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Document Control Branch (Document Control Desk)

SUBJECT: Verifies that containment structural integrity & containment leakage rates are acceptable by TS SRs 4.6.1.6 & 4.6.1.2, respectively.Next periodic ILRT is due before May 2003. Basis of util's decision encl for info.

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FPL

Florida Power & Light Company, 6351 S. Ocean Drive, Jensen Beach, FL 34957

November 7, 1997

L-97-287
10 CFR 50.4

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

RE: St. Lucie Unit 1
Docket No. 50-335
Integrated Leak Test - Deferral
Steam Generator Replacement Outage

At meetings between Florida Power and Light Company (FPL) and the NRC Staff on June 17, 1997 and September 23, 1997, FPL presented the schedule for the steam generator replacement (SGR) outage. The schedule included the planned performance of a containment integrated leak rate test (ILRT).

FPL verifies containment structural integrity and containment leakage rates are acceptable by Technical Specification (TS) surveillance requirements, 4.6.1.6 and 4.6.1.2, respectively. These specifications rely on inspections and leakage testing performed in accordance with the Containment Leakage Rate Testing Program described in TS 6.8.4 (h). FPL will be performing the structural integrity inspections required by TS 4.6.1.2. However, FPL has determined that the containment leakage testing can be accomplished by the performance of a local leak rate test for the containment construction hatch weld and the ASME Section XI pressure test of the secondary side of the replacement steam generators. Therefore, an ILRT is not required for the SGR and the planned ILRT can be conducted at the next regularly scheduled ILRT. The next periodic ILRT is due before May 2003.

The basis of FPL's decision is attached for your information. FPL is prepared to discuss any staff questions on this determination.

Very truly yours,

J. Scarola for

J. A. Stall
Vice President
St. Lucie Plant

A017/1

Attachment

JAS/GRM

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant

ST. LUCIE UNIT 1
STEAM GENERATOR REPLACEMENT
EVALUATION FOR
CONTAINMENT TESTING REQUIREMENTS

1.0 DESCRIPTION AND PURPOSE

This evaluation will serve to document the testing requirements, as required by 10 CFR 50 Appendix J, which are applicable to the Unit 1 steel containment pressure vessel following the replacement of the two steam generators scheduled for the Fall 1997 refueling/steam generator replacement outage (SL1-15). The testing approach for the Unit 1 containment has been evaluated against the facility change criteria established by 10 CFR 50.59.

The two Unit 1 steam generators are being replaced during the Fall 1997 refueling outage (SL1-15). The St Lucie Unit 1 containment pressure vessel (Figure 1) is a free standing carbon steel vessel with four major penetrations (Figure 2): an escape lock; a personnel lock; a maintenance hatch; and a construction hatch. Access for the new replacement steam generators, as well as removal of the old steam generators, will require opening the existing construction hatch, which is an integrally reinforced penetration through the steel containment vessel wall with a welded closure cap. The construction hatch penetration (Figure 3) is a 28-foot diameter cylinder that has a welded ellipsoidal closure cover, which extends about 11 feet inside the steel containment pressure vessel. The double-sided full penetration butt weld attaching the construction hatch cover will be cut and the cover removed during the outage. When the old steam generators have been removed from containment and the new steam generators have been placed within containment, the construction hatch cover will be welded back in place on the end of the cylindrical construction hatch. The removal and welding of the construction hatch cover will be performed under the St. Lucie ASME Section XI Repair and Replacement Program. Repair and replacement activities on the containment vessel will be governed by ASME Section XI, Subsection IWE 1992 Edition, including 1992 Addendum. Unit 1 is currently in the second Ten Year In Service Inspection Interval in accordance with ASME Section XI 1983, including the Summer 1983 Addendum.

In addition to the cutting and welding performed on the construction hatch, all piping, including the reactor coolant system (RCS) hot leg, RCS intermediate leg, main steam, feedwater, blowdown piping, and instrumentation lines will be cut free from the existing steam generator nozzles to facilitate steam generator replacement. When the secondary system piping is welded, attaching them to the new steam generators, the new weld joints for the main steam and feedwater piping will form a part of the containment pressure boundary along with the construction hatch closure cover. The removal and welding of the steam generators and associated piping will be performed under the St. Lucie ASME Section XI program covering replacement activities.

Based on the original requirements of the St. Lucie plant Technical Specifications, which implemented the requirements of 10 CFR 50 Appendix J for periodic leakage rate testing, a periodic Integrated Leak Rate Test (ILRT) would have been scheduled during the Fall 1997 refueling outage (SL1-15). That test would have served to satisfy the Technical Specification leakage test surveillance requirements as well as serve as a test for the construction hatch reinstallation. This testing would also involve other steam generator replacement activities: (a) installation of pads on the containment dome; and (b) closure welds on the secondary side lines where they attach to the new steam generators. However, based on a recent revision to St. Lucie Unit 1 Technical Specifications that adopted 10 CFR 50 Appendix J, Option B, a normal surveillance ILRT is no longer required until May 2003. Accordingly, this evaluation will document acceptable alternatives to an ILRT for steam generator replacement activities in accordance with the provisions of 10 CFR 50 Appendix J, Option B (Reference 3), the St. Lucie Unit 1 Technical Specifications (Reference 2), and Updated FSAR (Reference 1).

