address the risk with these heavy load movements, the Office of Nuclear Regulatory Research (RES) accepted this concern as Generic Issue (GI) 186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants." RIS 2005-25 (Reference 3) was issued to clarify previous communications about the requirements of NUREG-0612.

As stated in Subsection 9.1.4.3.2, LaSalle Station is committed to Phase I of NUREG-0612. A Phase II response (Reference 7) was submitted to the NRC, but no commitments were made with respect to single-failure proof requirements. As stated in Subsection 9.1.4.1.1, however, the reactor building overhead crane (RBOC) is single-failure proof.

O.1 Safety Design Basis

The safety design basis for controlling heavy loads is to ensure that the risk associated with load-handling failures is acceptably low. This is ensured by the following:

a. Meeting the Phase I requirements of NUREG-0612, Section 5.1.1.

b. Use of a single failure proof crane and special lifting device that conforms to NUREG-0612 Section 5.1.1(4) for the reactor vessel head lift and single-failure proof equipment for dry cask storage lifts in the reactor building.

c. Address 10 CFR 50.65(a)(4) for heavy load lifts conducted during power operation, or during shutdown conditions.

O.2 Scope of Heavy Load Handling Systems

The NUREG-0612 Phase I submittal to the NRC (Reference 6) provides a list of the overhead handling systems that are within the scope of NUREG-0612 Phase I. This list has not been updated when load handling systems have been removed at LaSalle Station. The Phase I requirements of NUREG-0612 are applied (when applicable) to newer handling systems, which are not included with this list.

O.3 Control of Heavy Loads Program

LaSalle station is committed to Phase I of NUREG-0612, which consists of Sections 5.1.1(1) though 5.1.1(7). The Reactor Pressure Vessel Head (RPVH) is lifted with a single-failure proof crane and special lifting device that conforms to NUREG-0612 Section 5.1.1(4). Spent fuel casks are lifted with a single-failure proof crane and lifting devices.

O.3.1 Commitments to NUREG-0612 Phase I

O.3.1.1 Safe Load Paths

Safe load paths for the refueling floor are defined on drawing SK-5A. Deviations from the defined load paths should require written alternative procedures approved by the plant safety review committee per NUREG-0612. The entire reactor building refueling floor (with the exception of the fuel pool and open reactor cavity) is considered a safe load path zone (Reference 17). For heavy loads lifted in other parts of the plant, the safe load path is determined as required.

O.3.1.2 Load Handling Procedures

Procedures are in place to minimize the risk associated with load-handling failures. These procedures provide guidance on identification of required equipment; inspections and acceptance criteria required before movement of load; the steps and proper sequence to be followed in handling the load; defining the safe load path; and other special precautions.

O.3.1.3 Qualifications, Training, and Conduct of Crane Operators

Crane operators are qualified, trained, and conduct themselves in accordance with ANSI/ASME B30.2.

O.3.1.4 Special Lifting Devices

Special lifting devices are those devices that are designed specifically for handling a certain load or loads, such as the lifting rigs for the Reactor Vessel head or vessel internals, or the lifting device for the spent fuel cask. Special lifting devices are used when normal rigging is not adequate. Special lifting devices used at LaSalle Station for heavy loads satisfy the guidelines of ANSI N14.6-1978. Their designs are based on the combined maximum static and dynamic loads that could be imparted on the handling device based on characteristics of the crane which will be used, and the stresses are limited to those specified in Section 3.2.1.1 of ANSI N14.6-1978.

O.3.1.5 Lifting Devices that are not Specially Designed

Lifting devices that are “not specially designed” (i.e. slings) are installed and used in accordance with the guidelines of ANSI B30.9. Slings that are used are selected based on the sum of the static and maximum dynamic loads. Seismic loads are not considered when selecting slings.

O.3.1.6 Periodic Inspection and Testing of Cranes

All tests and inspections of equipment that is capable of handling heavy loads are performed in accordance with normal station procedures. These procedures are written in accordance with ANSI/ASME B30.2.

O.3.1.7 Design of Cranes

Per LaSalle Station regulatory commitments, cranes are not required to be designed single-failure proof. As stated in Subsection 9.1.4.1.1, the RBOC trolley meets the single-failure proof requirements of NOG-1-2004, NUREG-0554, and NUREG-0612, Appendix C. The RBOC bridge meets ASME NOG-1 criteria to the extent that the analysis follows ASME NOG-1 methodology for modeling, load combinations, and stress allowables. It also meets the single-failure proof requirements of NUREG-0554 and NUREG-0612, Appendix C. The RBOC was upgraded in order to move a spent fuel cask, rather than perform a load drop analysis.

O.3.2 Reactor Pressure Vessel Head (RPVH) Lifting Procedures

The lifting scheme consists of the RBOC and a special lifting device. As stated in Subsections O.3.1.7 and 9.1.4.2.3, the RBOC is designated as single-failure proof.

The special lifting device conforms to Phase I requirements as delineated in NUREG-0612 Section 5.1.1(4). In addition, the RPVH is kept as low as practical while it is being moved. The heavy load handling procedures discussed in Subsection O.3.1.2 are used for the RPVH lift.

O.3.3 Single Failure Proof Cranes for Spent Fuel Casks

See Subsection 9.1.4.2.3 for a description of the single-failure proof RBOC.

O.4 Safety Evaluation

LaSalle Station is committed to Phase I of NUREG-0612. As such, controls implemented by NUREG-0612 Phase I elements make the risk of a heavy load drop very unlikely. Because LaSalle is not committed to Phase II of NUREG-0612, load drop analyses or single-failure proof designs are not required. Special lifting devices are designed in accordance with ANSI N14.6-1978 with stresses limited to those specified in Section 3.2.1.1 of ANSI N14.6-1978. The majority of loads on the refuel floor are handled with a single-failure proof crane, which makes the risk associated with these lifts acceptably low. The risk associated with the movement of heavy loads is evaluated and controlled by station procedures.

The guidance in NEI 08-05 (Reference 5), which is enforced by EGM 07-006 (Reference 4), has also been followed to further reduce/assess the risk associated with the movement of heavy loads. Per the NEI document, the lifting devices used below the hook to make the reactor head lift are to meet the Phase I requirements as delineated in NUREG 0612, Section 5.1.1.(4). Finally, 10 CFR 50.65(a)(4) is followed by assessing the risk of heavy load lifts conducted during power operation, or during shutdown conditions.

O.5 References

1. NUREG-0612, Control of Heavy Loads at Nuclear Plants.
2. Generic Letter (GL) 85-11, Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants" NUREG-0612. (Generic Letter 85-11).
3. RIS 2005-25, Clarification of NRC Guidelines for Control of Heavy Loads.
4. Enforcement Guidance Memorandum (EGM) 07-006, Enforcement Discretion for Heavy Load Handling Activities.
5. NEI 08-05, Rev. 0, Industry Initiative on Control of Heavy Loads.
6. E. D. Swartz (CE) to D. G. Eisenhut (NRC). Subject: Response to NUREG 0612, June 22, 1981.
7. E. D. Swartz (CE) to D. G. Eisenhut (NRC). Subject: Response to NUREG 0612, September 22, 1981.
8. J. S. Abel (CE) to D. G. Eisenhut (NRC). Subject: Control of Heavy Loads (NUREG 0612), May 15, 1981.
9. A. Schwencer (NRC) to L. O. DelGeorge (CE), Subject: Control of Heavy Loads, July 2, 1982.
10. E. D. Swartz (CE) to D. G. Eisenhut (NRC), Subject: NUREG 0612 Control of Heavy Loads, October 19, 1982.
11. C. W. Schroeder (CE) to A. Schwencer (NRC), Subject: NUREG 0612 Control of Heavy Loads, April 5, 1983.
12. A. Schwencer (NRC) to D. L. Farrar (CE), Subject: Control of Heavy Loads - Phase II - NUREG 0621, June 21, 1983.
13. E. D. Swartz (CE) to H. R. Denton (NRC), Subject: NUREG 0612 Control of Heavy Loads, June 22, 1983.
14. P. L. Barnes (CE) to H. R. Denton (NRC), Subject: Control of Heavy Loads - Phase II, September 16, 1983.
15. P. L. Barnes (CE) to H. R. Denton (NRC), Subject: Resolution of “Control of Heavy Loads at Nuclear Power Plants”, December 2, 1983.
16. A. Schwencer (NRC) to D. L. Farrar (CE), Subject: Control of heavy Loads, March 12, 1985.
17. Action Tracking Item 1119368-04, Design Engineering to provide a basis for the entire refuel floor being a safe load path.
18. NUREG-0519, Supplement 5, Appendix D.
19. Action Tracking Item 945440-07, Review your station compliance with NUREG 612.

20. NRC Bulletin 96-02, Movement of heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment.

21. J. B. Hosmer (CE) to US NRC, Subject: ComEd Response to NRC Bulletin 96-02, May 13, 1996.

22. R. M. Pulsifer (NRC) to O. D. Kingsley (CE), Subject: Completion of Licensing Action for NRC Bulletin 96-02, May 20, 1998.

APPENDIX P

UPDATED FINAL SAFETY ANALYSIS REPORT SUPPLEMENT (LICENSE RENEWAL)

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Note:
Section/Subsection identifiers begin with the letter “A” (instead of “P”) to maintain consistency with the License Renewal Application and the NRC Safety Evaluation Report.

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APPENDIX P

UPDATED FINAL SAFETY ANALYSIS REPORT SUPPLEMENT (LICENSE RENEWAL)

A.1.0 Introduction

The application for a renewed operating license is required by 10 CFR 54.21(d) to include a FSAR Supplement. Note that the NRC imposed a requirement within the renewed operating licenses (i.e., license condition) requiring that changes to the UFSAR supplement (i.e., Appendix P) be evaluated in accordance with 10 CFR 50.59. This appendix, which includes the following sections, comprises the FSAR supplement:

- Section A.1.1 contains a listing of the aging management programs that correspond to NUREG-1801 Chapter XI programs, including the status of the programs at the time the License Renewal Application was submitted.
- Section A.1.2 contains a listing of the plant-specific aging management programs, including the status of the programs at the time the License Renewal Application was submitted.
- Section A.1.3 contains a listing of aging management programs that correspond to NUREG-1801 Chapter X programs associated with Time-Limited Aging Analyses, including the status of the programs at the time the License Renewal Application was submitted.
- Section A.1.4 contains a listing of the Time-Limited Aging Analyses summaries (TLAAs).
- Section A.1.5 contains a discussion of the Quality Assurance Program and Administrative Controls.
- Section A.1.6 contains a discussion of the Operating Experience.
- Section A.2.0 contains a summarized description of the aging management programs.
- Section A.2.1 contains a summarized description of the NUREG-1801 Chapter XI programs for managing the effects of aging.
- Section A.2.2 contains a summarized description of the plant-specific programs for managing the effects of aging.
- Section A.3.0 contains a summarized description of the NUREG-1801 Chapter X programs that support the TLAAs.

- Section A.4.0 contains a summarized description of the TLAAs applicable to the period of extended operation.
- Section A.5.0 contains the License Renewal Commitment List.

The integrated plant assessment for license renewal identified new and existing aging management programs necessary to provide reasonable assurance that systems, structures, and components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis (CLB) for the period of extended operation. The period of extended operation is defined as beginning at the unit's current operating license expiration date and extending 20 years beyond that point.

A.1.0.1 Commitment Implementation Schedule Information

LRA Appendix A, Section A.5.0, License Renewal Commitment List contains the specific implementation schedule requirements for each commitment.

Consistent with the License Renewal Commitment list, when used throughout Appendix A, the phrase "prior to the period of extended operation" means that:

- Implementation of new aging management programs and enhancements to existing aging management programs will be completed no later than six months prior to the respective period of extended operation (PEO) for each LaSalle County Station unit; and
- Inspection or testing activities identified for completion prior to the PEO will be completed either:
    - No later than six months prior to the respective PEO for each LaSalle County Station unit, or
    - Prior to the end of the last refueling outage before the PEO for each respective unit,

  whichever occurs later.

A.1.1 NUREG-1801 Chapter XI Aging Management Programs

The NUREG-1801 Chapter XI Aging Management Programs (AMPs) are described in the following sections. The AMPs are either consistent with generally accepted industry methods as discussed in NUREG-1801 or require enhancements.

The following list reflects the status of these programs at the time of the License Renewal Application (LRA) submittal. Commitments for program additions and enhancements are identified in the Appendix P, Section A.5.0, License Renewal Commitment List.

1. ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (Section A.2.1.1) [Existing]

2. Water Chemistry (Section A.2.1.2) [Existing]

3. Reactor Head Closure Stud Bolting (Section A.2.1.3) [Existing]

4. BWR Vessel ID Attachment Welds (Section A.2.1.4) [Existing]

5. BWR Feedwater Nozzle (Section A.2.1.5) [Existing]

6. BWR Control Rod Drive Return Line Nozzle (Section A.2.1.6) [Existing]

7. BWR Stress Corrosion Cracking (Section A.2.1.7) [Existing]

8. BWR Penetrations (Section A.2.1.8) [Existing]

9. BWR Vessel Internals (Section A.2.1.9) [Existing - Requires Enhancement]

10. Flow-Accelerated Corrosion (Section A.2.1.10) [Existing]

11. Bolting Integrity (Section A.2.1.11) [Existing - Requires Enhancement]

12. Open-Cycle Cooling Water System (Section A.2.1.12) [Existing - Requires Enhancement]

13. Closed Treated Water Systems (Section A.2.1.13) [Existing - Requires Enhancement]

14. Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (Section A.2.1.14) [Existing - Requires Enhancement]

15. Compressed Air Monitoring (Section A.2.1.15) [Existing - Requires Enhancement]

16. Fire Protection (Section A.2.1.16) [Existing - Requires Enhancement]

17. Fire Water System (Section A.2.1.17) [Existing - Requires Enhancement]
18. Aboveground Metallic Tanks (Section A.2.1.18) [Existing - Requires Enhancement]
19. Fuel Oil Chemistry (Section A.2.1.19) [Existing - Requires Enhancement]
20. Reactor Vessel Surveillance (Section A.2.1.20) [Existing - Requires Enhancement]
21. One-Time Inspection (Section A.2.1.21) [New]
22. Selective Leaching (Section A.2.1.22) [New]
23. Unit 1 One-time Inspection of ASME Code Class 1 Small-Bore Piping (Section A.2.1.23) [New]
24. External Surfaces Monitoring of Mechanical Components (Section A.2.1.24) [New]
25. Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (Section A.2.1.25) [New]
26. Lubricating Oil Analysis (Section A.2.1.26) [Existing]
27. Monitoring of Neutron-Absorbing Materials Other Than Boraflex (Section A.2.1.27) [Existing - Requires Enhancement]
28. Buried and Underground Piping (Section A.2.1.28) [Existing - Requires Enhancement]
29. ASME Section XI, Subsection IWE (Section A.2.1.29) [Existing - Requires Enhancement]
30. ASME Section XI, Subsection IWL (Section A.2.1.30) [Existing - Requires Enhancement]
31. ASME Section XI, Subsection IWF (Section A.2.1.31) [Existing - Requires Enhancement]
32. 10 CFR Part 50, Appendix J (Section A.2.1.32) [Existing]
33. Masonry Walls (Section A.2.1.33) [Existing - Requires Enhancement]
34. Structures Monitoring (Section A.2.1.34) [Existing - Requires Enhancement]

35. RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants (Section A.2.1.35) [Existing - Requires Enhancement]

36. Protective Coating Monitoring and Maintenance Program (Section A.2.1.36) [Existing]

37. Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section A.2.1.37) [New]

38. Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits (Section A.2.1.38) [New]

39. Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section A.2.1.39) [New]

40. Metal Enclosed Bus (Section A.2.1.40) [Existing - Requires Enhancement]

41. Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section A.2.1.41) [New]

A.1.2 Plant-Specific Aging Management Programs

The plant-specific aging management programs are described in the following sections. The following list reflects the status of these programs at the time of the License Renewal Application (LRA) submittal. Commitments for program additions and enhancements are identified in Appendix P, Section A.5.0, License Renewal Commitment List.

1. Service Level III and Service Level III Augmented Coatings Monitoring and Maintenance Program (Section A.2.2.1) [New]

2. Unit 2 Inspection of ASME Code Class 1 Small-Bore Piping Program (Section A.2.2.2) [New]

A.1.3 NUREG-1801 Chapter X Aging Management Programs

The NUREG-1801 Chapter X Aging Management Programs (AMP) associated with Time-Limited Aging Analyses are described in the following sections. The AMPs are either consistent with generally accepted industry methods as discussed in NUREG-1801 Chapter X or require enhancements. The following list reflects the status of these programs at the time of the License Renewal Application (LRA) submittal. Commitments for program additions and enhancements are identified in Appendix P, Section A.5.0, License Renewal Commitment List.

1. Fatigue Monitoring (Section A.3.1.1)
   [Existing - Requires Enhancement]

2. Concrete Containment Tendon Prestress (Section A.3.1.2)
   [Existing - Requires Enhancement]

3. Environmental Qualification (EQ) of Electric Components
   (Section A.3.1.3) [Existing]

A.1.4 Time-Limited Aging Analyses

Summaries of the Time-Limited Aging Analyses applicable to the period of extended operation are included in the following sections:

1. Reactor Vessel and Internals Neutron Embrittlement Analyses (Section A.4.2)

2. Neutron Fluence Analyses (Section A.4.2.1)

3. Upper-Shelf Energy Analyses (Section A.4.2.2)

4. Adjusted Reference Temperature Analyses (Section A.4.2.3)

5. Pressure – Temperature Limits (Section A.4.2.4)

6. Axial Weld Failure Probability Assessment Analyses (Section A.4.2.5)

7. Circumferential Weld Failure Probability Assessment Analyses (Section A.4.2.6)

8. Reactor Pressure Vessel Reflood Thermal Shock Analyses (Section A.4.2.7)

9. RPV Core Plate Rim Hold-Down Bolt Loss of Preload Analysis (Section A.4.2.8)

10. Jet Pump Riser Brace Clamp Loss of Preload Analysis (Section A.4.2.9)

11. Jet Pump Slip Joint Repair Clamp Loss of Preload Analysis (Section A.4.2.10)

12. Metal Fatigue Analyses (Section A.4.3)

13. ASME Section III, Class 1 Fatigue Analyses (Section A.4.3.1)

14. ASME Section III, Class 2 and 3 and ANSI B31.1 Allowable Stress Analyses (Section A.4.3.2)

15. Environmental Fatigue Analyses for RPV and Class 1 Piping (Section A.4.3.3)

16. Reactor Vessel Internals Fatigue Analyses (Section A.4.3.4)

17. High-Energy Line Break (HELB) Analyses Based Upon Fatigue (Section A.4.3.5)

18. Main Steam Relief Valve Discharge Piping Fatigue Analyses (Section A.4.3.6)

19. Environmental Qualification (EQ) of Electric Components (Section A.4.4)

20. Environmental Qualification (EQ) of Electric Components (Section A.4.4.1)

21. Concrete Containment Tendon Prestress Analyses (Section A.4.5)

22. Concrete Containment Tendon Prestress Analyses (Section A.4.5.1)

23. Primary Containment Fatigue Analyses (Section A.4.6)

24. Primary Containment Liner and Penetrations Fatigue Analyses (Section A.4.6.1)

25. Primary Containment Refueling Bellows Fatigue Analysis (Section A.4.6.2)

26. Primary Containment Downcomer Vents Fatigue Analysis (Section A.4.6.3)

27. Other Plant-Specific Analyses (Section A.4.7)

28. Reactor Building Crane Cyclic Loading Analysis (Section A.4.7.1)

29 Main Steam Line Flow Restrictors Erosion Analysis (Section A.4.7.2)

A.1.5 Quality Assurance Program and Administrative Controls

The Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Appendix A.2, "Quality Assurance for Aging Management Programs (Branch Technical Position IQMB-1)" of NUREG-1800. The Quality Assurance Program includes the elements of corrective action, confirmation process, and administrative controls, and is applicable to the safety-related and nonsafety-related systems, structures, and components (SSCs) that are subject to Aging Management Review (AMR). In many cases, existing activities were found adequate for managing aging effects during the period of extended operation.

A.1.6 Operating Experience

Operating experience from plant-specific and industry sources is captured and systematically reviewed on an ongoing basis in accordance with the quality assurance program, which meets the requirements of 10 CFR Part 50, Appendix B, and the operating experience program, which meets the requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements," Item I.C.5, "Procedures for Feedback of Operating Experience to Plant Staff." The operating experience program interfaces with and relies on active participation in the Institute of Nuclear Power Operations' operating experience program, as endorsed by the NRC. The Exelon fleet OPEX program that is implemented at LSCS conforms to the recommendations of LR-ISG-2011-05, "Ongoing Review of Operating Experience." In accordance with this program, all incoming operating experience items are screened to determine whether they may involve age-related degradation or aging management impacts. Items so identified are further evaluated and the AMPs are either enhanced or new AMPs are developed, as appropriate, when it is determined through these evaluations that the effects of aging may not be adequately managed. Training on age-related degradation and aging management is provided, commensurate with their role in the process, to those personnel responsible for implementing the AMPs and who may submit, screen, assign, evaluate, or otherwise process plant-specific and industry operating experience. Plant-specific operating experience associated with aging management and age-related degradation is reported to the industry in accordance with guidelines established in the operating experience program.

A.2.0 Aging Management Programs

A.2.1 NUREG-1801 Chapter XI Aging Management Programs

This section provides summaries of the NUREG-1801 Chapter XI programs credited for managing the effects of aging.

A.2.1.1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD aging management program is an existing condition monitoring program that consists of periodic volumetric, surface, and visual examinations of ASME Class 1, 2, and 3 components including welds, pump casings, valve bodies, integral attachments, and pressure-retaining bolting for assessment, identification of signs of degradation, and establishment of corrective actions. The examinations are implemented in accordance with 10 CFR 50.55a and ASME Code, Section XI Subsections IWB, IWC, and IWD. These activities include examinations, evaluations, and monitoring and trending of results to confirm that effects of cracking and loss of fracture toughness are managed effectively during the period of extended operation.

A.2.1.2 Water Chemistry

The Water Chemistry aging management program is an existing mitigative program whose activities consist of monitoring and control of water chemistry to manage the aging of components exposed to treated water. Major component types managed by the program include the reactor vessel, reactor internals, piping, piping elements and piping components, heat exchangers, and tanks. The Water Chemistry program keeps peak levels of various contaminants below system-specific limits based on the industry recognized guidelines of the Boiling Water Reactor Vessel and Internals Project (BWRVIP 190, Revision 1, Electric Power Research Institute - 3002002623) for the mitigation of loss of material, reduction of heat transfer and cracking aging effects. In addition, the Water Chemistry program is also credited for mitigating loss of material and cracking for components exposed to sodium pentaborate, steam and reactor coolant environments. Chemistry programs are used to control water chemistry for impurities that accelerate corrosion to mitigate aging effects on component surfaces.

A.2.1.3 Reactor Head Closure Stud Bolting

The Reactor Head Closure Stud Bolting aging management program is an existing condition monitoring and preventive program that manages reactor head closure studs and associated nuts, washers, bushings, and flange threads for cracking and loss of material. The program is implemented through station procedures based on the examination requirements specified in ASME Code, Section XI, Subsection IWB, Table IWB-2500-1 and preventive measures to mitigate cracking as delineated in NRC Regulatory Guide 1.65 and NUREG-1339, with the exception that stud bolting components having a measured yield strength greater than 150 ksi are used.

A.2.1.4 BWR Vessel ID Attachment Welds

The BWR Vessel ID Attachment Welds aging management program is an existing condition monitoring program that includes the inspection and evaluation recommendations within BWRVIP-48-A and the requirements of ASME Code, Section XI, Subsection IWB. The program is implemented through station procedures that provide for mitigation of cracking of reactor vessel internal components through management of reactor water chemistry and monitoring for cracking through in-vessel examinations of the reactor vessel internal attachment welds.

A.2.1.5 BWR Feedwater Nozzle

The BWR Feedwater Nozzle aging management program is an existing condition monitoring program that manages the effects of cracking in the feedwater nozzles by augmented inservice inspection (ISI) in accordance with the requirements of the ASME Code, Section XI, Subsection IWB, Table IWB-2500-1 and the recommendations provided within BWROG Licensing Topical Report, GE-NE-523-A71-0594-A. The program includes periodic ultrasonic inspection of critical regions of the reactor vessel feedwater nozzles.

A.2.1.6 BWR Control Rod Drive Return Line Nozzle

The BWR Control Rod Drive Return Line (CRDRL) Nozzle aging management program is an existing condition monitoring program that manages the effects of cracking in the CRDRL reactor pressure vessel nozzle. The CRDRL nozzle has been capped to mitigate thermal fatigue cracking on both units, and the CRD return flow was not rerouted to the reactor vessel. Therefore, augmented inspections in accordance with NUREG-0619 and Generic Letter 80-095 are not required. The program includes inservice inspection (ISI) examinations in accordance with ASME Code, Section XI, Subsection IWB, Table IWB-2500-1. Volumetric ultrasonic inspection is performed on the CRDRL nozzle including the nozzle-to-vessel weld, nozzle blend radius, and nozzle-to-cap welds. The CRDRL nozzle-to-cap weld examinations are performed at a frequency specified by the BWR Stress Corrosion Cracking (A.2.1.7) program that implements commitments from NRC Generic Letter 88-01 and BWRVIP-75-A. The nozzle, cap, and associated welds are included in the visual inspection (VT-2) during the reactor pressure test performed each refueling outage.

A.2.1.7 BWR Stress Corrosion Cracking

The BWR Stress Corrosion Cracking aging management program is an existing condition monitoring and mitigative program that manages intergranular stress corrosion cracking (IGSCC) in relevant piping and piping welds made of stainless steel and nickel based alloy by augmented inservice inspections, regardless of code classification, as delineated in NUREG-0313, Revision 2, and NRC Generic Letter 88-01 and its Supplement 1. The program includes preventive measures to mitigate IGSCC, and inspection and flaw evaluation to monitor IGSCC and its effects. The schedule and extent of the inspections are performed in accordance with the NRC staff-approved BWRVIP-75-A report.

A.2.1.8 BWR Penetrations

The BWR Penetrations aging management program is an existing condition monitoring program that manages the effects of cracking of reactor vessel instrumentation penetrations, CRD housing and incore-monitoring housing penetrations, and the SLC/Core Plate dP penetration exposed to reactor coolant through-water chemistry controls and inservice inspections. In addition to the requirements of ASME Code, Section XI, Subsection IWB, the BWR Penetrations program incorporates the inspection and evaluation recommendations of BWRVIP-49-A, "Instrument Penetration Inspection and Flaw Evaluation Guidelines," BWRVIP-47-A, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines," BWRVIP-27-A, “BWR Standby Liquid Control System/Core Plate dP Inspection and Flaw Evaluation Guidelines,” and the water chemistry recommendations described in the Water Chemistry (A.2.1.2) program.

A.2.1.9 BWR Vessel Internals

The BWR Vessel Internals aging management program is an existing condition monitoring and mitigative program that manages the effects of cracking, loss of material, and loss of fracture toughness of reactor pressure vessel internal components through condition monitoring activities that consist of examinations that are implemented through station procedures consistent with the recommendations of the BWRVIP guidelines and ASME Code, Section XI, Table IWB-2500-1. The program also addresses aging degradation of X-750 alloy that is used for BWR vessel internal components. The program also mitigates these aging effects by managing water chemistry per the Water Chemistry (A.2.1.2) program.

The program will include aging management of reactor internal components fabricated from Cast Austenitic Stainless Steel (CASS) for loss of fracture toughness due to thermal aging and neutron embrittlement.

The BWR Vessel Internals aging management program will be enhanced to:

1. Perform an assessment of the susceptibility of reactor vessel internal components fabricated from CASS to loss of fracture toughness due to thermal aging embrittlement. If material properties cannot be determined to perform the screening, they will be assumed susceptible to thermal aging for the purposes of determining program examination requirements.

2. Perform an assessment of the susceptibility of reactor vessel internal components fabricated from CASS to loss of fracture toughness due to neutron irradiation embrittlement.

3. Specify the required periodic inspection of CASS components determined to be susceptible to loss of fracture toughness due to thermal aging and neutron irradiation embrittlement. The initial inspections will be performed either prior to or within five years after entering the period of extended operation.

4. Install core plate wedges no later than six months prior to the period of extended operation, or before the end of the last refueling outage prior to the period of extended operation, whichever occurs later; or, submit an inspection plan for the core plate rim hold-down bolts with a supporting analysis for NRC approval at least 2 years prior to entering the period of extended operation.

These enhancements will be implemented prior to the period of extended operation.

A.2.1.10 Flow-Accelerated Corrosion

The Flow-Accelerated Corrosion (FAC) aging management program is an existing condition monitoring program based on EPRI guidelines in NSAC-202L-R4, “Recommendations for an Effective Flow Accelerated Corrosion Program.” The program provides guidance for prediction, detection, and monitoring of wall thinning in piping and components. Analytical evaluations and periodic examinations of locations that are most susceptible to wall thinning due to FAC are used to predict the amount of wall thinning. Program activities include analyses to determine critical locations, baseline inspections to determine the extent of thinning at these critical locations, and followup inspections to confirm the predictions. Repairs and replacements are performed as necessary.

The Flow-Accelerated Corrosion program also manages wall thinning caused by mechanisms other than FAC, such as cavitation, flashing, droplet impingement, and solid particle impingement, in situations where periodic monitoring is used in lieu of eliminating the cause of various erosion mechanisms.

A.2.1.11 Bolting Integrity

The Bolting Integrity aging management program is an existing condition monitoring program. The program manages loss of preload, cracking, and loss of material due to corrosion of closure bolting on pressure-retaining joints within the scope of license renewal. The Bolting Integrity program incorporates NRC and industry recommendations delineated in NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," EPRI NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants," and EPRI TR-104213, "Bolted Joint Maintenance & Applications Guide."

The program credits visual inspection of pressure-retaining bolted joints in ASME Class 1, 2, and 3 systems for leakage and age-related degradation during system pressure tests performed in accordance with ASME Section XI, Subsections IWB, IWC, and IWD. In addition, the Bolting Integrity program credits volumetric, surface, and visual inspections of ASME Class 1, 2, and 3 bolts, nuts, washers, and associated bolting components performed in accordance with ASME Section XI. The integrity of pressure-retaining bolted joints in non-ASME Class 1, 2, 3 and MC systems is monitored by detection of visible leakage, evidence of past leakage, or other age-related degradation during maintenance activities and walkdowns in plant areas that contain systems within scope of license renewal. Inspection activities of closure bolting on pressure-retaining joints within the scope of license renewal in submerged environments will be performed in conjunction with associated component maintenance activities.

The Bolting Integrity program is supplemented by ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (A.2.1.1) program for inspection of safety-related closure bolting on pressure-retaining joints. Inspection activities for closure bolting on pressure-retaining joints in buried and underground environments are performed by the Buried and Underground Piping (A.2.1.28) program when closure bolting on pressure-retaining joints are exposed by excavation.

The Primary Containment (MC) pressure bolting is managed as part of ASME Section XI, Subsection IWE (A.2.1.29) program. The ASME Section XI, Subsection IWF (A.2.1.31) program manages ASME Class 1, 2, 3 and MC piping and component supports bolting. Structural bolting, other than ASME Class 1, 2, 3, and MC piping and component supports is managed as part of the Structures Monitoring (A.2.1.34) program and R.G. 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants (A.2.1.35) program. Crane and hoist bolting is managed by the Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (A.2.1.14) program. The heating and

Mechanical Components (A.2.1.24) program. Reactor head closure bolting is managed by the Reactor Head Closure Stud Bolting (A.2.1.3) program. The above bolting is not included in the Bolting Integrity program.

The Bolting Integrity aging management program will be enhanced to:

1. Provide guidance to ensure proper specification of bolting material, lubricant and sealants, storage, and installation torque or tension to prevent or mitigate degradation and failure of closure bolting for pressure-retaining bolted joints.

2. Prohibit the use of lubricants containing molybdenum disulfide on pressure-retaining bolted joints.

3. Minimize the use of high strength bolting (actual measured yield strength equal to or greater than 150 ksi) for pressure-retaining bolted joints in portions of systems within the scope of the Bolting Integrity program. High strength bolting (regardless of code classification) will be monitored for cracking in accordance with ASME Section XI, Table IWB-2500-1, Examination Category B-G-1.

4. Perform inspection of submerged bolting for the emergency core cooling systems (ECCS) and reactor core isolation cooling (RCIC) system suction strainers in the suppression pool for loss of material and loss of preload during each ISI inspection interval.

5. Perform inspection of submerged bolting for the service water diver safety barriers and diesel fire pump suction screens for loss of material and loss of preload each refuel interval.

6. Perform inspection of submerged bolting for the Lake Screen House traveling screens framework for loss of material and loss of preload each refuel interval.

These enhancements will be implemented prior to the period of extended operation.

A.2.1.12 Open-Cycle Cooling Water System

The Open-Cycle Cooling Water system (OCCWS) aging management program is an existing preventive, mitigative, condition monitoring, and performance monitoring program based on the implementation of NRC GL 89-13, which includes (a) surveillance and control of bio-fouling; (b) tests to verify heat transfer; (c) routine inspection and maintenance program; (d) system walkdown inspection; and (e) review of maintenance, operating, and training practices and procedures. The OCCWS program applies to components constructed of various materials, including steel, stainless steel, cast iron, aluminum alloys, and copper alloys.

The OCCWS program manages heat exchangers, piping, piping elements, and piping components in safety-related and nonsafety-related raw water systems that are exposed to a raw water environment for loss of material and reduction of heat transfer. The guidelines of NRC Generic Letter 89-13 are implemented through LSCS GL 89-13 activities for heat exchangers and the LSCS raw water corrosion program for piping segments. System and component testing, visual inspections, non-destructive examination, and chemical injection are conducted to ensure that identified aging effects are managed such that system and component intended functions and integrity are maintained.

The OCCWS program includes those plant systems that transfer heat from safety-related systems, structures, and components to the ultimate heat sink as defined in GL 89-13. Periodic heat transfer testing, visual inspection, and cleaning of safety-related heat exchangers with a heat transfer intended function is performed in accordance with LSCS commitments to GL 89-13 to verify heat transfer capabilities. Additionally, safety-related piping segments are non-destructively examined to ensure that there is no loss of material which results in loss of intended function.

The Open-Cycle Cooling Water System aging management program will be enhanced to:

1. Perform a minimum of 10 microbiologically influenced corrosion (MIC) degradation inspections for aboveground piping in the Essential Cooling Water System every 24 months until the rate of MIC occurrences no longer meets the criteria for recurring internal corrosion. The selected inspection locations will be periodically reviewed to validate their relevance and usefulness and adjusted as appropriate. Evaluation of the inspection results will include; (1) a comparison to the nominal wall thickness or previous wall thickness measurements to determine rate of corrosion degradation; (2) a comparison to the design minimum allowable wall thickness to determine the acceptability of the component for continued use; and (3) a determination of reinspection interval. A portion of these inspection locations will be selected with process conditions similar (e. g. flow, temperature) to those in buried portions of the piping to provide sufficient understanding of the condition of the buried piping.

2. Perform a minimum of 10 MIC degradation inspections for in scope aboveground piping in the Nonessential Cooling Water System every 24 months until the rate of MIC occurrences no longer meets the criteria for recurring internal corrosion. The selected inspection locations will be periodically reviewed to validate their relevance and usefulness and adjusted as appropriate. Evaluation of the inspection results will include (1) a comparison to the nominal wall thickness or previous wall thickness measurements to determine rate of corrosion degradation; (2) a comparison to the design minimum allowable wall thickness to determine the acceptability of the component for continued use; and (3) a determination of reinspection interval. A portion of these inspection locations will be selected with process conditions similar (e. g. flow, temperature) to those in buried portions of the piping to provide sufficient understanding of the condition of the buried piping.

3. Select an inspection method that will provide indication of suitable wall thickness to perform inspections on a representative sample of buried piping to supplement the aboveground piping inspection locations.

4. Perform visual inspections of the interior surface of buried portions of the Essential Cooling Water System and Nonessential Cooling Water System whenever the piping internal surface is made accessible due to maintenance and repair activities.

These enhancements will be implemented prior to the period of extended operation.

A.2.1.13 Closed Treated Water Systems

The Closed Treated Water Systems program is an existing mitigative and condition monitoring program that manages the loss of material, cracking, and reduction of heat transfer in piping, piping components, piping elements, tanks, and heat exchangers exposed to a closed cycle cooling water environment. The Closed Treated Water Systems program includes (a) nitrite-based water treatment, including pH control and the use of corrosion inhibitors, to modify the chemical composition of the water such that the function of the equipment is maintained and such that the effects of corrosion are minimized; (b) chemical testing of the water to ensure that the water treatment program maintains the water chemistry within acceptable guidelines; and (c) inspections to determine the presence or extent of corrosion, stress corrosion cracking, or fouling. The inspections include existing visual inspections of the internal surface of the components performed whenever the system boundary is opened as well as new periodic inspections as described in the enhancement below. The Closed Treated Water Systems aging management program will be enhanced to:

1. Perform condition monitoring, including periodic visual inspections and non-destructive examinations, to verify the effectiveness of water chemistry control to mitigate aging effects. A representative sample of piping and components will be selected based on likelihood of corrosion, stress corrosion cracking, or fouling, and inspected at an interval not to exceed once in 10 years during the period of extended operation. The selection of components to be inspected will focus on locations which are most susceptible to age-related degradation, where practical.

2. Monitor and trend drywell penetration cooling coil outlet temperatures monthly to ensure that adequate cooling is being provided to the concrete adjacent to the drywell penetrations.

These enhancements will be implemented prior to the period of extended operation.

A.2.1.14 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems

The Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems aging management program is an existing condition monitoring program that manages the effects of loss of material on the bridge, bridge rails, bolting and trolley structural components for those cranes, hoists, and rigging beams that are within the scope of license renewal. The program also manages loss of preload of associated bolted connections. Procedures and controls implement the guidance on the control of overhead heavy load cranes specified in NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants.” The program utilizes periodic inspections as described in the ASME B30 series of standards for inspection, monitoring, and detection of aging effects.

The Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems aging management program will be enhanced to:

1. Provide additional guidance to include inspection of structural components, rails, and bolting for loss of material due to corrosion; rails for loss of material due to wear; and bolted connections for loss of preload.

This enhancement will be implemented prior to the period of extended operation.

A.2.1.15 Compressed Air Monitoring

The Compressed Air Monitoring aging management program is an existing condition monitoring program that manages loss of material on piping and components in the compressed air systems. The Compressed Air Monitoring program includes monitoring of air moisture content and contaminants such that specified limits are maintained, and inspection of components for indications of loss of material due to corrosion.

This program is based on the LSCS response to NRC GL 88-14, “Instrument Air Supply Problems;” and utilizes guidance and standards provided by ANSI/ISA-S7.3, “Quality Standard for Instrument Air,” INPO’s Significant Operating Experience Report (SOER) 88-01, “Instrument Air System Failures;” and ASME OM-S/G-1998, Part 17, “Performance Testing of Instrument Air Systems in Light-Water Reactor Power Plants.” The Compressed Air Monitoring program activities implement the moisture content and contaminant criteria of ANSI/ISA-S7.3 (incorporated into ISA-S7.0.1-1996). Program activities include air quality checks at various locations to ensure that dew point, particulates, and hydrocarbons are maintained within the specified limits, and inspections of the internal surfaces of select compressed air system components for signs of loss of material due to corrosion.

The Compressed Air Monitoring aging management program will be enhanced to:

1. Inspect the internal surfaces of system filters, compressors, and after-coolers for signs of corrosion and corrosion products.

2. Perform analysis and trending of air quality monitoring results and visual inspection results.

3. Document deficiencies which are identified during visual inspections of the internal surfaces of system components in the corrective action program.

These enhancements will be implemented prior to the period of extended operation.

A.2.1.16 Fire Protection

The Fire Protection aging management program is an existing condition monitoring and performance monitoring program that includes fire barrier visual inspections and low pressure carbon dioxide system visual inspections and functional tests to manage the identified aging effects. The fire barrier inspection program requires periodic visual inspection of fire barrier penetration seals, fire barrier walls, ceilings, floors, and other materials that perform a fire barrier function. Periodic visual inspection and functional testing of the fire rated doors and visual inspection of fire rated dampers is performed to ensure that their functionality is maintained. The program also includes visual inspections and periodic operability tests of the low pressure carbon dioxide fire suppression systems using the National Fire Protection Association Codes and Standards for guidance.

The Fire Protection aging management program will be enhanced to:

1. Perform periodic visual inspection of combustible liquid spill retaining curbs.

2. Perform periodic visual inspection for identification of corrosion that may lead to loss of material on the external surfaces of the low pressure carbon dioxide fire suppression systems.

3. Provide additional inspection guidance to identify aging effects as follows:

   a. Fire barrier walls, ceilings, and floors degradation such as spalling, cracking, and loss of material for concrete.

   b. Elastomeric fire barrier material degradation such as loss of material, shrinkage, separation from walls and components, increased hardness, and loss of strength.

4. Provide additional inspection guidance to identify degradation of fire barrier penetration seals for aging effects such as loss of material, cracking, increased hardness, shrinkage, and loss of strength.

These enhancements will be implemented prior to the period of extended operation.

A.2.1.17 Fire Water System

The Fire Water System aging management program is an existing condition monitoring, performance monitoring, and preventive program that manages loss of material due to corrosion, including MIC, fouling, and flow blockage. The program manages these aging effects through the use of system pressure monitoring, system header flushing, buried ring header flow testing, pump performance testing, hydrant full flow flushing and flow verification, sprinkler and deluge system flushing and flow testing as well as flow testing and visual inspections performed using the guidance of NFPA 25, 2011 Edition.

The program applies to water-based fire protection systems that consist of sprinklers, fittings, valves, hydrants, hose stations, standpipes, pumps, and aboveground and buried piping and components. The program manages aging of fire protection system components exposed to raw water. Aging of the external surfaces of buried fire main piping is managed as described in the Buried and Underground Piping program.

Testing or replacement of sprinklers that have been in place for 50 years is performed using the guidance of NFPA 25, 2011 Edition.

The water-based fire protection system is normally maintained at required operating pressure and is monitored such that loss of system pressure is immediately detected and corrective actions initiated.

The Fire Water System aging management program will be enhanced to:

1. Perform volumetric examinations at five locations on the carbon steel aboveground fire water piping susceptible to microbiologically induced corrosion (MIC) every year to identify loss of material. Additional locations will be examined if these volumetric examinations or plant operating experience identify significant degradation. For through-wall leaks and material loss greater than 50 percent of nominal wall, four additional locations will be examined. Where the identified material loss is 30 percent to 50 percent of nominal wall thickness and the calculated remaining life is less than two years, two additional locations will be examined.

2. Perform visual inspections, for loss of material and flow obstructions, of the accessible header piping and sparger external surfaces for the deluge systems located within filter plenums on a once per refueling cycle frequency. The visual inspection will include verification that the piping and spargers are in their proper position and that there are no obstructions to the desired spray patterns.

3. Perform internal visual inspections of sprinkler and deluge system piping to identify internal corrosion and obstructions to flow. If the presence of sufficient foreign organic or inorganic material to obstruct pipe or sprinklers is detected during pipe inspections, the material is removed and its source is determined and corrected where possible. Followup volumetric examinations will be performed if internal visual inspections detect age-related degradation in excess of what would be expected accounting for design, previous inspection experience, and inspection interval.

   The internal visual inspections will consist of the following:

   a. Wet pipe sprinkler systems – 50 percent of the wet pipe sprinkler systems in scope for license renewal will have visual internal inspections of piping performed every five years consistent with NFPA 25, 2011 Edition, Section 14.2.

   b. Dry pipe sprinkler systems - Dry pipe sprinkler systems in scope for license renewal will have visual internal inspections of piping performed every five years consistent with NFPA 25, 2011 Edition, Section 14.2.

   c. Deluge systems - Deluge systems in scope for license renewal, except for the charcoal filter deluge systems, will have visual internal inspections of piping performed every five years consistent with NFPA 25, 2011 Edition, Section 14.2.

      i. The in scope charcoal filter deluge systems will have visual internal inspections performed on one of the 11 systems every five years. If degraded conditions are identified, the inspections will be expanded to include all 11 charcoal filter systems every five years.

   d. Sprinkler and deluge systems that are normally dry but may be wetted as the result of testing or actuations will have additional tests and inspections on piping segments that cannot be drained or piping segments that allow water to collect.

      i. These additional inspections, if required, will be performed in each five-year interval beginning five years prior to the period of extended operation.

ii. This additional inspection consists of either a flow test or flush sufficient to detect potential flow blockage or a visual inspection of 100 percent of the internal surface of piping segments that cannot be drained or piping segments that allow water to collect.

iii. In addition, in each five-year interval of the period of extended operation, 20 percent of the length of piping segments that cannot be drained or piping segments that allow water to collect is subject to volumetric wall thickness inspections.

4. Perform obstruction evaluations when degraded conditions are identified by visual inspections, flow testing, or volumetric examinations. The obstruction evaluations will include an extent of condition determination, need for increased inspections, and followup examinations if internal visual inspections detect age-related degradation in excess of what would be expected accounting for design, previous inspection experience, and inspection interval.

5. Perform flow tests for hose stations at the hydraulically most limiting locations for each zone of the system on a five-year frequency to demonstrate the capability to provide the design pressure at required flow.

6. Perform annual air tests on deluge systems supporting charcoal filter units excluding the “A” and “B” Auxiliary Electric Equipment Room Supply Air Filter units and the High Radiation Sampling System Filter unit. Perform visual internal inspections of the excluded filter units deluge systems in the event blockage was found on any deluge system that could be generic in nature.

7. Perform visual inspections of all charcoal filter unit deluge nozzles for proper orientation and verification that the nozzles are not obstructed on a 24 month frequency.

8. Include inspection for water leakage and loss of fluid in the glass bulbs of sprinkler heads, when performing visual inspections of sprinkler systems.

9. Include in main drain test acceptance criteria, the monitoring of flowing pressures from test to test. If there is a ten percent reduction in full flow pressure when compared to previously performed tests, an issue report shall be generated in the corrective action program to determine the cause and correct if necessary.

10. Maintain yard loop flow testing at a two year frequency until such time that the restricted section of piping from the pump house to Node 515 is restored to normal flow conditions.

These enhancements will be implemented prior to the period of extended operation.

A.2.1.18 Aboveground Metallic Tanks

The Aboveground Metallic Tanks aging management program includes outdoor tanks sited on soil or concrete and indoor large volume tanks containing water designed with internal pressures approximating atmospheric pressure that are sited on concrete. The program is an existing condition monitoring program which will be enhanced to provide for management of loss of material and loss of sealing for metallic tanks within the scope of the program. The Unit 1 and Unit 2 cycled condensate storage tanks are the only tanks within the scope of this program. These tanks are fabricated from aluminum plates and are not coated or insulated, and are not susceptible to cracking. The program includes caulking at the tank interface with the tank foundation as a preventive measure to mitigate corrosion. Visual inspections are performed to monitor degradation of the tank surfaces and caulking (caulking inspections are supplemented with physical manipulation). Visual inspections of interior tank surfaces are performed to detect loss of material. The bottoms of the tanks are examined volumetrically. These inspections and examinations ensure that significant degradation is not occurring and that the intended function of the cycled condensate storage tanks is maintained during the period of extended operation.

The Aboveground Metallic Tanks aging management program will be enhanced to:

1. Perform a visual inspection of the tank shell, roof, and bottom interior surfaces for signs of loss of material on one of the cycled condensate storage tanks within five years prior to the period of extended operation. This inspection shall include both wetted and non-wetted surfaces and may be either direct visual inspection from inside the tanks or volumetric examination from outside the tank. A volumetric examination from outside the tank will include 25 percent of the tank surface area. Should the one-time inspection identify degradation, periodic inspections with an inspection frequency based on the rate of degradation will be established for both tanks.

2. Perform a visual inspection of the exterior surfaces of both cycled condensate storage tanks for loss of material each refueling interval.

3. Perform a volumetric examination of the tank bottom for both cycled condensate storage tanks for signs of loss of material whenever the tanks are drained. At a minimum, an inspection shall be performed within 10 years prior to the period of extended operation and subsequent inspections shall be performed in each 10-year period during the period of extended operation. The next inspection scope will include 100% of the accessible areas of each of the tank bottoms that are within 30 inches of the shell. Included in this scope are the patch plates that are directly exposed to the sand bed below. Additionally, 10 random locations of approximately one square foot each, outside of the 30 inch band, will be inspected. This inspection program will encompass approximately 20% of the tank bottom and will inspect all the susceptible areas which were found during the baseline inspections. Based on the results of this inspection, the scope will be reassessed for future tank bottom inspections, per the Corrective Action Program.

4. Perform an inspection of the caulking at the perimeter of the cycled condensate storage tank bases for signs of degradation each refueling interval (caulking inspections are supplemented with physical manipulation).

These enhancements will be implemented prior to the period of extended operation.

A.2.1.19 Fuel Oil Chemistry

The Fuel Oil Chemistry aging management program is an existing mitigative and condition monitoring program that includes activities which provide assurance that contaminants are maintained at acceptable levels in fuel oil for systems and components within the scope of license renewal. The Fuel Oil Chemistry program manages loss of material in piping, piping elements, piping components and tanks in a fuel oil environment. The fuel oil tanks within the scope of license renewal are maintained by monitoring and controlling fuel oil contaminants in accordance with the Technical Specifications, Technical Requirements Manual, and ASTM guidelines. Fuel oil sampling and analysis is performed in accordance with approved procedures for new fuel oil and stored fuel oil. Fuel oil tanks are periodically drained of accumulated water and sediment, cleaned, and internally inspected. These activities effectively manage the effects of aging by maintaining potentially harmful contaminants at low concentrations.

The Fuel Oil Chemistry aging management program will be enhanced to:

1. Perform periodic (quarterly) sampling and analysis for water and sediment content, particulate concentration, and the levels of microbiological organisms for diesel fuel storage tanks 0DO01T, 1DO01T, and 2DO01T.

2. Perform periodic (quarterly) sampling and analysis for water and sediment content, particulate concentration, and the levels of microbiological organisms for HPCS diesel fuel storage tanks 1DO02T and 2DO02T.

3. Perform periodic (quarterly) sampling and analysis for water and sediment content and the levels of microbiological organisms for diesel generator day tanks 0DO02T, 1DO05T, and 2DO05T.

4. Perform periodic (quarterly) sampling and analysis for water and sediment content and the levels of microbiological organisms for HPCS diesel day tanks 1DO04T and 2DO04T.

5. Perform periodic (quarterly) sampling and analysis for water and sediment content, particulate concentration, and the levels of microbiological organisms for diesel fire pump day tanks 0FP01TA and 0FP01TB.

6. Perform periodic internal inspections of diesel fire pump day tanks 0FP01TA and 0FP01TB at least once during the 10-year period prior to the period of extended operation, and at least once every 10 years during the period of extended operation. Each diesel fuel tank will be drained and cleaned, the internal surfaces visually inspected (if physically possible), and, if evidence of degradation is observed during inspections, or if visual inspection is not possible, these diesel fuel tanks will be volumetrically inspected.

7. Perform volumetric inspection of diesel fuel storage tanks 0DO01T, 1DO01T, and 2DO01T; HPCS diesel fuel storage tanks 1DO02T and 2DO02T; diesel generator day tanks 0DO02T, 1DO05T, and 2DO05T; and HPCS diesel day tanks 1DO04T and 2DO04T if evidence of degradation is observed during visual inspection, or if visual inspection is not possible.

8. Perform periodic (quarterly) trending of water and sediment content, particulate concentration, and the levels of microbiological organisms for all fuel oil tanks within the scope of the program.

These enhancements will be implemented prior to the period of extended operation.

A.2.1.20 Reactor Vessel Surveillance

The Reactor Vessel Surveillance aging management program is an existing condition monitoring program that manages the loss of fracture toughness due to neutron irradiation embrittlement of the reactor vessel beltline materials. The program meets the requirements of 10 CFR 50, Appendix H. The program manages the surveillance capsules in each unit and ensures that the specimen exposure, capsule withdrawal, sample testing, and capsule storage meet the requirements of

10 CFR 50, Appendix H and ASTM E-185. The program evaluates neutron embrittlement by projecting Upper-Shelf Energy (USE) for reactor materials and impact on Adjusted Reference Temperature for the development of pressure-temperature limit curves. Embrittlement evaluations are performed in accordance with Regulatory Guide 1.99, Revision 2. The Reactor Vessel Surveillance program is part of the BWRVIP Integrated Surveillance Program (ISP) described in BWRVIP-86 Revision 1-A and approved by the NRC staff. The schedule for removing surveillance capsules is in accordance the timetable specified in BWRVIP-86 Revision 1-A for the current license term and for the period of extended operation.

The program monitors plant operating conditions to ensure appropriate steps are taken if reactor vessel exposure conditions are altered; such as the review and updating of 60-year fluence projections to support upper-shelf energy calculations and pressure-temperature limit curves. The program also includes condition monitoring by removal and analysis of surveillance capsules as part of the BWRVIP ISP. These measures are effective in detecting the extent of embrittlement to prevent significant degradation of the reactor pressure vessel during the period of extended operation.

The Reactor Vessel Surveillance aging management program will be enhanced to:

1. Establish a maximum fluence limit of 6.25E+17 n/cm^2 (39.15 EFPY) for monitoring the limiting Unit 1 axial welds to ensure that the axial weld failure probability does not exceed 5.02E-06 per reactor year.

This enhancement will be implemented prior to the period of extended operation.

2. Complete a probabilistic axial weld failure analysis for Unit 1 that demonstrates the 60-year axial weld failure probability is no greater than 5.02E-06 per reactor year. Submit the analysis to the NRC for review and approval.

This enhancement will be implemented at least 3 years prior to the limiting axial welds reaching the fluence limit specified above.

A.2.1.21 One-Time Inspection

The One-Time Inspection aging management program is a new condition monitoring program that will be used to verify the system-wide effectiveness of the Water Chemistry (A.2.1.2) program, Fuel Oil Chemistry (A.2.1.19) program, and Lubricating Oil Analysis (A.2.1.26) program which are designed to prevent or minimize aging to the extent that it will not cause a loss of intended function during the period of extended operation. The program manages loss of material, cracking, and reduction of heat transfer in piping, piping components, piping elements, heat exchangers, and other components within the scope of license renewal. The program provides inspections focusing on locations that are isolated from the flow stream, that are stagnant, or that have low flow for extended periods and are susceptible to the gradual accumulation or concentration of agents that promote certain aging effects. The inspections will include a representative sample of the system population and will focus on the bounding or lead components most susceptible to aging due to time in service, and severity of operating conditions. The program either verifies that unacceptable degradation is not occurring or triggers additional actions that will assure the intended function of affected components will be maintained during the period of extended operation.

This new aging management program will be implemented prior to the period of extended operation. The one-time inspections will be performed within the 10 years prior to entering the period of extended operation.

A.2.1.22 Selective Leaching

The Selective Leaching aging management program is a new condition monitoring program that will include one-time visual inspections of a representative sample of susceptible components within the scope of license renewal. These one-time inspections will include visual examinations, coupled with either hardness measurement or other mechanical examination techniques such as destructive testing, scraping, or chipping, of selected components that may be susceptible to selective leaching. This is to determine whether loss of material due to selective leaching is occurring, and whether the process will affect the ability of the components to perform their intended function for the period of extended operation. Metallurgical evaluation may also be performed. If loss of material is identified, further evaluation of the extent of condition will be performed under the corrective action program, which may include an expansion of the sample size and locations. Components in the scope of this program include components constructed of gray cast iron or copper alloy with 15 percent or greater zinc, that are exposed to raw water, treated water, closed cycle cooling water, waste water, and soil environments.

This new aging management program will be implemented prior to the period of extended operation. One-time inspections will be conducted within the five years prior to entering the period of extended operation.

A.2.1.23 Unit 1 One-time Inspection of ASME Code Class 1 Small-Bore Piping

The Unit 1 One-time Inspection of ASME Code Class 1 Small-Bore Piping aging management program is a new condition monitoring program that will manage the aging effect of cracking in ASME Code Class 1 small-bore piping that is less than nominal pipe size (NPS) 4-inches, and greater than or equal to NPS 1-inch. The program implements one-time inspection of a sample of piping full penetration (butt) and partial penetration (socket) welds that are susceptible to cracking using volumetric examinations. The inspection sample size will include at least 3 percent of the population of program butt welds with a maximum of 10 program butt welds, and at least 3 percent of the population of program socket welds with a maximum of 10 program socket welds. This results in 4 butt weld inspections and 10 socket weld inspections. Inspection of socket welds will be performed by a volumetric examination technique demonstrated to be capable of detecting cracking. If such a volumetric examination technique is not available by the time of the inspections, the examination method will be by destructive examination. If destructive examination is used, then each weld receiving a destructive examination can be credited as equivalent to having volumetrically examined two welds. Inspections required by the program will augment ASME Code, Section XI requirements.

Only two instances of cracking of ASME Code Class 1 small-bore piping at LSCS Unit 1 occurred during the first year of operation and were corrected by a design change. Therefore, this one-time inspection program is applicable and adequate to manage this aging effect during the period of extended operation. A plant-specific periodic inspection program will be implemented if evidence of cracking caused by IGSCC or fatigue is revealed in ASME Class 1 small-bore piping.

This new aging management program will be implemented prior to the period of extended operation. One-time inspections will be performed within the six years prior to entering the period of extended operation.

A.2.1.24 External Surfaces Monitoring of Mechanical Components

The External Surfaces Monitoring of Mechanical Components aging management program is a new condition monitoring program that directs visual inspections of external surfaces of components be performed during system inspections and walkdowns. The program consists of periodic visual inspection of metallic and elastomeric components such as piping, piping components, ducting, and other components within the scope of license renewal. The program manages aging effects of metallic and elastomeric materials through visual inspection of external surfaces for evidence of loss of material, cracking, and changes in material properties. When appropriate for the component and material, visual inspections are supplemented by physical manipulation to detect hardening and loss of strength of elastomers.

The External Surfaces Monitoring of Mechanical Components program includes visual inspection of the metallic jacketing on thermal insulation to ensure that the jacketing is performing its function to protect the insulation from damage, such as in-leakage of moisture, that could reduce the thermal resistance of the insulation.

Inspections are performed at a frequency not to exceed one refueling cycle. This frequency accommodates inspections of components that may be in locations that are normally only accessible during outages. Surfaces that are not readily visible during plant operations and refueling outages are inspected when they are made accessible and at such intervals that would ensure the components' intended functions are maintained.

A sample of outdoor component surfaces that are insulated and a sample of indoor insulated components exposed to condensation (due to the in scope component being operated below the dew point), are periodically inspected, under the insulation, every 10 years during the period of extended operation. Inspections subsequent to the initial inspection will consist of examination of the exterior surface of the insulation for indications of damage to the jacketing or protective outer layer of the insulation if the initial inspection verifies no loss of material beyond that which could have been present during initial construction. If the external visual inspections of the insulation reveal damage to the exterior surface of the insulation or if there is evidence of water intrusion through the insulation, then periodic inspections under insulation to detect corrosion under insulation will continue. Removal of tightly-adhering insulation that is impermeable to moisture is not required unless there is evidence of damage to the moisture barrier.

The external surfaces of components that are buried are inspected via the Buried and Underground Piping (A.2.1.28) program. The external surface of aboveground tanks is inspected via the Aboveground Metallic Tanks (A.2.1.18) program.

This new aging management program will be implemented prior to the period of extended operation.

A.2.1.25 Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components aging management program is a new condition monitoring program that will consist of inspections of the internal surfaces of metallic and elastomeric components such as piping, piping components and piping elements, ducting components, tanks, heat exchanger components, elastomers, and other components that are exposed to environments of condensation, diesel exhaust, and waste water. These internal inspections will be performed during the periodic system and component surveillances or during the performance of maintenance activities when the surfaces are made accessible for visual inspection. At a minimum, in each 10-year period during the period of extended operation, a representative sample of 20 percent of the population (defined as components having the same combination of material, environment, and aging effect) or a maximum of 25 components per population will be inspected. Where practical, the inspections will focus on the bounding or lead components most susceptible to aging because of time in service and severity of operating conditions. Opportunistic inspections will continue in each period even after meeting the sampling limit.

The program will manage the aging effects of loss of material, reduction of heat transfer, flow blockage, and cracking for metallic components. The program will also manage the aging effects of loss of material, hardening and loss of strength, and change in material properties for elastomeric components. The program will include visual inspections to ensure that existing environmental conditions are not causing material degradation or flow blockage that could result in a loss of the component's intended function. For example, the program verifies that plugging of the suppression pool or drywell spray nozzles due to corrosion of upstream piping does not occur. For certain materials, such as elastomers, physical manipulation to detect hardening or loss of strength will be used to supplement the visual examinations conducted under this program.

In addition, a review of LSCS operating experience has revealed instances of recurring internal corrosion in plant floor drain piping that is within the scope of the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program. This program will include periodic inspections on this population of carbon steel piping in the floor drain systems to ensure that recurring aging effects are adequately managed.

This new aging management program will be implemented prior to the period of extended operation.

A.2.1.26 Lubricating Oil Analysis

The Lubricating Oil Analysis aging management program is an existing condition monitoring program that provides monitoring of oil condition to manage loss of material and reduction of heat transfer in piping, piping components, piping elements, heat exchangers, and tanks within the scope of license renewal exposed to a lubricating oil environment. Sampling, analysis, and condition monitoring activities identify specific wear products and verify that the contamination levels (primarily water and particulates) and the physical properties of lubricating oil are maintained within acceptable limits to ensure that component intended functions are maintained.

A.2.1.27 Monitoring of Neutron-Absorbing Materials Other Than Boraflex

The Monitoring of Neutron-Absorbing Materials Other Than Boraflex aging management program is an existing condition monitoring program that includes periodic inspection and analysis of test coupons of the neutron-absorbing material in the spent fuel storage racks to determine if the neutron-absorbing capability of the material has degraded. This program ensures that a five percent sub-criticality margin in the spent fuel pool is maintained during the period of extended operation by monitoring for loss of material, changes in dimension, and loss of neutron-absorption capacity of the material.

The Monitoring of Neutron-Absorbing Materials Other Than Boraflex aging management program will be enhanced to:

1. Maintain the test coupon exposure such that it is bounding for the neutron-absorbing material in all spent fuel racks, by relocating the coupon tree to a different spent fuel rack cell location each cycle and by surrounding the coupons with a greater number of freshly discharged fuel assemblies than that of any other cell location.

This enhancement will be implemented prior to the period of extended operation.

A.2.1.28 Buried and Underground Piping

The Buried and Underground Piping aging management program is an existing preventive, mitigative, and condition monitoring program that manages the external surface aging effects of cracking and loss of material for buried and underground piping. The Buried and Underground Piping program includes preventive and mitigative techniques, such as external coatings for external corrosion control, the application of cathodic protection, and the quality of backfill utilized.

The program includes periodic directed inspections, with the number of inspections based on the quality of coating and backfill, and the availability and effectiveness of the cathodic protection system as determined in annual cathodic protection system surveys. Alternatively, soil corrosion probes may be used to verify acceptable loss of material rates.

Program inspections are conducted by qualified individuals. If coated piping and components show evidence of degraded coating and corrosion, the remaining wall thickness of the affected area will be measured and projected to the end of the period of extended operation. If the projected wall thickness does not meet acceptance criteria then the size of the directed inspections will be expanded.

External inspection of buried components will occur opportunistically when they are excavated for any reason.

The Buried and Underground Piping aging management program will be enhanced to:

1. Manage cracking for stainless steel piping, utilizing a method that has been demonstrated to be capable of detecting cracking, whenever coatings are removed and expose the base material.

2. Ensure all underground carbon steel Essential Cooling Water System and Nonessential Cooling Water System piping and components within the scope of license renewal are coated in accordance with Table 1 of NACE SP0169-2007.

3. Define acceptable coating conditions as coating exhibiting either no evidence of degradation, or, the type and extent of coating damage evaluated as insignificant by: (a) an individual possessing a NACE Coating Inspector Program Level 2 or 3 operator qualification, (b) an individual who has attended the Electric Power Research Institute (EPRI) Comprehensive Coatings Course and completed the EPRI Buried Pipe Condition Assessment and Repair Training Computer Based Training Course, or (c) a coatings specialist qualified in accordance with an ASTM standard endorsed in Regulatory Guide 1.54, Rev. 2, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants."

4. Perform inspection quantities of buried piping within the scope of license renewal in accordance with LR-ISG-2015-01 Table XI.M41-2, and LR-ISG-2015-01 Appendix B, "Evaluation and Technical Basis" section, items 4.a and 4.b, during each 10-year period, beginning 10 years prior to the period of extended operation. The number of inspections of buried piping will be based upon the as-found results of cathodic protection system availability and

effectiveness. The length of piping for each inspection will be based on the recommendations in LR-ISG-2015-01, Appendix B, “Evaluation and Technical Basis” section, item 4.c.

5. Perform direct visual inspections of underground Essential Cooling Water System and Nonessential Cooling Water System piping within the scope of license renewal during each 10-year period, beginning 10 years prior to the period of extended operation.

6. When measured pipe wall thickness, projected to the end of the period of extended operation, does not meet the minimum pipe wall thickness requirements, the number of inspections within the affected piping categories will be doubled or increased by five (5), whichever is smaller. If adverse indications are found in the expanded sample, an analysis will be conducted to determine the extent of condition and extent of cause. The size of the followup inspections will be determined based on the analysis. Timing of the additional inspections will be based on the severity of the identified degradation and the consequences of leakage. The additional inspections will be performed within the same 10-year inspection interval in which the original degradation was identified, or within 4-years after the end of the 10-year interval if the degradation was identified in the latter half of the 10-year interval. Expansion of sample size may be limited by the extent of piping subject to the observed degradation mechanism or if the piping system or portion of the system is replaced.

7. Use only the -850mV relative to a CSE (copper/copper sulfate reference electrode), instant off criterion specified in NACE SP0169-2007 for acceptance criteria for steel piping and determination of cathodic protection system effectiveness in performing cathodic protection surveys. Alternatively, soil corrosion probes may also be used to demonstrate cathodic protection effectiveness during the annual surveys. An upper limit of -1200mV for pipe-to-soil potential measurements of coated pipes will also be established, so as to preclude potential damage to coatings.

8. Conduct an extent of condition evaluation if observed coating damage caused by non-conforming backfill has been evaluated as significant. The extent of condition evaluation will be conducted to ensure that the as-left condition of backfill in the vicinity of the observed damage will not lead to further degradation.

These enhancements will be implemented prior to the period of extended operation.

A.2.1.29 ASME Section XI, Subsection IWE

The ASME Section XI, Subsection IWE aging management program is an existing condition monitoring program based on ASME Code and complies with the provisions of 10 CFR 50.55a. The program consists of periodic visual and volumetric examination of pressure-retaining components of steel and concrete containments for signs of degradation, assessment of damage, and corrective actions. The program includes aging management of surfaces and components such as the containment liner plate surfaces and components, including its integral attachments, drywell floor liner, downcomers and bracing, penetration sleeves and closures, vacuum breaker piping and valves, pressure-retaining bolting for containment closure, personnel airlock and equipment hatches, drywell head, and other pressure-retaining components for loss of material, loss of preload, loss of leak tightness, and fretting or lockup. LaSalle County Station primary containments are BWR Mark II concrete containments. High strength containment closure bolting susceptible to cracking is not used; therefore, surface examination to detect cracking is not applicable.

Examination methods include visual and volumetric testing as required by ASME Section XI, Subsection IWE, as approved in 10 CFR 50.55a. Observed conditions that have the potential for impacting an intended function are evaluated for acceptability in accordance with ASME requirements or corrected in accordance with corrective action program.

The ASME Section XI, Subsection IWE aging management program will be enhanced to:

1. Provide guidance for proper specification of bolting material, lubricant and sealants, and installation torque or tension to prevent or mitigate degradation and failure of structural bolting.

2. If leakage from the reactor cavity pool drain line welds exists, then perform ultrasonic thickness measurements on the Unit 2 drywell liner at 0 and 180 degrees for several feet below elevation 813. The inspections will begin in 2015 and a periodic inspection frequency will be established based on the inspection results.

These enhancements will be implemented prior to the period of extended operation.

A.2.1.30 ASME Section XI, Subsection IWL

The ASME Section XI, Subsection IWL aging management program is an existing condition monitoring program that consists of (a) periodic visual inspection of concrete surfaces for reinforced and prestressed concrete containments; and (b) periodic visual inspection and sample tendon testing of unbonded post-tensioning systems for prestressed concrete containments for signs of degradation, assessment of damage, and corrective actions, and testing of the tendon corrosion protection medium and free water. Measured tendon lift-off forces are compared to predicted tendon forces calculated in accordance with RG 1.35.1.

Reinforced concrete surfaces are inspected for material degradation, including loss of material, cracking, increase in porosity and permeability, and loss of bond. A sample of each tendon wire type (vertical and hoop) for the post-tensioning system is tested for loss of prestress. One tendon wire of each type is also examined for loss of material and subject to physical testing to determine yield strength, ultimate tensile strength, and elongation. The end anchorage for the unbonded post-tensioning system is inspected for loss of material.

This program is in accordance with ASME Section XI, Subsection IWL, as approved in 10 CFR 50.55a.

The ASME Section XI, Subsection IWL aging management program will be enhanced to:

1. Explicitly require that areas of concrete deterioration and distress be recorded in accordance with the guidance provided in ACI 349.3R.

2. Include quantitative acceptance criteria, based on the "Evaluation Criteria" provided in Chapter 5 of ACI 349.3R, that will be used to augment the qualitative assessment of the Responsible Engineer.

These enhancements will be implemented prior to the period of extended operation.

A.2.1.31 ASME Section XI, Subsection IWF

The ASME Section XI, Subsection IWF aging management program is an existing condition monitoring program that consists of periodic visual examinations of ASME Class 1, 2, 3, and MC piping and component supports and high-strength structural bolting for signs of degradation (such as loss of material, loss of mechanical function, and loss of preload), evaluation, and corrective actions. The program is implemented through corporate and station procedures, in accordance with the requirements of the ASME Code, Section XI, Subsection IWF, as approved in 10 CFR 50.55a. The monitoring methods are effective in detecting the applicable aging effects and the frequency of monitoring is adequate to prevent significant degradation.

The ASME Section XI, Subsection IWF aging management program will be enhanced to:

1. Provide guidance for proper specification of bolting material, storage, lubricant and sealants, and installation torque or tension to prevent or mitigate degradation and failure of structural bolting. Requirements for high strength bolts shall include the preventive actions for storage, lubricants, and stress corrosion cracking potential discussed in Section 2 of RCSC (Research Council on Structural Connections) publication "Specification for Structural Joints Using ASTM A325 or A490 Bolts." Lubricants that contain molybdenum disulfide (MoS2) shall not be applied to high strength bolts within the scope of license renewal. Bolting material with actual measured yield strength greater than or equal to 150 ksi or 1,034 MPa, in sizes greater than 1 inch nominal diameter shall not be used in supports for ASME Class 1, 2, and 3 piping and components or supports for MC components.

2. Provide guidance, regarding the selection of supports to be inspected on subsequent inspections, when a support is repaired in accordance with the corrective action program. The enhanced guidance will ensure that the supports inspected on subsequent inspections are representative of the general population.

These enhancements will be implemented prior to the period of extended operation.

A.2.1.32 10 CFR Part 50, Appendix J

The 10 CFR Part 50, Appendix J aging management program is an existing condition monitoring program that monitors leakage rates through the containment pressure boundary, including the containment liner and welds, penetrations, fittings, and other access openings, in order to detect degradation of the containment pressure boundary. Corrective actions are taken if leakage rates exceed acceptance criteria. The program provides for aging management of pressure boundary degradation due to aging effects such as loss of material, loss of leak tightness, or loss of preload in various systems penetrating containment. The program also detects loss of sealing due to degradation of gaskets and seals. Consistent with the current licensing basis, the containment leak rate tests are performed in accordance with the regulations and guidance provided in 10 CFR Part 50, Appendix J, Option B, Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," NEI 94-01, Revision 0 "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and ANSI/ANS 56.8, "Containment System Leakage Testing Requirements."

A.2.1.33 Masonry Walls

The Masonry Walls aging management program is an existing condition monitoring program that is implemented as part of the Structures Monitoring (A.2.1.34) program. Masonry wall condition monitoring is based on guidance provided in IE Bulletin 80-11, "Masonry Wall Design," and NRC Information Notice 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11," and is implemented through station procedures.

The Masonry Walls program manages the effects of loss of material and cracking of concrete masonry walls, and will inspect for separation, along with gaps between the supports and masonry walls. The program relies on periodic visual inspections on an interval not to exceed five years to monitor and maintain the condition of masonry walls within the scope of license renewal. Masonry walls that are considered fire barriers are also managed by the Fire Protection (A.2.1.16) program.

The Masonry Walls aging management program will be enhanced to:

1. Provide guidance for inspection of masonry walls for separation and gaps between the supports and masonry walls.

2. Require that personnel performing inspections and evaluations meet the qualifications described in ACI 349.3R.

These enhancements will be implemented prior to the period of extended operation.

A.2.1.34 Structures Monitoring

The Structures Monitoring aging management program is an existing condition monitoring program that was developed to implement the requirements of 10 CFR 50.65 and is based on NUMARC 93-01, Revision 2 “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” and Regulatory Guide 1.160, Revision 2 “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.” The program includes elements of the Masonry Walls (A.2.1.33) program and the RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants (A.2.1.35) program. The program consists of periodic inspection and monitoring the condition of structures and structural component supports, to ensure that aging degradation leading to loss of intended functions will be detected and the extent of degradation can be determined. The inspections are conducted on a frequency not to exceed five years. Groundwater will be periodically sampled on a five-year frequency, and tested to ensure the groundwater remains non-aggressive.

The Structures Monitoring aging management program will be enhanced to:

1. Add the following components and commodities:

    a. Pipe, electrical, and equipment component support members

    b. Pipe whip restraints and jet impingement shields

    c. Panels, racks, cabinets, and other enclosures

    d. Sliding surfaces

    e. Sumps

    f. Electrical cable trays and conduits

    g. Electrical duct banks

    h. Tube tracks

    i. Transmission tower (including takeoff towers) and foundation (including cycled condensate storage tank foundations)

    j. Penetration seals and sleeves

    k. Blowout panels

    l. Permanent drywell shielding

    m. Transformer foundation

    n. Bearing pads

    o. Compressible joints

    p. Hatches, plugs, handholes, and manholes

    q. Metal components (decking, vent stack, and miscellaneous steel)

    r. Building features – doors and seals, bird screens, louvers, windows, and siding

    s. Concrete curbs and anchors

    t. Turbine Building smoke and heat vent housings

2. Provide guidance for proper specification of bolting material, lubricant and sealants, and installation torque or tension to prevent or mitigate degradation and failure of structural bolting.

3. Revise storage requirements for high strength bolts to include recommendations of Research Council on Structural Connections (RCSC) Specification for Structural Joints Using High Strength Bolts, Section 2.0.

4. Require acceptance and evaluation of structural concrete using quantitative criteria based on Chapter 5 of ACI 349.3R.

5. Monitor raw water and ground water chemistry on a frequency not to exceed five years for pH, chlorides, and sulfates and verify that it remains non-aggressive, or evaluate results exceeding criteria to assess impact, if any, on below-grade concrete.

6. Monitor concrete for increase in porosity and permeability, inspection of accessible sliding surfaces for indication of significant loss of material due to wear or corrosion, debris, or dirt.

7. Evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas, and examine representative samples of the exposed portions of the below grade concrete when excavated for any reason.

8. Require that personnel performing inspections and evaluations meet the qualifications specified within ACI 349.3R with respect to knowledge of inservice inspection of concrete and visual acuity requirements.

9. Clarify that loose bolts and nuts and cracked high strength bolts are not acceptable unless accepted by engineering evaluations.

10. Inspect the fiberglass outer covering for the permanent drywell shielding for signs of rips and tears. If a rip or tear is found, repair or replace the permanent drywell shielding.

These enhancements will be implemented prior to the period of extended operation.

A.2.1.35 RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants

The RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants aging management program is an existing condition monitoring program implemented through the Structures Monitoring (A.2.1.34) program. The program consists of inspection and surveillance programs to provide management of aging for slopes, canals, intake structure, submerged portions of the Core Standby Cooling System Pond, and other water-control structure features associated with emergency cooling water systems or flood protection based on RG 1.127, Revision 1. The program also includes structural steel and structural bolting associated with water-control structures, steel piles, and sheeting required for the stability of embankments and channel slopes, and miscellaneous steel, and trash racks. There are no dams, wood piles, or sluice gates associated with the emergency cooling water systems or flood protection of the plant based on RG 1.127, Revision 1.

The Structures Monitoring program is an existing program which will be enhanced to include management of aging effects for water-control structures. The program monitors the condition of the Lake Screen House and the safety-related portions of the Cooling Lake. The Structures Monitoring program addresses age-related deterioration, degradation due to extreme environmental conditions, and the effects of natural phenomena that may affect the intended function of the water-control structures. The program is used to manage conditions such as, loss of material, loss of preload, cracking, increase in porosity and permeability, loss of strength, or loss of form. Elements of the program are designed to detect degradation and take corrective actions to prevent the loss of an intended function.

The RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants aging management program will be enhanced to:

1. Include monitoring of the following:

    a. Submerged Core Standby Cooling System Pond and Intake Flume (includes earthen walls, south flume concrete retaining wall, and north flume sheet piling retaining wall)

    b. Core Standby Cooling System outfall structure

    c. Bar racks and miscellaneous steel

    d. Shad net anchors

    e. Lake Screen House (includes service water tunnel)

2. Monitor raw water and ground water chemistry at least once every five years for pH, chlorides, and sulfates and verify that it remains non-aggressive, or evaluate results exceeding criteria to assess impact, if any, on buried or submerged concrete.

3. Provide guidance for proper specification of bolting material, lubricant and sealants, and installation torque or tension to prevent or mitigate degradation and failure of structural bolting, and preventative actions for storage of materials to prevent stress corrosion cracking.

4. Require acceptance and evaluation of structural concrete using quantitative criteria based on Chapter 5 of ACI 349.3R.

5. Require inspection of accessible in scope portions of the Cooling Lake and Lake Screen House immediately following the occurrence of significant natural phenomena, which includes intense local rainfalls and large floods.

6. Require:

    a. The evaluation of the acceptability of inaccessible areas when conditions exist in the accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas.

    b. Examination of the exposed portions of the below grade concrete when excavated for any reason.

These enhancements will be implemented prior to the period of extended operation.

A.2.1.36 Protective Coating Monitoring and Maintenance Program

The Protective Coating Monitoring and Maintenance Program is an existing mitigative and condition monitoring program that manages the effects of loss of coating integrity of Service Level I coatings inside primary containment. The Protective Coating Monitoring and Maintenance Program manages coating system selection, application, visual inspections, assessments, repairs, and maintenance of Service Level I protective coatings as defined in RG 1.54, Revision 1 or latest revision.

Service Level I coatings will prevent or minimize the loss of material due to corrosion but these coatings are not credited for managing the effects of corrosion for the carbon steel containment liners and components. This program ensures that the Service Level I coatings maintain adhesion so as to not affect the intended function of the emergency core cooling systems (ECCS) suction strainers.

The program also provides controls over the amount of unqualified coating which is defined as coating inside the primary containment that has not passed the required laboratory testing, including irradiation and simulated design basis accident (DBA) conditions. Unqualified coating may fail in a way to affect the intended function of the ECCS suction strainers. Therefore, the quantity of unqualified coating is controlled to ensure that the amount of unqualified coating in the primary containment is kept within acceptable design limits.

A.2.1.37 Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

The Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program is a new condition monitoring program that will be used to manage the effects of reduced insulation resistance of the insulation material for non-EQ cables and connections during the period of extended operation. Accessible cables and connections located in adverse localized environments will be visually inspected at least once every 10 years for cable jacket and connection insulation surface anomalies, such as embrittlement, discoloration, cracking, melting, swelling, or surface contamination that could indicate incipient conductor insulation aging degradation from temperature, radiation, or moisture. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service environment for the cable or connection.

This new program will be implemented prior to the period of extended operation. In addition, the first inspections will be completed prior to the period of extended operation.

A.2.1.38 Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits

The Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits aging management program is a new condition monitoring program that will be used to manage the effects of reduced insulation resistance of non-EQ cable and connection insulation in instrumentation circuits with sensitive, high-voltage, low-level current signals. The program applies to the in scope portions of the Neutron Monitoring System, the Process Radiation Monitoring System, and the Area Radiation Monitoring System. The circuits for these instruments are located in areas where the cables and connections could be exposed to adverse localized environments caused by temperature, radiation, or moisture. These adverse localized environments can result in reduced insulation resistance causing increases in leakage currents. Other instrument circuits in the Neutron Monitoring System, Process Radiation Monitoring System, and Area Radiation Monitoring System are not in scope of this aging management program either because they are managed by the Environmental Qualification (EQ) of Electric Components program; they do not perform a license renewal intended function; or they are not sensitive high voltage, low-level signal circuits.

Calibration testing will be performed for the in scope circuits when the cables are included as part of the calibration circuit. The calibration results will be reviewed to provide an indication of the existence of aging effects based on acceptance criteria for instrumentation circuit performance. Review of results obtained during normal calibration may detect severe aging degradation prior to the loss of the cable and connection intended function. A proven cable test (such as insulation resistance tests, time domain reflectometry tests, or other testing judged to be effective in determining cable system insulation condition) will be performed for the in scope circuits when the cables are not included as part of the calibration.

This new program will be implemented prior to the period of extended operation. In addition, the first review of calibration or surveillance results and cable test results will be completed prior to the period of extended operation. Cable test frequency will be based on engineering evaluation and will be performed at least once every 10 years. Calibration and assessment of results will be performed at least once every 10 years during the period of extended operation.

A.2.1.39 Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

The Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program is a new condition monitoring program that will be used to manage the effects of reduced insulation resistance of non-EQ, in scope, inaccessible power cables. For this program, power is defined as greater than or equal to 400 V. These inaccessible power cables may at times be exposed to significant moisture. Power cable exposure to significant moisture may cause reduced insulation resistance that can potentially lead to failure of the cable's insulation system.

The cables within the scope of this program will be tested using one or more proven tests for detecting reduced insulation resistance of the cable’s insulation system due to wetting or submergence, such as dielectric loss (dissipation factor or power factor), AC voltage withstand, partial discharge, step voltage, time domain reflectometry, insulation resistance and polarization index, line resonance analysis, or other testing that is state-of-the-art at the time the test is performed. The cables will be tested at least once every six years. More frequent testing may occur based on test results and operating experience. The first tests will be completed prior to the period of extended operation.

Periodic actions will be taken to prevent inaccessible cables from being exposed to significant moisture. Manholes associated with the cables included in this program will be inspected for water collection with subsequent corrective actions (e.g., water removal), as necessary. Prior to the period of extended operation, the frequency of inspections for accumulated water will be established and adjusted based on plant-specific operating experience with cable wetting or submergence, including water accumulation over time and event driven occurrences such as heavy rain or flooding. The inspection includes direct observation that cables are not wetted or submerged, that cables/splices and cable support structures are intact, and dewatering/drainage systems (i.e., sump pumps) and associated alarms operate properly. Operation of dewatering devices will be verified prior to any known or predicted heavy rain or flooding event. The first inspections will be completed prior to the period of extended operation. During the period of extended operation, the inspections will occur at least annually.

This new aging management program will be implemented prior to the period of extended operation.

A.2.1.40 Metal Enclosed Bus

The Metal Enclosed Bus aging management program is an existing condition monitoring program that will be enhanced to manage the identified aging effects of in scope metal enclosed bus during the period of extended operation. The internal portions of the accessible bus enclosure assemblies are inspected for cracks, corrosion, foreign debris, excessive dust buildup, and evidence of water intrusion. The bus insulation is visually inspected for signs of reduced insulation resistance, such as embrittlement, cracking, chipping, melting, discoloration, swelling, or surface contamination which may indicate overheating or aging degradation. The internal bus insulating supports are visually inspected for structural integrity and signs of cracks. External surfaces are visually inspected for loss of material due to general, pitting, and crevice corrosion. Enclosure assembly elastomers are visually inspected for surface cracking, crazing, scuffing, dimensional change, shrinkage, discoloration, hardening, and loss of strength. A sample of accessible bolted connections is inspected for increased resistance of connection by measuring the connection resistance using a micro-ohmmeter. The sample will be of 20 percent of the accessible metal enclosed bus bolted connection population with a maximum sample size of 25.

The inspections and resistance measurements will be performed at least once every 10 years for indications of aging degradation. The Metal Enclosed Bus aging management program will be enhanced prior to the period of extended operation.

The Metal Enclosed Bus program will be enhanced to:

1. Specify internal inspections will be performed for accessible non-segregated bus duct sections that are in scope for license renewal.

2. Clarify requirements for visual inspections of internal portions (bus enclosure assemblies); bus insulation; internal bus insulating supports; accessible gaskets, boots and sealants; and bus duct external surfaces.

3. Specify a sample size of 20 percent of the accessible bolted connection population, with a maximum sample size of 25, will be inspected for increased resistance of connection by either thermography or measuring the connection resistance using a micro ohmmeter.

4. Specify an inspection frequency of at least every 10 years.

These enhancements will be implemented prior to the period of extended operation.

A.2.1.41 Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program is a new condition monitoring program. The program will implement one-time testing of a representative sample of non-EQ electrical cable connections to ensure that either increased resistance of connection is not occurring or that the existing preventive maintenance program is effective such that a periodic inspection program is not required. This one-time program will provide additional confirmation to support industry operating experience that shows that electrical connections have not experienced a high degree of failures and that existing installation and maintenance practices are effective. This one-time program will also confirm that there are no aging effects requiring management during the period of extended operation. A representative sample of non-EQ electrical cable connections will be selected for one-time testing considering voltage level (medium and low voltage), circuit loading (high loading), connection type, and location (high temperature, high humidity, and vibration). The sample tested will be 20 percent of the population with a maximum sample size of 25 connections. The specific type of test performed will be a proven test for detecting increased resistance of connections, such as thermography, contact resistance measurement, or another appropriate test.

This new aging management program will be implemented prior to the period of extended operation. The one-time tests will be completed prior to the period of extended operation.

A.2.2 Plant-Specific Aging Management Programs

This section provides summaries of the plant-specific programs credited for managing the effects of aging.

A.2.2.1 Service Level III and Service Level III Augmented Coatings Monitoring and Maintenance Program

The Service Level III and Service Level III Augmented Coatings Monitoring and Maintenance Program is a new condition monitoring program that performs periodic visual inspections of internal coatings of in scope components. The Service Level III and Service Level III Augmented Coatings Monitoring and Maintenance Program manages the loss of coating integrity in heat exchangers, piping, piping components, piping elements, strainer bodies, and tanks.

The Service Level III and Service Level III Augmented Coatings Monitoring and Maintenance Program will include periodic visual inspections to verify the integrity of internal coatings designed to adhere to and protect the base metal. The maximum interval of subsequent coating inspections will comply with Table 4a of GALL Report AMP XI.M42 in draft LR-ISG-2013-01 dated January 6, 2014 (ADAMS Accession No. ML13262A442). The training and qualification of individuals performing coating inspections is conducted in accordance with ASTM international standards endorsed in RG 1.54.

Inspections are performed for signs of coating failures and precursors to coating failures including peeling, delamination, blistering, cracking, flaking, chipping, rusting, and mechanical damage. For coated surfaces determined to not meet acceptance criteria, physical testing is performed where physically possible (i.e., sufficient room to conduct testing) in conjunction with repair or replacement of the coating. A coatings specialist qualified to ASTM D-7108 will evaluate the results of coating inspections. Inspection results that do not satisfy established acceptance criteria are entered into the LSCS 10 CFR 50, Appendix B corrective action program. The corrective action program ensures that conditions adverse to quality are promptly corrected.

This new aging management program will be implemented prior to the period of extended operation. Baseline inspections will occur in the 10-year period prior to the period of extended operation.

A.2.2.2 Unit 2 Inspection of ASME Code Class 1 Small-Bore Piping Program

The Unit 2 Inspection of ASME Code Class 1 Small-Bore Piping Program is a new condition monitoring program that will manage the aging effect of cracking in ASME Code Class 1 small-bore piping that is less than nominal pipe size (NPS) 4-inches, and greater than or equal to NPS 1-inch. The program implements inspection of a sample of piping full penetration (butt) and partial penetration (socket) welds that are susceptible to cracking using volumetric or destructive examinations.

Unit 2 has experienced one failure of a Class 1 small-bore piping socket weld during its first 31 years of operation; therefore, a plant-specific aging management program is required. An extent of condition evaluation has identified other Unit 2 socket welds that may be susceptible to the age-related causal factors for that failure. Periodic inspections will be performed on those socket welds that are susceptible to the causal factors associated with the weld failure and one-time inspection will be performed on those welds that are not susceptible to those causes. Periodic inspections of 50 percent of the socket weld population that is susceptible to the causal factors associated with the failure (i.e., 5 welds) will be performed prior to the period of extended operation and every 10 years during the period of extended operation. One-time inspection will be performed on those welds that are not susceptible to the causal factors associated with the weld failure. The one-time inspection sample size will include at least 3 percent of the weld population or a maximum of 10 welds of each weld type (e.g. full penetration or socket weld). This results in one-time inspections of 3 butt welds and 10 socket welds.

Inspections required by the program will augment ASME Code, Section XI requirements. Inspection of socket welds will be performed by a volumetric examination technique demonstrated to be capable of detecting cracking or by destructive examination. If destructive examination is used, then each weld receiving a destructive examination can be credited as equivalent to having volumetrically examined two welds. The inspection of butt welds associated with Class 1 small-bore piping will be performed using volumetric examination techniques specified by ASME Code, Section XI. If any additional Class 1 small-bore welds fail or age-related cracking or degradation is identified by one-time or periodic inspection, the causes for the conditions will be evaluated using the corrective action program. The welds within systems that are susceptible to the causes associated with the condition will be included in the periodic inspection program.

This new aging management program will be implemented prior to the period of extended operation. The first sample of periodic inspections and the one-time inspections will be performed within the six years prior to entering the period of extended operation.

A.3.0 NUREG-1801 Chapter X Aging Management Programs

A.3.1 Evaluation of Chapter X Aging Management Programs

Aging management programs evaluated in Chapter X of NUREG-1801 are associated with Time-Limited Aging Analysis for metal fatigue of the reactor coolant pressure boundary, concrete containment tendon prestress, and environmental qualification (EQ) of electric components. These programs are evaluated in this section.

A.3.1.1 Fatigue Monitoring

The Fatigue Monitoring aging management program is an existing preventive program that manages fatigue damage of reactor coolant pressure boundary and other components subject to the reactor coolant, treated water, steam, and air-indoor uncontrolled environments.

The Fatigue Monitoring program includes monitoring and tracking of the critical thermal, pressure, and seismic transients that occur during plant operations to ensure that the cumulative usage factor (CUF) for each analyzed component does not exceed the design limit of 1.0 through the period of extended operation. The program includes monitoring of the transients specified in Technical Specification 5.5.5, Component Cyclic or Transient Limits, UFSAR Table 5.2-4, Plant Events, and UFSAR Table 3.9-24, Applicable Thermal Transients. These transients are further defined in the following General Electric drawings: Unit 1 Reactor Vessel Thermal Cycles, Unit 2 Reactor Vessel Thermal Cycles, and Reactor Vessel Nozzle Thermal Cycles.

The program requires comparison of the actual event parameters to the applicable design transient definitions to ensure the actual transients are bounded by the design transients. The program also includes counting of the operational transients to ensure that the cumulative number of occurrences of each transient type is maintained below the most limiting number of cycles used in the Class 1 fatigue analyses, which is the cycle limit for each transient type. Maintaining the cumulative cycle counts below the analyzed cycle limits ensures that the CUF does not exceed of 1.0. The fatigue analyses may be revised to account for increased numbers of cycles or increased transient severity as necessary to ensure that the CUF does not exceed 1.0.

Environmental fatigue analyses have been prepared for the limiting locations within the Unit 1 and Unit 2 reactor pressure vessels and for ASME Class 1 piping systems. In some cases, reduced numbers of cycles were analyzed, based on 60-year projections that justify the reduced numbers of cycles. The Fatigue Monitoring program will be enhanced by incorporating the most limiting numbers of cycles for each transient type used in the environmental fatigue analyses as administrative cycle limits. If an administrative cycle limit is approached, corrective actions are triggered, which may include revision of the affected environmental fatigue calculations and an expansion of the sample of locations evaluated for environmental fatigue, if warranted. The corrective action confirms that either the locations analyzed for environmental fatigue remain limiting or, if a new limiting location is identified, it is also analyzed for environmental fatigue.

The Fatigue Monitoring aging management program will be enhanced to:

1. Impose administrative transient cycle limits corresponding to the limiting numbers of cycles used in the environmental fatigue calculations.

2. Evaluate the impact of the reactor coolant environment on Class 1 components including valves and pumps if they are more limiting than those considered in NUREG/CR-6260.

These enhancements will be implemented prior to the period of extended operation.

A.3.1.2 Concrete Containment Tendon Prestress

The Concrete Containment Tendon Prestress aging management program is an existing condition monitoring program that is part of the containment inservice inspection program that is based on ASME Section XI, Subsection IWL criteria, as supplemented by the requirements of 10 CFR 50.55a(b)(2)(viii). The program monitors and manages the loss of tendon prestress in the concrete containment prestressing system for the period of extended operation. The prestressing tendons are used to impart compressive forces in the prestressed concrete containments to resist the internal pressure inside the containment that would be generated in the event of a LOCA. The prestressing forces generated by the tendons diminish over time due to losses in prestressing forces in the tendons and in the surrounding concrete. The regression and predicted lower limit analyses have been extrapolated through the end of the period of extended operation and the trend lines for each tendon group (vertical or horizontal (hoop) tendon types) have been shown to remain above the predicted lower limit and minimum required values for each tendon group. The program ensures that, during each inspection, the trend lines of the measured prestressing forces show that they meet the requirements of 10 CFR 50.55a(b)(2)(viii)(B). Measured forces and trend lines are compared to predicted lower limits, and minimum required values and corrective actions are taken if unacceptable results or trends are identified. The program also incorporates related plant-specific and industry operating experience.

The Concrete Containment Tendon Prestress aging management program will be enhanced as follows:

1. For each surveillance interval, trending lines will be updated through the period of extended operation as part of the regression analysis and compared to the predicted lower limit and minimum required values for each tendon group.

This enhancement will be implemented prior to the period of extended operation.

A.3.1.3 <u>Environmental Qualification (EQ) of Electric Components</u>

The Environmental Qualification (EQ) of Electric Components is an existing preventive program that manages the aging of electrical equipment within the scope of 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants." The program establishes, demonstrates, and documents the level of qualification, qualified configurations, maintenance, surveillance and replacements necessary to meet 10 CFR 50.49. A qualified life is determined for equipment within the scope of the program and appropriate actions such as replacement or refurbishment are taken prior to or at the end of the qualified life of the equipment so that the aging limit is not exceeded. The various aging effects addressed by this program are adequately managed so that the intended functions of components within the scope of 10 CFR 50.49 are maintained consistent with the current licensing basis during the period of extended operation.

A.4.0 Time-Limited Aging Analyses

A.4.1 Identification and Evaluation of Time-Limited Aging Analyses

As part of the application for a renewed license, 10 CFR 54.21(c) requires that an evaluation of Time-Limited Aging Analyses (TLAAs) for the period of extended operation be provided. The TLAAs identified and evaluated to meet these requirements are described below.

10 CFR 54.21(c)(2) also requires that the application for a renewed license include a list of plant-specific exemptions granted pursuant to 10 CFR 50.12 and in effect that are based upon TLAAs as defined in 10 CFR 54.3. It also requires an evaluation that justifies the continuation of these exemptions for the period of extended operation. No exemptions were identified that are based upon a TLAA. Therefore, no further evaluation is required.

A.4.2 Reactor Vessel and Internals Neutron Embrittlement Analyses

10 CFR 50.60 requires that all light-water reactors meet the fracture toughness, P-T limits, and material surveillance program requirements for the reactor coolant pressure boundary as set forth in 10 CFR 50 Appendices G and H. The current reactor pressure vessel embrittlement calculations for LSCS that evaluate reduction of fracture toughness of the Unit 1 and Unit 2 reactor pressure vessel beltline materials for 40 years are based upon a predicted end-of-license fluence applicable for 32 Effective Full Power Years (EFPY). These analyses were identified as TLAAs as defined in 10 CFR 54.21(c) and were re-evaluated for the increased neutron fluence associated with 60 years of operation as described in the subsections below.

This section also includes evaluations of the increased neutron fluence on reactor internal components, including potential loss of preload for the core plate rim hold-down bolts and loss of preload for repair clamps that have been installed on select jet pumps.

A.4.2.1 Neutron Fluence Analyses

High energy (>1 MeV) neutron fluence was projected for 60 years, or 54 Effective Full Power Years (EFPY), for the RPV beltline welds and shells using the Radiation Analysis Model Application (RAMA) Fluence Methodology. Use of this model was performed in accordance with NRC Regulatory Guide 1.190. The 54 EFPY fluence projections are used in the evaluations of the neutron embrittlement TLAAs.

The 54 EFPY fluence values have been projected for reactor vessel beltline materials, which include the reactor vessel plate materials, welds, and nozzle forgings that will be exposed to 1.0 E+17 neutrons/cm2 or more during 60 years of operation. Fluence projections have also been determined for specific reactor vessel internal components, both to evaluate fluence-based TLAAs and to determine when specified fluence threshold values may be exceeded that are used to invoke specific aging management requirements for these components, such as inspections.

A.4.2.2 Upper-Shelf Energy Analyses

10 CFR 50, Appendix G, Paragraph IV.A.1.a, requires that the reactor vessel beltline materials must maintain Charpy upper-shelf energy (USE) throughout the life of the vessel of no less than 50 ft-lb, unless it is demonstrated in a manner approved by the Director, Office of Nuclear Regulation, that lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

USE values were computed for all LSCS reactor vessel ferritic materials within the beltline that will be exposed to over 1.0 E+17 n/cm2 by the end of the period of extended operation (54 EFPY). The 54 EFPY USE values for the beltline materials were determined using methods consistent with Regulatory Guide 1.99, Revision 2.

The 54 EFPY USE values for LSCS beltline materials remain within the limits of 10 CFR 50, Appendix G requirements by having USE values of at least 50 ft-lb. Therefore, the analyses are projected through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.4.2.3 Adjusted Reference Temperature Analyses

The adjusted reference temperature (ART) of the limiting beltline material is used to adjust the beltline P-T limits to account for irradiation effects. The initial nil-ductility reference temperature, RT_{NDT}, is the temperature at which a un-irradiated ferritic steel changes in fracture characteristics from ductile to brittle behavior. RT_{NDT} is evaluated according to the procedures in the ASME Code, Section III. Neutron embrittlement increases the RT_{NDT} beyond its initial value.

10 CFR 50, Appendix G defines the fracture toughness requirements for the life of the vessel. The shift in the initial RT_{NDT} (ΔRT_{NDT}) is evaluated as the difference in the 30 ft-lb index temperatures from the average Charpy curves measured before and after irradiation. This increase (ΔRT_{NDT}) means that higher temperatures are required for the material to continue to act in a ductile manner. The ART is defined as Initial RT_{NDT} + ΔRT_{NDT} + Margin. The Margin term is defined in Regulatory Guide 1.99, Revision 2.

54 EFPY ART values were computed for LSCS beltline materials in accordance with Regulatory Guide 1.99, Revision 2. The 54 EFPY ART values of the limiting beltline materials for each unit remain below 200 degrees F, which is the RT_{NDT} limit.

The adjusted reference temperature analyses have been projected through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.4.2.4 Pressure – Temperature Limits

10 CFR 50, Appendix G requires that the reactor pressure vessel be maintained within established Pressure-Temperature (P-T) limits, particularly during heatup and cooldown operations. These limits specify the minimum allowable temperature as a function of reactor pressure. As the reactor pressure vessel is exposed to increased neutron irradiation, its fracture toughness is reduced. The P-T limits must be periodically adjusted to account for the anticipated reactor vessel fluence.

The current P-T limits are based upon 32 EFPY fluence projections, consistent with the nominal amount of power to be generated over 40 years of plant operation. The P-T limits satisfy the criteria of 10 CFR 54.3(a) and have been identified as TLAAs.

In accordance with NUREG-1800, Revision 2, Section 4.2.2.1.3, the P-T limits for the period of extended operation need not be submitted as part of the LRA since the P-T limits need to be updated through the 10 CFR 50.90 licensing process when necessary. It further states that for those plants that have approved pressure-temperature limit reports (PTLRs) the P-T limits for the period of extended operation will be updated at the appropriate time through the plant's administrative section of the TS and the plant's PTLR process. In either case, the 10 CFR 50.90 or the PTLR processes, whichever constitutes the current licensing basis at the time, will be used to ensure that the P-T limits for the period of extended operation are updated prior to expiration of the 32 EFPY P-T limit curves.

A.4.2.5 Axial Weld Failure Probability Assessment Analyses

The BWRVIP recommendations for inspection of reactor pressure vessel shell welds in BWRVIP-05 include examination of 100 percent of the axial welds and inspection of the circumferential welds only at the intersections of these welds with the axial welds. BWRVIP-05 contains generic analyses supporting a conclusion in the NRC Final Safety Evaluation Report (FSER) dated July 28, 1998, that the generic-plant axial weld failure probability is orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability and used this analysis to justify relief from inspection of the circumferential welds. The staff provided separate conditional failure probability assessments in the Supplement to the Final Safety Evaluation of the BWRVIP-05 Report, dated March 7, 2000. Since these NRC staff failure probability assessments are applicable to LSCS Units 1 and 2, they are identified as TLAAs requiring evaluation through the period of extended operation.

For LSCS Unit 1, the limiting axial weld mean RT_{NDT} value at 54 EFPY exceeds the mean RT_{NDT} value determined for the limiting Combustion Engineering (CE), Mod 2 variant, reactor vessel. Therefore, the associated axial weld failure probability for Unit 1 at 54 EFPY is not bounded by the conditional axial weld failure probability of 5.02E-06 determined for the CE reactor vessel. The effects of aging for the Unit 1 reactor vessel axial welds will be managed in accordance with 10 CFR 54.21(c)(1)(iii) by:

1. The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (A.2.1.1) program, which requires periodic examination of the axial welds in accordance with the requirements of ASME Section XI, Table IWB-2500-1, and

2. The Reactor Vessel Surveillance (A.2.1.20) program, which manages neutron embrittlement by monitoring neutron fluence and ensures that neutron embrittlement analyses are updated as necessary to evaluate bounding neutron fluence values for each unit. The Reactor Vessel Surveillance program is enhanced as follows: 1) prior to the period of extended operation, establish a maximum fluence limit of 6.25E+17 n/cm2 (39.15 EFPY) for monitoring the limiting Unit 1 axial welds to ensure that the axial weld failure probability does not exceed 5.02E-06 per reactor year; and 2) complete a probabilistic axial weld failure analysis for Unit 1 that demonstrates the 60-year axial weld failure probability is no greater than 5.02E-06 per reactor year. Submit the analysis to the NRC for review and approval at least 3 years prior to the limiting axial welds reaching the fluence limit specified above.

The combination of periodic axial weld examinations by the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program and management of neutron embrittlement by the enhanced Reactor Vessel Surveillance program ensures that the axial weld failure probability will not exceed 5.02E-06 and ensures that loss of fracture toughness due to neutron irradiation embrittlement will not affect the structural integrity of the Unit 1 reactor vessel through the period of extended operation.

For LSCS Unit 2, the limiting axial weld mean RT_{NDT} value at 54 EFPY is bounded by the mean RT_{NDT} value determined for the limiting axial weld in the Chicago Bridge and Iron (CB&I) reactor vessel evaluated in the supplement to the final safety evaluation for BWRVIP-05. Therefore, the NRC conditional axial weld failure probability of 2.73E-06 is bounding for LSCS Unit 2. Therefore, this analysis has been projected through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.4.2.6 Circumferential Weld Failure Probability Assessment Analyses

ASME Section XI governs the inspection of the reactor pressure vessel circumferential welds, as implemented by the LSCS Inservice Inspection Program. LSCS has received relief from examination of circumferential welds. The relief from inspection is based on assessment of the probability of failure of the limiting circumferential weld. This assessment is based on 32 EFPY fluence values associated with 40 years of operation and has therefore been identified as a TLAA requiring evaluation for the period of extended operation.

In order to perform the LSCS circumferential weld failure probability assessment for 60 years, 54 EFPY fluence values were projected for the limiting circumferential weld for each reactor vessel. Using the 54 EFPY fluence values, the LSCS Mean RT_{NDT} values were projected for each unit and compared to the NRC analytical results for 64 EFPY provided in the FSER to BWRVIP-05. Although the conditional failure probability has not been calculated for the LSCS units, the LSCS Mean RT_{NDT} values for the period of extended operation are significantly less than the NRC RT_{NDT} value used in determining the conditional failure probability. Therefore, the NRC conditional failure probability is bounding for LSCS Unit 1 and Unit 2, consistent with the requirements defined in GL 98-05.

Reapplication for relief from circumferential weld examination will be made under 10 CFR 50.55a(a)(3) in time for NRC review and approval prior to the period of extended operation. The plant-specific information described above demonstrates that at the end of the period of extended operation, the circumferential beltline weld materials meet the limiting conditional failure probability for circumferential welds specified in the SER of BWRVIP-05. These analyses will be managed in accordance with 10 CFR 54.21(c)(1)(iii) by requesting relief from circumferential weld inspection using the 10 CFR 50.55a process.

A.4.2.7 Reactor Pressure Vessel Reflood Thermal Shock Analyses

A generic fracture mechanics evaluation was performed to evaluate the effects of a postulated loss of coolant accident (LOCA) on the structural integrity of a BWR-6 reactor pressure vessel. The rupture of a main steam line was determined to bound all other LOCA events with respect to this evaluation. Several emergency core cooling systems are activated at different times after the LOCA and the vessel is flooded with cooling water. The vessel depressurization and the subsequent injection of cold water to reflood the reactor vessel produce a rapid reduction in temperature and high thermal stresses in the vessel. The analysis concluded that the reactor pressure vessel has a considerable margin to failure by brittle fracture even in the presence of postulated pre-existing flaws. This generic analysis envelopes LSCS and is based on BWR vessel material properties and cumulative fluence assumed for 40 years of operation. Therefore, this analysis has been identified as a TLAA requiring evaluation for the period of extended operation.

An updated 60-year fracture mechanics evaluation was performed for the reflood thermal shock event to evaluate the component with the limiting material properties from the LSCS Unit 1 and Unit 2 RPV beltline plates, axial welds, and circumferential welds, which bounds the remainder. The analysis also evaluated the limiting N6 RHR/LPCI nozzle. The analysis determined that during the period of extended operation, each RPV has sufficient toughness margin to prevent unacceptable flaw propagation due to thermal shock during reflooding after LOCA events.

The reactor pressure vessel reflood thermal shock analysis has been projected through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.4.2.8 RPV Core Plate Rim Hold-Down Bolt Loss of Preload Analysis

The RPV core plate is attached to the core support structure by stainless steel hold-down bolts that are preloaded during initial installation. These bolts are subject to stress relaxation (loss of preload) due to irradiation effects. An analysis was performed concluding that a reduction in preload as high as 19 percent over the 40-year life of the bolts is acceptable to meet design requirements. A subsequent re-evaluation determined that this maximum relaxation value of 19 percent is applicable to an average fluence value of 8.0 E+19 n/cm2 over the entire length of the bolt located at the azimuthal location with peak fluence. These analyses were identified as TLAAs.

In order to determine if these analyses will remain valid for 60 years, RAMA fluence projections were prepared for LSCS for 54 EFPY for the core plate rim bolt located at the azimuthal location with peak fluence. In order to determine the average fluence value along the length of the bolt, fluence projections were made at 30 discrete points along the length of the bolt at their centerline. These results were integrated and divided by the length of the bolt, resulting in an average fluence value of 3.60 E+19 n/cm2 for Unit 1, and 3.85 E+19 n/cm2 for Unit 2. This is well below the average value of 8.0 E+19 n/cm2 previously evaluated. Therefore, the analysis remains valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.2.9 Jet Pump Riser Brace Clamp Loss of Preload Analysis

Jet pump riser brace repair clamps have been installed on two jet pump risers inside the Unit 1 RPV to structurally replace the riser brace yoke-to-riser welds (RS-8 and RS-9) as a repair for crack indications in the welds. The structural evaluation for the clamp assumed a 46 percent loss of preload during 40 years of operation due to irradiation effects. The evaluated 46 percent load relaxation was based on an end-of-life fluence value of 3.2 E+20 n/cm2, assuming 40 years of operation from the time of clamp installation. This analysis was identified as a TLAA.

The 60-year neutron fluence at the riser brace clamps was projected to be 2.0 E+20 n/cm2 at 54 EFPY, which is less than the 3.2 E+20 n/cm2 fluence value assumed in determining the acceptable load relaxation in the design analysis for the clamp. Therefore, the design analysis remains valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.2.10 Jet Pump Slip Joint Repair Clamp Loss of Preload Analysis

Jet pump slip joint repair clamps have been designed and installed at LSCS Unit 1 to minimize vibration and wear of the jet pump assemblies. The design analysis for the clamp determined that loss of preload would result from neutron fluence during the design life of the clamps. The loss of preload due to fluence was evaluated to the end of the 40-year license period. This analysis was identified as a TLAA.

The neutron fluence at the jet pump clamps was projected to be 1.27 E+20 n/cm2 from the time of the first clamp installation in 2004 through the end of the period of extended operation. The limiting neutron fluence to maintain acceptable clamp preload is 1.17 E+20 n/cm2. Since the fluence is predicted to exceed the analyzed value prior to the end of the period of extended operation, the analysis must be updated or other corrective actions must be taken prior to exceeding the analyzed fluence value, potentially including repair or replacement of the clamps. License renewal commitment 47 will be utilized to manage this TLAA through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.4.3 Metal Fatigue Analyses

Metal fatigue was considered explicitly in the design process for pressure boundary components designed in accordance with ASME Section III, Class A or Class 1 requirements. Metal fatigue was evaluated implicitly for components designed in accordance with ASME Section III, Class 2 or 3 requirements or ANSI B31.1 requirements. Each of these fatigue analyses and evaluations are considered to be Time-Limited Aging Analyses requiring evaluation for the period of extended operation in accordance with 10 CFR 54.21(c) as described below.

A.4.3.1 ASME Section III, Class 1 Fatigue Analyses

The LSCS reactor pressure vessel and reactor coolant pressure boundary piping and components were designed in accordance with the ASME Code Section III, Class 1 design requirements. Fatigue analyses and fatigue exemptions were prepared for these components to determine the effects of transients that result in cyclic loadings caused by changes in system temperature and pressure and for seismic loading cycles, as well as for hydrodynamic suppression pool loadings resulting from LOCA events. These Class 1 fatigue analyses are included in stress reports that evaluated an explicit number and type of transients to envelope the number of occurrences projected during the 40-year design life of the plant. Each analysis was required to demonstrate that the cumulative usage factor (CUF) for the component will not exceed the design limit of 1.0 when the component is exposed to all of the postulated transients. The Class 1 valve analyses were required to demonstrate that the valves can be operated for a minimum of 2,000 cycles and that the fatigue usage factor for step changes in fluid temperature does not exceed a limit of 1.0. Each of these stress reports include fatigue analyses and fatigue exemptions, where applicable, that have been identified as TLAAs since they are based upon a set of 40-year design transients, which are the CLB fatigue design cycles. When used in Class 1 fatigue analyses, these design cycles become CLB fatigue design cycle limits that must not be exceeded.

Each of the Class 1 fatigue analyses and fatigue exemptions was evaluated for 60 years by determining if the numbers of cycles assumed in the 40-year analysis will remain bounding of the numbers of cycles projected for the component through the end of the period of extended operation. These 60-year projections were based upon cumulative cycles to-date plus future cycles which were based upon past occurrences. Nearly all of the 60-year projections show that the CLB fatigue design cycle limits will not be reached by the end of the period of extended operation. However, for Unit 1, the 60-year projections for Startup and Shutdown transients slightly exceed the design cycle limits for the reactor vessel, but are less than the Class 1 piping CLB fatigue design cycle limits. The Fatigue Monitoring program is credited for managing these fatigue TLAAs in accordance with 10 CFR 54.21(c)(1)(iii), and the program includes requirements that trigger corrective action prior to exceeding the transient limits to ensure the component cumulative usage factor (CUF) is not permitted to exceed the design limit of 1.0 for these components. Corrective action may include repair or replacement of affected components or reanalysis of affected Class 1 components for an increased number of cycles.

The effects of aging on the intended functions of components analyzed in accordance with ASME Section III, Class 1 requirements will be adequately managed by the Fatigue Monitoring (A.3.1.1) program through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.4.3.2 ASME Section III, Class 2 and 3 and ANSI B31.1 Allowable Stress Analyses

Piping designed in accordance with ASME Section III, Class 2 or 3, or ANSI B31.1 Piping Code design rules is not required to have an explicit analysis of cumulative fatigue usage, but cyclic loading is considered in the design process. If the numbers of anticipated thermal cycles exceed specified limits, these codes require the application of a stress range reduction factor to the allowable stress to prevent damage from cyclic loading. This is considered to be an implicit fatigue analysis since it is based upon the anticipated number of cycles for the life of the component.

These codes first require the overall number of thermal and pressure cycles expected during the 40-year lifetime of these components to be determined. A stress range reduction factor is then determined for that number of cycles using the applicable design code. If the total number of cycles is 7,000 or less, the stress range reduction factor of 1.0 is applied, which would not reduce the allowable stress values. For higher numbers of cycles, the stress range reduction factor limits the allowable stresses that can be applied to the piping.

Class 2 and 3 and ANSI B31.1 piping systems that are extended from Class 1 systems are affected by the same operational transients that result in thermal cycles for the attached Class 1 piping. These transient cycles are monitored by the Fatigue Monitoring (A.3.1.1) program. The 60-year cycle projections for these transients demonstrate that the total number of thermal cycles for these piping systems will not exceed 10 percent of the 7,000 cycle threshold that would result in a reduction in the stress range reduction factor. Therefore, these TLAAs have been demonstrated to remain valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

For the remaining Class 2 and 3 and ANSI B31.1 piping systems that are affected by thermal transients that are different than those monitored for Class 1 systems, an operational review was performed. This includes portions of the Reactor Core Isolation Cooling, Fire Protection, and Diesel Generator and Auxiliaries Systems. The review concluded that the total number of thermal cycles for these systems, projected through the period of extended operation, will not exceed 20 percent of the 7,000 cycle threshold. Therefore, the stress range reduction factors originally selected for the Class 2 and 3 and ANSI B31.1 piping systems remain applicable and these TLAAs have been demonstrated to remain valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.3.3 <u>Environmental Fatigue Analyses for RPV and Class 1 Piping</u>

NUREG-1800, Revision 2 provides a recommendation for evaluating the effects of the reactor water environment on the fatigue life of ASME Section III Class 1 components that contact reactor coolant. One method to satisfy this recommendation is to assess the impact of the reactor coolant environment on a sample of critical components as described in NUREG/CR-6260. Additional component locations are evaluated if they are considered to be more limiting than those considered in NUREG/CR-6260.

Environmental fatigue calculations were performed for component locations listed in NUREG/CR-6260 for the newer-vintage BWR. In order to ensure that any other locations that may not be bounded by the NUREG/CR-6260 locations were evaluated, environmental fatigue calculations were performed for each RPV component location that normally contacts reactor coolant and has a reported cumulative usage factor (CUF) in the RPV stress report and for the limiting wetted surface location within each ASME Class 1 system (or group of systems that are affected by the same transients). These environmental fatigue calculations were performed for the limiting wetted location for each material type that contacts reactor coolant. The Fatigue Monitoring program will be enhanced to evaluate the impact of the reactor coolant environment on Class 1 components including valves and pumps if they are more limiting than those considered in NUREG/CR-6260.

NUREG-1800, Revision 2, specifies options for evaluating environmental effects. The formulae specified in the option listed below for each material were used in evaluating the LSCS components for environmental effects:

<u>Carbon and Low Alloy Steels</u>

- Those provided in Appendix A of NUREG/CR-6909, using the fatigue design curve for carbon and low alloy steel provided in NUREG/CR-6909 (Figure A.1 and A.2, respectively, and Table A.1).

<u>Austenitic Stainless Steels</u>

- The formula provided in NUREG/CR-6909, using the fatigue design curve for austenitic stainless steel provided in NUREG/CR-6909 (Figure A.3 and Table A.2).

<u>Nickel Alloys</u>

- The formula provided in NUREG/CR-6909, using the fatigue design curve for austenitic stainless steel provided in NUREG/CR-6909 (Figure A.3 and Table A.2).

Additional refinements were performed as appropriate and several locations required a reduction in the numbers of postulated cycles. The resulting environmentally-adjusted CUF values (CUFen) were demonstrated not to exceed the design Code limit of 1.0.

These environmental fatigue analyses will be managed by the Fatigue Monitoring (A.3.1.1) program in the same manner as the ASME Section III, Class 1 fatigue analyses. The program ensures that the cumulative number of occurrences of each transient type is maintained below the number of cycles used in the most limiting fatigue analysis.

If a cycle limit is approached, corrective actions are triggered to prevent exceeding the limit. The fatigue analyses may be revised to account for increased numbers of cycles or transient severity such that the CUF value does not exceed the Code design limit of 1.0, including environmental effects where applicable.

Prior to the period of extended operation, the Fatigue Monitoring program will be enhanced to impose administrative transient cycle limits corresponding to the limiting numbers of cycles used in the environmental fatigue calculations.

The effects of aging on the intended functions will be adequately managed by the Fatigue Monitoring (A.3.1.1) program through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.4.3.4 Reactor Vessel Internals Fatigue Analyses

The RPV and RPV internal components were included in the NSSS New Loads Design Adequacy Evaluations performed for each unit to address the effects of plant-specific seismic loadings and suppression pool hydrodynamic structural loadings on NSSS equipment. These evaluations included fatigue analyses of components if the applied loadings exceed certain thresholds. These fatigue analyses and fatigue exemptions have been identified as TLAAs that require evaluation for the period of extended operation.

The fatigue analyses and fatigue exemptions performed for the reactor internals components are based upon the same set of design transients as those used in the fatigue analyses for the reactor pressure vessel. Nearly all of the 60-year projections show that the design cycle limits will not be reached by the end of the period of extended operation. However, for Unit 1, the 60-year projections for Startup and Shutdown transients slightly exceed the design cycle limits for the reactor vessel internals components. To ensure that these fatigue analyses and fatigue exemptions will remain valid, the Fatigue Monitoring (A.3.1.1) program will be used to manage fatigue of these components through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.4.3.5 High-Energy Line Break (HELB) Analyses Based Upon Fatigue

High-Energy Line Break (HELB) analyses for LSCS used the CUF values from the ASME Class 1 fatigue analyses as input in determining intermediate break locations. Since the Class 1 fatigue analyses that provided the CUF values are based upon 40-year transient assumptions, the HELB analyses have been identified as TLAAs.

Transient cycle projections for Class 1 piping were performed that determined the 40-year transient cycle limits for piping are not expected to be exceeded in 60 years. The Fatigue Monitoring (A.3.1.1) program is credited with ensuring that the numbers of actual transient cycles do not exceed the numbers of transient cycles analyzed in the Class 1 piping fatigue analyses that provided the CUF values less than 0.1 used in the HELB analyses. Therefore, these fatigue analyses will be managed through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.4.3.6 <u>Main Steam Relief Valve Discharge Piping Fatigue Analysis</u>

The Main Steam Relief Valve (MSRV) discharge lines have been evaluated for cumulative fatigue usage based on the number of transient cycles predicted to occur in 40 years. Therefore, these fatigue analyses have been identified as TLAAs.

The MSRV discharge lines were analyzed for dynamic and hydrodynamic loads from normal and upset conditions and loss of coolant accident (LOCA) events that result in MSRV actuations. They were analyzed for 2,800 SRV actuations and a resulting 8,400 acoustic cycles.

The 60-year transient projections show that the total number of MSRV actuations will not exceed the number analyzed for 40 years in either unit. No OBE or SSE event has occurred to-date and neither event has a 60-year projection that exceeds the design limit of one event. Therefore, the Main Steam Relief Valve discharge piping fatigue analysis remains valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.4 <u>Environmental Qualification (EQ) of Electric Components</u>

A.4.4.1 <u>Environmental Qualification (EQ) of Electric Components</u>

Thermal, radiation, and cyclical aging analyses of plant electrical and I&C components, developed to meet 10 CFR 50.49 requirements, have been identified as time-limited aging analyses (TLAAs) for LSCS. The NRC has established nuclear station environmental qualification (EQ) requirements in 10 CFR 50.49 and 10 CFR 50, Appendix A, Criterion 4. 10 CFR 50.49 specifically requires that an EQ program be established to demonstrate that certain electrical components located in harsh plant environments are qualified to perform their safety function in those harsh environments after the effects of inservice aging. Harsh environments are defined as those areas of the plant that could be subject to the harsh environmental effects of a loss of coolant accident (LOCA), high-energy line break (HELB), or post-LOCA radiation. 10 CFR 50.49 requires that the effects of significant aging mechanisms be addressed as part of environmental qualification.

The Environmental Qualification (EQ) of Electric Components (A.3.1.3) program will manage the effects of aging effects for the components associated with the environmental qualification through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii). This program implements the requirements of 10 CFR 50.49 (as further defined and clarified by NUREG-0588, and RG 1.89. Reanalysis of component aging evaluations is performed on a routine basis to extend the qualifications of components as part of the EQ Program. Important attributes for the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met).

Under the EQ Program, the reanalysis of an aging evaluation could extend the qualification of the component. If the qualification cannot be extended by reanalysis, the component must be refurbished, replaced, or requalified prior to exceeding the period for which the current qualification remains valid. A reanalysis is to be performed in a timely manner such that sufficient time is available to refurbish, replace, or requalify the component if the reanalysis is unsuccessful.

A.4.5 Concrete Containment Tendon Prestress Analyses

A.4.5.1 Concrete Containment Tendon Prestress Analyses

The containment tendon prestressing forces were calculated during the original design considering the magnitude of the tendon relaxation and concrete creep and shrinkage over the 40-year life of the plant. The predicted lower limit force values and regression analyses, utilizing actual measured tendon forces are used to evaluate the acceptability of the containment structure to perform its intended function over the current 40-year life of the plant, and therefore have been identified as TLAAs requiring evaluation for the period of extended operation.

The ASME Section XI, Subsection IWL (A.2.1.30) program performs periodic surveillances of individual tendon prestressing values. Predicted lower limit (PLL) force values are calculated for each tendon prior to the surveillances to estimate the magnitude of the tendon relaxation and concrete creep and shrinkage for the given surveillance year. The prestressing forces are measured and plotted, and trend lines are developed to ensure the tendon group (vertical and horizontal tendon types) prestressing values remain above the respective minimum required values (MRVs) until the next scheduled surveillance and for the 40-year license period.

The Concrete Containment Tendon Prestress (A.3.1.2) program will monitor and manage the TLAAs and the associated loss of tendon prestressing forces through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii). The regression analyses are periodically updated following successive surveillances to ensure that estimated values remain above the MRVs until the next scheduled surveillance and for the 60-year life of the plant. Individual measured tendon prestressing forces will be compared to predicted PLL values and trend lines developed for the period of extended operation.

A.4.6 Primary Containment Fatigue Analyses

A.4.6.1 Primary Containment Liner and Penetrations Fatigue Analyses

The LSCS primary containment liner, penetrations, and Class MC components were designed and analyzed for the transient cycles predicted for 40 years. The containment liner and Class MC components are non-Class 1 and are designed in accordance with Subsection NE of the ASME B&PV Code, Section III. These analyses have been identified as TLAAs.

The 60-year transient cycle projections for Unit 2 demonstrate that the transient cycle limits used in the containment analyses will not be exceeded in 60 years. This includes normal, upset, and emergency events, and design basis accidents. The 60-year transient cycle projections for Unit 1 show that the transient cycle limits for most events will not be exceeded in 60 years, but the numbers of startup and shutdown cycles are projected to slightly exceed their design limits. For both units, the Fatigue Monitoring program will be used to monitor and track the analyzed transients and to trigger corrective action prior to exceeding the transient limits to ensure the component cumulative usage factor (CUF) is not permitted to exceed the design limit of 1.0 for these components. The effects of aging on the intended functions of components analyzed in accordance with ASME Section III, Class 1 and non-Class 1 requirements will be managed by the Fatigue Monitoring (A.3.1.1) program through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

There are certain other primary containment penetrations that contain non-Class 1 piping that were analyzed for transients that are not monitored by the Fatigue Monitoring (A.3.1.1) program. Due to the high numbers of allowable cycles in these analyses, they are dispositioned in accordance with 10 CFR 54.21(c)(1)(i) – the analyses remain valid through the period of extended operation.

The analysis for primary containment penetration M-76 was reevaluated based on 60-year projections of RCIC injection cycles that are monitored by the Fatigue Monitoring program plus 60-year projected cycles for surveillance test cycles not monitored by the Fatigue Monitoring (A.3.1.1) program. This analysis is dispositioned in accordance with 10 CFR 54.21(c)(ii) – the analysis has been projected through the period of extended operation.

A.4.6.2 Primary Containment Refueling Bellows Fatigue Analysis

The refueling bellows and supports were analyzed for cumulative fatigue usage based on the number of transient cycles predicted to occur in 40 years, but this conservatively included 200 startup and shutdown cycles. Therefore, these fatigue analyses have been identified as TLAAs that require evaluation for the period of extended operation.

Transient cycle projections were performed that determined the 40-year transient cycle limits will not be exceeded in 60 years based upon the average rate of occurrences to-date. The analyses remain valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.6.3 Primary Containment Downcomer Vents Fatigue Analysis

The downcomer vents and bracing inside the suppression chamber were analyzed for cumulative fatigue usage based on the number of transient cycles predicted to occur in 40 years. Therefore, this fatigue analysis has been identified as a TLAA.

The downcomers and bracing were analyzed for seismic loads and thermal and cyclic loads resulting from Main Steam Relief Valve (MSRV) openings and the discharge of steam from the drywell to the suppression pool during a loss of coolant accident (LOCA) event. The 60-year transient projections show that the total number of MSRV actuations will not exceed the number analyzed for 40 years in either unit. No OBE or SSE event has occurred to-date and neither event has a 60-year projection that exceeds the design limit of one event. Therefore, the primary containment downcomer vent fatigue analysis has been demonstrated to remain valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.7 Other Plant–Specific Analyses

A.4.7.1 Reactor Building Crane Cyclic Loading Analysis

The LSCS reactor building crane is designed to meet the fatigue requirements of the NOG-1-2004 and Crane Manufacturers Association of America (CMAA) Specification 70 for a Class A, Standby or Infrequent Service Crane, as discussed in UFSAR Section 9.1.4.2.3, "Reactor Building Crane." For this crane, the CMAA design considerations allow for between 20,000 and 100,000 load cycles. 20,000 cycles is a conservative limitation on load cycles for this crane. This evaluation of cycles over the 40-year plant life has been identified as a TLAA that requires evaluation for the period of extended operation.

The evaluation of the reactor building crane cyclic loading TLAA included (1) reviewing the existing 40-year design basis to determine the number of load cycles considered in the design of the crane, (2) developing a 60-year projection for load cycles for the crane, and (3) comparing the 60-year projected number of cycles to the limiting value of 20,000 load cycles. The number of cycles projected for 60 years of operation is less than 20 percent of the limiting design value. Therefore, the reactor building crane cyclic loading analysis remains valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.7.2 Main Steam Line Flow Restrictors Erosion Analysis

A main steam line flow restrictor is welded into each of the four main steam lines between the main steam relief valves and the inboard main steam isolation valve (MSIV). The restrictor assemblies consist of a stainless steel venturi-type nozzle welded into the carbon steel main steam line piping. The restrictors are designed to limit steam flow prior to MSIV closure in the event of a main steam line break outside of primary containment.

The analysis of main steam line flow restrictor erosion is discussed in UFSAR Section 5.4.4. UFSAR Section 5.4.4 indicates that very slow erosion occurs with time and such slight enlargement has no safety significance. Since the erosion evaluation was based on 40 years of operation, erosion of the main steam line flow restrictor has been identified as a TLAA that requires evaluation for the period of extended operation.

Calculations indicate that even with erosion rates as high as 0.004 inches per year the increase in choked flow rate would be no more than five percent after 40 years of operation. The increase in choked flow rate is projected to be no more than 10 percent after 60 years. The LSCS analysis for radiation dose consequences resulting from a main steam line break outside containment projects a dose of less than one percent of the 10 CFR 100 limits. Therefore, sufficient margin exists to allow for the projected increase in steam flow resulting from erosion of the main steam line flow restrictor through the end of the period of extended operation.

The analysis has been projected through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.5.0 License Renewal Commitment List

Note: This table contains commitments related to the renewed LaSalle County Station, Units 1 and 2 operating licenses. Because these commitments are contained within the UFSAR, any potential changes to these commitments require evaluation in accordance with 10 CFR 50.59 as defined per the Exelon commitment management process. In addition, the commitment management process must be followed to ensure that the commitment-tracking database is updated with the latest commitment implementation information. Refer to Passport AR 1603204 for LaSalle license renewal commitment tracking information.

Table A.5-1 License Renewal Commitment List

NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
1	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Existing program is credited.	Ongoing	Section A.2.1.1
2	Water Chemistry	Existing program is credited.	Ongoing	Section A.2.1.2
3	Reactor Head Closure Stud Bolting	Existing program is credited.	Ongoing	Section A.2.1.3
4	BWR Vessel ID Attachment Welds	Existing program is credited.	Ongoing	Section A.2.1.4
5	BWR Feedwater Nozzle	Existing program is credited.	Ongoing	Section A.2.1.5
6	BWR Control Rod Drive Return Line Nozzle	Existing program is credited.	Ongoing	Section A.2.1.6
7	BWR Stress Corrosion Cracking	Existing program is credited.	Ongoing	Section A.2.1.7
8	BWR Penetrations	Existing program is credited.	Ongoing	Section A.2.1.8

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NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
9	BWR Vessel Internals	BWR Vessel Internals is an existing program that will be enhanced to: 1. Perform an assessment of the susceptibility of reactor vessel internal components fabricated from CASS to loss of fracture toughness due to thermal aging embrittlement. If material properties cannot be determined to perform the screening, they will be assumed susceptible to thermal aging for the purposes of determining program examination requirements. 2. Perform an assessment of the susceptibility of reactor vessel internal components fabricated from CASS to loss of fracture toughness due to neutron irradiation embrittlement. 3. Specify the required periodic inspection of CASS components determined to be susceptible to loss of fracture toughness due to thermal aging and neutron irradiation embrittlement. The initial inspections will be performed either prior to or within five years after entering the period of extended operation. 4. Install core plate wedges no later than six months prior to the period of extended operation, or before the end of the last refueling outage prior to the period of extended operation, whichever occurs later; or, submit an inspection plan for the core plate rim hold-down bolts with a supporting analysis for NRC approval at least 2 years prior to entering the period of extended operation.	Program to be enhanced no later than six months prior to the period of extended operation. Additional schedule information identified in commitment.	Section A.2.1.9 Exelon Letter RS-16-003 01/07/2016 Exelon Letter RS-16-007 01/14/2016
10	Flow-Accelerated Corrosion	Existing program is credited.	Ongoing	Section A.2.1.10
11	Bolting Integrity	Bolting Integrity is an existing program that will be enhanced to: 1. Provide guidance to ensure proper specification of bolting material, lubricant and sealants, storage, and installation torque or tension to prevent or mitigate degradation and failure of closure bolting for pressure-retaining bolted joints. 2. Prohibit the use of lubricants containing molybdenum disulfide on pressure-retaining bolted joints.	Program to be enhanced no later than six months prior to the period of extended operation. Additional schedule information identified in commitment.	Section A.2.1.11 Exelon Letter RS-15-194 08/06/2015 Exelon Letter RS-16-003 01/07/2016

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Table A.5-1 License Renewal Commitment List

NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
		3. Minimize the use of high strength bolting (actual measured yield strength equal to or greater than 150 ksi) for pressure-retaining bolted joints in portions of systems within the scope of the Bolting Integrity program. High strength bolting (regardless of code classification) will be monitored for cracking in accordance with ASME Section XI, Table IWB-2500-1, Examination Category B-G-1. 4. Perform inspection of submerged bolting for the emergency core cooling systems (ECCS) and reactor core isolation cooling (RCIC) system suction strainers in the suppression pool for loss of material and loss of preload during each ISI inspection interval. 5. Perform inspection of submerged bolting for the service water diver safety barriers and diesel fire pump suction screens for loss of material and loss of preload each refuel interval. 6. Perform inspection of submerged bolting for the Lake Screen House travelling screens framework for loss of material and loss of preload each refueling interval.		
12	Open-Cycle Cooling Water System	Open-Cycle Cooling Water System is an existing program that will be enhanced to: 1. Perform a minimum of 10 microbiologically influenced corrosion (MIC) degradation inspections for aboveground piping in the Essential Cooling Water System every 24 months until the rate of MIC occurrences no longer meets the criteria for recurring internal corrosion. The selected inspection locations will be periodically reviewed to validate their relevance and usefulness and adjusted as appropriate. Evaluation of the inspection results will include; (1) a comparison to the nominal wall thickness or previous wall thickness measurements to determine rate of corrosion degradation; (2) a comparison to the design minimum allowable wall thickness to determine the acceptability of the component for continued use; and (3) a determination of re-inspection interval. A portion of these inspection locations will be selected with process conditions similar (e. g. flow, temperature) to those in buried portions of the piping to provide sufficient understanding of the condition of the buried piping.	Program to be enhanced no later than six months prior to the period of extended operation. Additional schedule information identified in commitment.	Section A.2.1.12 Exelon Letter RS-16-003 01/07/2016

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NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
		2. Perform a minimum of 10 MIC degradation inspections for in scope aboveground piping in the Nonessential Cooling Water System every 24 months until the rate of MIC occurrences no longer meets the criteria for recurring internal corrosion. The selected inspection locations will be periodically reviewed to validate their relevance and usefulness and adjusted as appropriate. Evaluation of the inspection results will include (1) a comparison to the nominal wall thickness or previous wall thickness measurements to determine rate of corrosion degradation; (2) a comparison to the design minimum allowable wall thickness to determine the acceptability of the component for continued use; and (3) a determination of re-inspection interval. A portion of these inspection locations will be selected with process conditions similar (e. g. flow, temperature) to those in buried portions of the piping to provide sufficient understanding of the condition of the buried piping. 3. Select an inspection method that will provide indication of suitable wall thickness to perform inspections on a representative sample of buried piping to supplement the aboveground piping inspection locations. 4. Perform visual inspections of the interior surface of buried portions of the Essential Cooling Water System and Nonessential Cooling Water System whenever the piping internal surface is made accessible due to maintenance and repair activities.		
13	Closed Treated Water Systems	Closed Treated Water Systems is an existing program that will be enhanced to: 1. Perform condition monitoring, including periodic visual inspections and non-destructive examinations, to verify the effectiveness of water chemistry control to mitigate aging effects. A representative sample of piping and components will be selected based on likelihood of corrosion, stress corrosion cracking, or fouling, and inspected at an interval not to exceed once in 10 years during the period of extended operation. The selection of components to be inspected will focus on locations which are most susceptible to age-related degradation, where practical. 2. Monitor and trend drywell penetration cooling coil outlet temperatures monthly to ensure that adequate cooling is being provided to the concrete adjacent to the drywell penetrations.	Program to be enhanced no later than six months prior to the period of extended operation. Additional schedule information identified in commitment.	Section A.2.1.13 Exelon Letter RS-15-193 08/06/2015 Exelon Letter RS-16-003 01/07/2016

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Table A.5-1 License Renewal Commitment List				
NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
14	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems is an existing program that will be enhanced to: 1. Provide additional guidance to include inspection of structural components, rails, and bolting for loss of material due to corrosion; rails for loss of material due to wear; and bolted connections for loss of preload.	Program to be enhanced no later than six months prior to the period of extended operation.	Section A.2.1.14 Exelon Letter RS-16-003 01/07/2016
15	Compressed Air Monitoring	Compressed Air Monitoring is an existing program that will be enhanced to: 1. Inspect the internal surfaces of system filters, compressors, and after-coolers for signs of corrosion and corrosion products. 2. Perform analysis and trending of air quality monitoring results and visual inspection results. 3. Document deficiencies which are identified during visual inspections of the internal surfaces of system components in the corrective action program.	Program to be enhanced no later than six months prior to the period of extended operation.	Section A.2.1.15 Exelon Letter RS-16-003 01/07/2016
16	Fire Protection	Fire Protection is an existing program that will be enhanced to: 1. Perform periodic visual inspection of combustible liquid spill retaining curbs. 2. Perform periodic visual inspection for identification of corrosion that may lead to loss of material on the external surfaces of the low pressure carbon dioxide fire suppression systems. 3. Provide additional inspection guidance to identify aging effects as follows: a. Fire barrier walls, ceilings, and floors degradation such as spalling, cracking, and loss of material for concrete. b. Elastomeric fire barrier material degradation such as loss of material, shrinkage, separation from walls and components, increased hardness and loss of strength. 4. Provide additional inspection guidance to identify degradation of fire barrier penetration seals for aging effects such as loss of material, cracking, increased hardness, shrinkage, and loss of strength.	Program to be enhanced no later than six months prior to the period of extended operation.	Section A.2.1.16 Exelon Letter RS-16-003 01/07/2016

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Table A.5-1 License Renewal Commitment List

NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
17	Fire Water System	Fire Water System is an existing program that will be enhanced to: 1. Perform volumetric examinations at five locations on the carbon steel aboveground fire water piping susceptible to microbiologically induced corrosion (MIC) every year to identify loss of material. Additional locations will be examined if these volumetric examinations or plant operating experience identify significant degradation. For through-wall leaks and material loss greater than 50 percent of nominal wall, four additional locations will be examined. Where the identified material loss is 30 percent to 50 percent of nominal wall thickness and the calculated remaining life is less than two years, two additional locations will be examined. 2. Perform visual inspections, for loss of material and flow obstructions, of the accessible header piping and sparger external surfaces for the deluge systems located within filter plenums on a once per refueling cycle frequency. The visual inspection will include verification that the piping and spargers are in their proper position and that there are no obstructions to the desired spray patterns. 3. Perform internal visual inspections of sprinkler and deluge system piping to identify internal corrosion and obstructions to flow. If the presence of sufficient foreign organic or inorganic material to obstruct pipe or sprinklers is detected during pipe inspections, the material is removed and its source is determined and corrected where possible. Followup volumetric examinations will be performed if internal visual inspections detect age-related degradation in excess of what would be expected accounting for design, previous inspection experience, and inspection interval. The internal visual inspections will consist of the following: a. Wet pipe sprinkler systems – 50 percent of the wet pipe sprinkler systems in scope for license renewal will have visual internal inspections of piping performed every five years consistent with NFPA 25, 2011 Edition, Section 14.2.	Program to be enhanced no later than six months prior to the period of extended operation. Inspections described in Enhancement 3.d.i and 3.d.ii, if required, will be completed either no later than six months prior to the PEO, or before the end of the last refueling outage prior to the PEO, whichever occurs later. Additional schedule information identified in commitment.	Section A.2.1.17 Exelon Letter RS-15-171 07/01/2015 Exelon Letter RS-16-003 01/07/2016

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NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
		b. Dry pipe sprinkler systems - Dry pipe sprinkler systems in scope for license renewal will have visual internal inspections of piping performed every five years consistent with NFPA 25, 2011 Edition, Section 14.2. c. Deluge systems - Deluge systems in scope for license renewal, except for the charcoal filter deluge systems, will have visual internal inspections of piping performed every five years consistent with NFPA 25, 2011 Edition, Section 14.2. i. The in scope charcoal filter deluge systems will have visual internal inspections performed on one of the 11 systems every five years. If degraded conditions are identified, the inspections will be expanded to include all 11 charcoal filter systems every five years. d. Sprinkler and deluge systems that are normally dry but may be wetted as the result of testing or actuations will have additional tests and inspections on piping segments that cannot be drained or piping segments that allow water to collect. i. These additional inspections, if required, will be performed in each five-year interval beginning five years prior to the period of extended operation. ii. This additional inspection consists of either a flow test or flush sufficient to detect potential flow blockage or a visual inspection of 100 percent of the internal surface of piping segments that cannot be drained or piping segments that allow water to collect. iii. In addition, in each five-year interval of the period of extended operation, 20 percent of the length of piping segments that cannot be drained or piping segments that allow water to collect is subject to volumetric wall thickness inspections.		

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Table A.5-1 License Renewal Commitment List				
NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
		4. Perform obstruction evaluations when degraded conditions are identified by visual inspections, flow testing, or volumetric examinations. The obstruction evaluations will include an extent of condition determination, need for increased inspections, and followup examinations if internal visual inspections detect age-related degradation in excess of what would be expected accounting for design, previous inspection experience, and inspection interval. 5. Perform flow tests for hose stations at the hydraulically most limiting locations for each zone of the system on a five-year frequency to demonstrate the capability to provide the design pressure at required flow. 6. Perform annual air tests on deluge systems supporting charcoal filter units excluding the "A" and "B" Auxiliary Electric Equipment Room Supply Air Filter units and the High Radiation Sampling System Filter unit. Perform visual internal inspections of the excluded filter units deluge systems in the event blockage was found on any deluge system that could be generic in nature. 7. Perform visual inspections of all charcoal filter unit deluge nozzles for proper orientation and verification that the nozzles are not obstructed on a 24 month frequency. 8. Include inspection for water leakage and loss of fluid in the glass bulbs of sprinkler heads, when performing visual inspections of sprinkler systems. 9. Include in main drain test acceptance criteria, the monitoring of flowing pressures from test to test. If there is a ten percent reduction in full flow pressure when compared to previously performed tests, an issue report shall be generated in the corrective action program to determine the cause and correct if necessary. 10. Maintain yard loop flow testing at a two year frequency until such time that the restricted section of piping from the pump house to Node 515 is restored to normal flow conditions.		

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NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
18	Aboveground Metallic Tanks	Aboveground Metallic Tanks is an existing program that will be enhanced to: 1. Perform a visual inspection of the tank shell, roof, and bottom interior surfaces for signs of loss of material on one of the cycled condensate storage tanks within five years prior to the period of extended operation. This inspection shall include both wetted and non-wetted surfaces and may be either direct visual inspection from inside the tanks or volumetric examination from outside the tank. A volumetric examination from outside the tank will include 25 percent of the tank surface area. Should the one-time inspection identify degradation, periodic inspections with an inspection frequency based on the rate of degradation will be established for both tanks. 2. Perform a visual inspection of the exterior surfaces of both cycled condensate storage tanks for loss of material each refueling interval. 3. Perform a volumetric examination of the tank bottom for both cycled condensate storage tanks for signs of loss of material whenever the tanks are drained. At a minimum, an inspection shall be performed within 10 years prior to the period of extended operation and subsequent inspections shall be performed in each 10-year period during the period of extended operation. The next inspection scope will include 100% of the accessible areas of each of the tank bottoms that are within 30 inches of the shell. Included in this scope are the patch plates that are directly exposed to the sand bed below. Additionally, 10 random locations of approximately one square foot each, outside of the 30 inch band, will be inspected. This inspection program will encompass approximately 20% of the tank bottom and will inspect all the susceptible areas which were found during the baseline inspections. Based on the results of this inspection, the scope will be reassessed for future tank bottom inspections, per the Corrective Action Program. 4. Perform an inspection of the caulking at the perimeter of the cycled condensate storage tank bases for signs of degradation each refueling interval (caulking inspections are supplemented with physical manipulation).	Program to be enhanced no later than six months prior to the period of extended operation. Inspections described in Enhancements 1 and 3 will be completed either no later than six months prior to the PEO, or before the end of the last refueling outage prior to the PEO, whichever occurs later. Additional schedule information identified in commitment.	Section A.2.1.18 Exelon Letter RS-15-193 08/06/2015 Exelon Letter RS-16-003 01/07/2016

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NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
19	Fuel Oil Chemistry	Fuel Oil Chemistry is an existing program that will be enhanced to: 1. Perform periodic (quarterly) sampling and analysis for water and sediment content, particulate concentration, and the levels of microbiological organisms for diesel fuel storage tanks 0DO01T, 1DO01T, and 2DO01T. 2. Perform periodic (quarterly) sampling and analysis for water and sediment content, particulate concentration, and the levels of microbiological organisms for HPCS diesel fuel storage tanks 1DO02T and 2DO02T. 3. Perform periodic (quarterly) sampling and analysis for water and sediment content and the levels of microbiological organisms for diesel generator day tanks 0DO02T, 1DO05T, and 2DO05T. 4. Perform periodic (quarterly) sampling and analysis for water and sediment content and the levels of microbiological organisms for HPCS diesel day tanks 1DO04T and 2DO04T. 5. Perform periodic (quarterly) sampling and analysis for water and sediment content, particulate concentration, and the levels of microbiological organisms for diesel fire pump day tanks 0FP01TA and 0FP01TB. 6. Perform periodic internal inspections of diesel fire pump day tanks 0FP01TA and 0FP01TB at least once during the 10-year period prior to the period of extended operation, and at least once every 10 years during the period of extended operation. Each diesel fuel tank will be drained and cleaned, the internal surfaces visually inspected (if physically possible), and, if evidence of degradation is observed during inspections, or if visual inspection is not possible, these diesel fuel tanks will be volumetrically inspected. 7. Perform volumetric inspection of diesel fuel storage tanks 0DO01T, 1DO01T, and 2DO01T; HPCS diesel fuel storage tanks 1DO02T and 2DO02T; diesel generator day tanks 0DO02T, 1DO05T, and 2DO05T; and HPCS diesel day tanks 1DO04T and 2DO04T if evidence of degradation is observed during visual inspection, or if visual inspection is not possible.	Program to be enhanced no later than six months prior to the period of extended operation. Inspections described in Enhancements 6 and 7 will be completed either no later than six months prior to the PEO, or before the end of the last refueling outage prior to the PEO, whichever occurs later. Additional schedule information identified in commitment.	Section A.2.1.19 Exelon Letter RS-16-003 01/07/2016

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NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
		8. Perform periodic (quarterly) trending of water and sediment content, particulate concentration, and the levels of microbiological organisms for all fuel oil tanks within the scope of the program.		
20	Reactor Vessel Surveillance	Reactor Vessel Surveillance is an existing program that will be enhanced to: 1. Establish a maximum fluence limit of 6.25E+17 n/cm^2 (39.15 EFPY) for monitoring the limiting Unit 1 axial welds to ensure that the axial weld failure probability does not exceed 5.02E-06 per reactor year. 2. Complete a probabilistic axial weld failure analysis for Unit 1 that demonstrates the 60-year axial weld failure probability is no greater than 5.02E-06 per reactor year. Submit the analysis to the NRC for review and approval.	Enhancement 1 to be implemented no later than six months prior to the period of extended operation. Enhancement 2 to be completed at least three years prior to the limiting Unit 1 axial welds reaching the fluence limit of 6.25E+17 n/cm^2 (39.15 EFPY).	Section A.2.1.20 Section A.4.2.5 Exelon Letter RS-15-223 08/26/2015 Exelon Letter RS-15-281 10/29/2015 Exelon Letter RS-16-003 01/07/2016
21	One-Time Inspection	One-Time Inspection is a new condition monitoring program that will be used to verify the system-wide effectiveness of the Water Chemistry (A.2.1.2) program, Fuel Oil Chemistry (A.2.1.19) program, and Lubricating Oil Analysis (A.2.1.26) program which are designed to prevent or minimize aging to the extent that it will not cause a loss of intended function during the period of extended operation.	Program to be implemented no later than six months prior to the period of extended operation. One-time inspections will be performed within the 10 years prior to the period of extended operation, and will be completed either no later than six months prior to the PEO, or before the end of the last refueling outage prior to the PEO, whichever occurs later.	Section A.2.1.21 Exelon Letter RS-16-003 01/07/2016

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NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
22	Selective Leaching	Selective Leaching is a new condition monitoring program that will include one-time inspections of a representative sample of susceptible components to determine if loss of material due to selective leaching is occurring.	Program to be implemented no later than six months prior to the period of extended operation. One-time inspections will be performed within the five years prior to the period of extended operation, and will be completed either no later than six months prior to the PEO, or before the end of the last refueling outage prior to the PEO, whichever occurs later.	Section A.2.1.22 Exelon Letter RS-16-003 01/07/2016
23	Unit 1 One-time Inspection of ASME Code Class 1 Small-Bore Piping	Unit 1 One-time Inspection of ASME Code Class 1 Small-Bore Piping is a new condition monitoring program that will manage the aging effect of cracking in ASME Code Class 1 small-bore piping that is less than nominal pipe size (NPS) 4-inches, and greater than or equal to NPS 1-inch.	Program to be implemented no later than six months prior to the period of extended operation. One-time Inspections will be performed within the six years prior to the period of extended operation, and will be completed either no later than six months prior to the PEO, or before the end of the last refueling outage prior to the PEO, whichever occurs later.	Section A.2.1.23 Exelon Letter RS-15-193 08/06/2015 Exelon Letter RS-16-003 01/07/2016

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NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
24	External Surfaces Monitoring of Mechanical Components	External Surfaces Monitoring of Mechanical Components is a new condition monitoring program that directs visual inspections of external surfaces of components be performed during system inspections and walkdowns.	Program to be implemented no later than six months prior to the period of extended operation.	Section A.2.1.24 Exelon Letter RS-16-003 01/07/2016
25	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components	Internal Surfaces in Miscellaneous Piping and Ducting Components is new condition monitoring program that will consist of inspections of the internal surfaces of metallic and elastomeric components such as piping, piping components and piping elements, ducting components, tanks, heat exchanger components, elastomers, and other components that are exposed to environments of condensation, diesel exhaust, and waste water.	Program to be implemented no later than six months prior to the period of extended operation.	Section A.2.1.25 Exelon Letter RS-16-003 01/07/2016
26	Lubricating Oil Analysis	Existing program is credited.	Ongoing	Section A.2.1.26
27	Monitoring of Neutron-Absorbing Materials Other than Boraflex	Monitoring of Neutron-Absorbing Materials Other than Boraflex is an existing program that will be enhanced to: 1. Maintain the test coupon exposure such that it is bounding for the neutron-absorbing material in all spent fuel racks, by relocating the coupon tree to a different spent fuel rack cell location each cycle and by surrounding the coupons with a greater number of freshly discharged fuel assemblies than that of any other cell location.	Program to be enhanced no later than six months prior to the period of extended operation.	Section A.2.1.27 Exelon Letter RS-16-003 01/07/2016
28	Buried and Underground Piping	Buried and Underground Piping is an existing program that will be enhanced to: 1. Manage cracking for stainless steel piping, utilizing a method that has been demonstrated to be capable of detecting cracking, whenever coatings are removed and expose the base material. 2. Ensure all underground carbon steel Essential Cooling Water System and Nonessential Cooling Water System piping and components within the scope of license renewal are coated in accordance with Table 1 of NACE SP0169-2007. 3. Define acceptable coating conditions as coating exhibiting either no evidence of degradation, or, the type and extent of coating damage	Program to be enhanced no later than six months prior to the period of extended operation. Enhancement 2 and inspections described in Enhancements 4 and 5 will be completed either no later than six months prior to the PEO, or before the end of the last refueling outage prior to the PEO, whichever occurs later.	Section A.2.1.28 Exelon Letter RS-16-003 01/07/2016 Exelon Letter RS-16-070 4/13/2016

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NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
		evaluated as insignificant by: (a) an individual possessing a NACE Coating Inspector Program Level 2 or 3 operator qualification, (b) an individual who has attended the Electric Power Research Institute (EPRI) Comprehensive Coatings Course and completed the EPRI Buried Pipe Condition Assessment and Repair Training Computer Based Training Course, or (c) a coatings specialist qualified in accordance with an ASTM standard endorsed in Regulatory Guide 1.54, Rev. 2, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants." 4. Perform inspection quantities of buried piping within the scope of license renewal in accordance with LR-ISG-2015-01 Table XI.M41-2 and LR-ISG-2015-01, Appendix B, "Evaluation and Technical Basis" section, items 4.a and 4.b, during each 10-year period, beginning 10 years prior to the period of extended operation. The number of inspections of buried piping will be based upon the as-found results of cathodic protection system availability and effectiveness. The length of piping for each inspection will be based on the recommendations in LR-ISG-2015-01, Appendix B, "Evaluation and Technical Basis" section, item 4.c. 5. Perform direct visual inspections of underground Essential Cooling Water System and Nonessential Cooling Water System piping within the scope of license renewal during each 10-year period, beginning 10 years prior to the period of extended operation. 6. When measured pipe wall thickness, projected to the end of the period of extended operation, does not meet the minimum pipe wall thickness requirements, the number of inspections within the affected piping categories will be doubled or increased by five (5), whichever is smaller. If adverse indications are found in the expanded sample, an analysis will be conducted to determine the extent of condition and extent of cause. The size of the followup inspections will be determined based on the analysis. Timing of the additional inspections will be based on the severity of the identified degradation and the consequences of leakage. The additional inspections will be performed within the same 10-year inspection interval in which the original degradation was identified, or within 4-years after the end of the 10-year interval if the degradation was identified in the latter half of the 10-year interval. Expansion of sample size may be limited by the extent of piping subject to the observed degradation mechanism or if the piping system or portion of the system is replaced.	Additional schedule information identified in commitment.	

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NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
		7. Use only the -850mV relative to a CSE (copper/copper sulfate reference electrode), instant off criterion specified in NACE SP0169-2007 for acceptance criteria for steel piping and determination of cathodic protection system effectiveness in performing cathodic protection surveys. Alternatively, soil corrosion probes may also be used to demonstrate cathodic protection effectiveness during the annual surveys. An upper limit of -1200mV for pipe-to-soil potential measurements of coated pipes will also be established, so as to preclude potential damage to coatings. 8. Conduct an extent of condition evaluation if observed coating damage caused by non-conforming backfill has been evaluated as significant. The extent of condition evaluation will be conducted to ensure that the as-left condition of backfill in the vicinity of the observed damage will not lead to further degradation.		
29	ASME Section XI, Subsection IWE	ASME Section XI, Subsection IWE is an existing program that will be enhanced to: 1. Provide guidance for proper specification of bolting material, lubricant and sealants, and installation torque or tension to prevent or mitigate degradation and failure of structural bolting. 2. If leakage from the reactor cavity pool drain line welds exists, then perform ultrasonic thickness measurements on the Unit 2 drywell liner at 0 and 180 degrees for several feet below elevation 813. The inspections will begin in 2015 and a periodic inspection frequency will be established based on the inspection results.	Program to be enhanced no later than six months prior to the period of extended operation. Additional schedule information identified in commitment.	Section A.2.1.29 Exelon Letter RS-16-003 01/07/2016
30	ASME Section XI, Subsection IWL	ASME Section XI, Subsection IWL is an existing program that will be enhanced to: 1. Explicitly require that areas of concrete deterioration and distress be recorded in accordance with the guidance provided in ACI 349.3R. 2. Include quantitative acceptance criteria, based on the "Evaluation Criteria" provided in Chapter 5 of ACI 349.3R, that will be used to augment the qualitative assessment of the Responsible Engineer.	Program to be enhanced no later than six months prior to the period of extended operation.	Section A.2.1.30 Exelon Letter RS-16-003 01/07/2016

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Table A.5-1 License Renewal Commitment List

NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
31	ASME Section XI, Subsection IWF	ASME Section XI, Subsection IWF is an existing program that will be enhanced to: 1. Provide guidance for proper specification of bolting material, storage, lubricant and sealants, and installation torque or tension to prevent or mitigate degradation and failure of structural bolting. Requirements for high strength bolts shall include the preventive actions for storage, lubricants, and stress corrosion cracking potential discussed in Section 2 of RCSC (Research Council on Structural Connections) publication "Specification for Structural Joints Using ASTM A325 or A490 Bolts." Lubricants that contain molybdenum disulfide (MoS_2) shall not be applied to high strength bolts within the scope of license renewal. Bolting material with actual measured yield strength greater than or equal to 150 ksi or 1,034 MPa, in sizes greater than 1 inch nominal diameter shall not be used in supports for ASME Class 1, 2, and 3 piping and components or supports for MC components. 2. Provide guidance, regarding the selection of supports to be inspected on subsequent inspections, when a support is repaired in accordance with the corrective action program. The enhanced guidance will ensure that the supports inspected on subsequent inspections are representative of the general population.	Program to be enhanced no later than six months prior to the period of extended operation.	Section A.2.1.31 Exelon Letter RS-15-180 7/15/2015 Exelon Letter RS-16-003 01/07/2016
32	10 CFR Part 50, Appendix J	Existing program is credited.	Ongoing	Section A.2.1.32
33	Masonry Walls	Masonry Walls is an existing program that will be enhanced to: 1. Provide guidance for inspection of masonry walls for separation and gaps between the supports and masonry walls. 2. Require that personnel performing inspections and evaluations meet the qualifications described in ACI 349.3R.	Program to be enhanced no later than six months prior to the period of extended operation.	Section A.2.1.33 Exelon Letter RS-15-180 7/15/2015 Exelon Letter RS-16-003 01/07/2016

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Table A.5-1 License Renewal Commitment List				
NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
34	Structures Monitoring	Structures Monitoring is an existing program that will be enhanced to: 1. Add the following components and commodities: a. Pipe, electrical, and equipment component support members b. Pipe whip restraints and jet impingement shields c. Panels, racks, cabinets, and other enclosures d. Sliding surfaces e. Sumps f. Electrical cable trays and conduits g. Electrical duct banks h. Tube tracks i. Transmission tower (including takeoff towers) and foundation (including cycled condensate storage tank foundations) j. Penetration seals and sleeves k. Blowout panels l. Permanent drywell shielding m. Transformer foundation n. Bearing pads o. Compressible joints p. Hatches, plugs, handholes, and manholes q. Metal components (decking, vent stack, and miscellaneous steel) r. Building features – doors and seals, bird screens, louvers, windows, and siding s. Concrete curbs, and anchors t. Turbine Building smoke and heat vent housings	Program to be enhanced no later than six months prior to the period of extended operation.	Section A.2.1.34 Exelon Letter RS-15-180 7/15/2015 Exelon Letter RS-15-238 09/17/2015 Exelon Letter RS-16-003 01/07/2016

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Table A.5-1 License Renewal Commitment List

NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
		2. Provide guidance for proper specification of bolting material, lubricant and sealants, and installation torque or tension to prevent or mitigate degradation and failure of structural bolting. 3. Revise storage requirements for high strength bolts to include recommendations of Research Council on Structural Connections (RCSC) Specification for Structural Joints Using High Strength Bolts, Section 2.0. 4. Require acceptance and evaluation of structural concrete using quantitative criteria based on Chapter 5 of ACI 349.3R. 5. Monitor raw water and ground water chemistry on a frequency not to exceed five years for pH, chlorides, and sulfates and verify that it remains non-aggressive, or evaluate results exceeding criteria to assess impact, if any, on below-grade concrete. 6. Monitor concrete for increase in porosity and permeability, inspection of accessible sliding surfaces for indication of significant loss of material due to wear or corrosion, debris, or dirt. 7. Evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas, and examine representative samples of the exposed portions of the below grade concrete, when excavated for any reason. 8. Require that personnel performing inspections and evaluations meet the qualifications specified within ACI 349.3R with respect to knowledge of inservice inspection of concrete and visual acuity requirements. 9. Clarify that loose bolts and nuts and cracked high strength bolts are not acceptable unless accepted by engineering evaluations. 10. Inspect the fiberglass outer covering for the permanent drywell shielding for signs of rips and tears. If a rip or tear is found, repair or replace the permanent drywell shielding.		

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Table A.5-1 License Renewal Commitment List

NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
35	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants is an existing program that will be enhanced to: 1. Include monitoring of the following: a. Submerged Core Standby Cooling System Pond and Intake Flume (includes earthen walls, south flume concrete retaining wall, and north flume sheet piling retaining wall) b. Core Standby Cooling System outfall structure c. Bar racks and miscellaneous steel d. Shad net anchors e. Lake Screen House (includes service water tunnel) 2. Monitor raw water and ground water chemistry at least once every five years for pH, chlorides, and sulfates and verify that it remains non-aggressive, or evaluate results exceeding criteria to assess impact, if any, on buried or submerged concrete. 3. Provide guidance for proper specification of bolting material, lubricant and sealants, and installation torque or tension to prevent or mitigate degradation and failure of structural bolting, and preventative actions for storage of materials to prevent stress corrosion cracking. 4. Require acceptance and evaluation of structural concrete using quantitative criteria based on Chapter 5 of ACI 349.3R. 5. Require inspection of accessible in scope portions of the Cooling Lake and Lake Screen House immediately following the occurrence of significant natural phenomena, which includes intense local rainfalls and large floods.	Program to be enhanced no later than six months prior to the period of extended operation.	Section A.2.1.35 Exelon Letter RS-16-003 01/07/2016

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Table A.5-1 License Renewal Commitment List

NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
		6. Require: a. The evaluation of the acceptability of inaccessible areas when conditions exist in the accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas. b. Examination of the exposed portions of the below grade concrete when excavated for any reason.		
36	Protective Coating Monitoring and Maintenance Program	Existing program is credited.	Ongoing.	Section A.2.1.36
37	Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements is a new program that will be used to manage reduced insulation resistance of the insulation material for non-EQ cables and connections. Accessible cables and connections located in adverse localized environments will be visually inspected at least once every 10 years for indications of reduced insulation resistance, such as embrittlement, discoloration, cracking, melting, swelling, or surface contamination.	Program to be implemented no later than six months prior to the period of extended operation. Initial inspections will be completed either no later than six months prior to the PEO, or before the end of the last refueling outage prior to the PEO, whichever occurs later. Additional schedule information identified in commitment.	Section A.2.1.37 Exelon Letter RS-16-003 01/07/2016

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Table A.5-1 License Renewal Commitment List				
NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
38	Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits is a new program that will be used to manage the effects of reduced insulation resistance of non-EQ cable and connection insulation of the in scope portions of Neutron Monitoring, Process Radiation Monitoring and Area Radiation Monitoring Systems. Calibration and cable tests will be performed and results will be assessed for reduced insulation resistance prior to the period of extended operation. Cable test frequency is based on engineering evaluation and is performed at least once every 10 years. Calibration and assessment of results is performed at least once every 10 years during the period of extended operation.	Program to be implemented no later than six months prior to the period of extended operation. Initial testing and assessment of results will be completed either no later than six months prior to the PEO, or before the end of the last refueling outage prior to the PEO, whichever occurs later. Additional schedule information identified in commitment.	Section A.2.1.38 Exelon Letter RS-16-003 01/07/2016
39	Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements is a new program that will be used to manage the effects of reduced insulation resistance of non-EQ, in scope, inaccessible power cables. Cables will be tested using one or more proven tests for detecting deterioration of the insulation system. The cables will be tested at least once every six years. More frequent testing may occur based on test results and operating experience. Manholes associated with the cables included in this aging management program will be inspected for water collection with subsequent corrective actions (e.g., water removal), as necessary. Prior to the period of extended operation, the frequency of inspections for accumulated water will be established and adjusted based on plant specific operating experience with cable wetting or submergence, including water accumulation over time and	Program to be implemented no later than six months prior to the period of extended operation. Initial tests and inspections will be completed either no later than six months prior to the PEO, or before the end of the last refueling outage prior to the PEO, whichever occurs later.	Section A.2.1.39 Exelon Letter RS-16-003 01/07/2016

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Table A.5-1 License Renewal Commitment List

NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
		event driven occurrences such as heavy rain or flooding. Operation of dewatering devices will be verified prior to any known or predicted heavy rain or flooding event. During the period of extended operation, the inspections will occur at least annually.	Additional schedule information identified in commitment.	
40	Metal Enclosed Bus	Metal Enclosed Bus is an existing program that will be enhanced to: 1. Specify internal inspections will be performed for accessible non-segregated bus duct sections that are in scope for license renewal. 2. Clarify requirements for visual inspections of internal portions (bus enclosure assemblies); bus insulation; internal bus insulating supports; accessible gaskets, boots and sealants; and bus duct external surfaces. 3. Specify a sample size of 20 percent of the accessible bolted connection population, with a maximum sample size of 25, will be inspected for increased resistance of connection by either thermography or measuring the connection resistance using a micro ohmmeter. 4. Specify an inspection frequency of at least every 10 years.	Program to be enhanced no later than six months prior to the period of extended operation. Additional schedule information identified in commitment.	Section A.2.1.40 Exelon Letter RS-16-003 01/07/2016
41	Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program is a new program that will implement one-time testing of a representative sample (20 percent with a maximum sample size of 25) of non-EQ electrical cable connections to ensure that either increased resistance of connection is not occurring or that the existing preventive maintenance program is effective such that a periodic inspection program is not required.	Program to be implemented no later than six months prior to the period of extended operation. One-time tests will be completed either no later than six months prior to the PEO, or before the end of the last refueling outage prior to the PEO, whichever occurs later.	Section A.2.1.41 Exelon Letter RS-16-003 01/07/2016

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Table A.5-1 License Renewal Commitment List				
NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
42	Service Level III and Service Level III Augmented Coatings Monitoring and Maintenance Program	Service Level III and Service Level III Augmented Coatings Monitoring and Maintenance Program is a new condition monitoring program that performs periodic visual inspections to verify the integrity of internal coatings designed to adhere to and protect the base metal.	Program to be implemented no later than six months prior to the period of extended operation. Baseline inspections will occur in the 10-year period prior to the period of extended operation, and will be completed either no later than six months prior to the PEO, or before the end of the last refueling outage prior to the PEO, whichever occurs later.	Section A.2.2.1 Exelon Letter RS-16-003 01/07/2016
43	Fatigue Monitoring	Fatigue Monitoring is an existing program that will be enhanced to: 1. Impose administrative transient cycle limits corresponding to the limiting numbers of cycles used in the environmental fatigue calculations. 2. Evaluate the impact of the reactor coolant environment on Class 1 components including valves and pumps if they are more limiting than those considered in NUREG/CR-6260.	Program to be enhanced no later than six months prior to the period of extended operation. Any additional environmental fatigue evaluations will be completed no later than six months prior to the period of extended operation.	Section A.3.1.1 Exelon Letter RS-16-003 01/07/2016

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Table A.5-1 License Renewal Commitment List

NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
44	Concrete Containment Tendon Prestress	Concrete Containment Tendon Prestress is an existing condition monitoring program that will be enhanced to: 1. For each surveillance interval, trending lines will be updated through the period of extended operation as part of the regression analysis and compared to the predicted lower limit and minimum required values for each tendon group.	Program to be enhanced no later than six months prior to the period of extended operation.	Section A.3.1.2 Exelon Letter RS-16-003 01/07/2016
45	Environmental Qualification (EQ) of Electric Components	Existing program is credited.	Ongoing	Section A.3.1.3
46	Operating Experience	Existing program is credited.	Ongoing	Section A.1.6
47	TLAA - Slip Joint Clamp	Prior to exceeding the limiting fluence value of 1.17E+20 n/cm^2 at the Unit 1 jet pump slip joint clamp location, estimated to be at 50.7 EFPY, revise the analysis for the slip joint clamps for a higher acceptable fluence value or take other corrective action such as repair or replacement of the clamps to ensure acceptable clamp preload.	Procedure tracking the cumulative EFPY values to be enhanced no later than six months prior to the period of extended operation For the analysis option, if selected, the revised analysis must be submitted to the NRC for review at least 2 years prior to exceeding 50.7 EFPY. For the repair or replacement options, if selected, repair or replacement must be implemented prior to exceeding 50.7 EFPY.	Section A.4.2.10 Exelon Letter RS-15-232 09/15/2015 Exelon Letter RS-16-003 01/07/2016 Exelon Letter RS-16-033 02/01/2016

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Table A.5-1 License Renewal Commitment List

NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE*	SOURCE
48	Unit 2 Inspection of ASME Code Class 1 Small-Bore Piping Program	Unit 2 Inspection of ASME Code Class 1 Small-Bore Piping Program is a new condition monitoring program that will manage the aging effect of cracking in ASME Code Class 1 small-bore piping that is less than nominal pipe size (NPS) 4-inches, and greater than or equal to NPS 1-inch.	Program to be implemented no later than six months prior to the period of extended operation. The one-time inspections and the first set of periodic inspections will be performed within the six years prior to the period of extended operation, and will be completed either no later than six months prior to the PEO, or before the end of the last refueling outage prior to the PEO, whichever occurs later.	Section A.2.2.2 Exelon Letter RS-15-193 08/06/2015 Exelon Letter RS-16-003 01/07/2016

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