FPL evaluated the proposed alternatives for testing of the containment vessel and secondary system piping as acceptable alternatives to a full ILRT, and evaluates compliance with the provisions of 10 CFR 50 Appendix J, Option B, plant Technical Specifications, and the Updated FSAR. This testing approach was evaluated against the criteria of 10 CFR 50.59 to ensure that it does not constitute an unreviewed safety question, or require changes to Technical Specifications.

2.0 LICENSING AND DESIGN BASIS REQUIREMENTS

2.1 TECHNICAL SPECIFICATION REQUIREMENTS

On September 12, 1995, the NRC approved a revision to 10 CFR Part 50, Appendix J (Reference 3), which became effective on October 26, 1995. This revision to 10 CFR 50, Appendix J, added "Option B", which addresses performance based requirements to allow licensees to voluntarily replace the prescriptive requirements of 10 CFR Part 50, Appendix J, with testing requirements based on both previous overall performance and performance of individual components. Performance based testing intervals are based on consideration of the operating history of a component and the potential risk associated with its failure.

On October 28, 1996, FPL submitted a proposed license amendment to the NRC (Reference 4) requesting a modification of Plant Technical Specifications to allow implementation of 10 CFR 50 Appendix J, Option B. The purpose of this amendment to Technical Specifications was to allow Type A, B, and C containment leakage tests to be conducted at intervals determined by performance based criteria. FPL developed the St. Lucie Administrative Procedure ADM 68.01, "Containment Leakage Rate Testing

Program," (Reference 5) which FPL incorporated by reference into plant Technical Specifications. This Administrative Procedure references the use of criteria from Regulatory Guide 1.163 (Reference 6), which specifies a method acceptable to the NRC for complying with 10 CFR 50 Appendix J, Option B, and endorses application of an exception to Regulatory Guide 1.163, which specifically allows the use of either Bechtel Topical Report BN-TOP-1 or ANSI/ANS 56.8-1994 (Reference 8) for Type A leakage rate testing.

Regulatory Guide 1.163 approved the intervals established in NEI 94-01 (Reference 7), which specifies an extension in Type A test frequency to at least one test in 10 years based upon two previous consecutive successful tests. The St. Lucie Unit 1 containment operating history has documented seven successful ILRTs with no associated failures to meet acceptance criteria.

The use of 10 CFR 50, Appendix J, Option B, including the exception to allow use of either Bechtel Topical Report BN-TOP-1 or ANSI/ANS 56.8-1994 for Type A leakage rate testing was subsequently approved by the NRC in License Amendment 149 to the Unit 1 Operating License (Reference 2).

Current St. Lucie Unit 1 Technical Specifications require verification that containment structural integrity and containment leakage rates are acceptable per Technical Specifications 4.6.1.6 and 4.6.1.2, respectively. These specifications rely on inspections and leakage testing performed in accordance with the Containment Leakage Rate Testing Program described in T.S. 6.8.4(h).

2.2 NRC SAFETY EVALUATION (SER)

The original NRC evaluation for containment is documented in Sections 3.8.1 and 6.2.1 of the NRC SER (Reference 9) of March 1976. That SER evaluated the peak containment transient pressure for the worst case primary pipe break, which was calculated as 38.4 psig. The NRC verified this value through their own calculation of peak containment pressure. The NRC concluded that the maximum containment pressure was correctly calculated by the applicant and was below the design internal pressure of 44 psig by an acceptable margin.

2.3 UPDATED FSAR DESIGN BASIS REQUIREMENTS

Containment Vessel

Section 3.8.2.1 of the Unit 1 Updated FSAR addresses the design, design leakage criteria, applicable codes, pressure testing, and post-operational testing and inspection requirements for the steel containment vessel. The containment vessel, including all of its penetrations, is a low leakage shell, which is designed to confine radioactive materials that could be

released following accidental loss of integrity of the reactor coolant pressure boundary. The containment vessel is a circular cylinder with an integral hemispherical dome and ellipsoidal bottom which houses the reactor pressure vessel, the reactor coolant piping and pumps, the steam generators, the reactor coolant pressurizer and pressurizer quench tank, and other branch connections, including the safety injection tanks.

The containment structure in combination with engineered safety features systems ensure that the radiological exposure to the public, resulting from a loss of reactor coolant accident, will remain below the guidelines established in 10 CFR 100. To accomplish this, the containment structure is designed to withstand the maximum calculated peak transient internal pressure and temperature resulting from the worst case design basis accident scenario. The containment vessel and its penetrations are designed in accordance with the ASME Code Section III, Class "B" 1968. As listed in Updated FSAR Table 6.2-1, the containment pressure boundary is designed for a "design internal pressure" (defined in ASME Code Article NE-3112) of 44 psig and coincident temperature of 264°F. The "maximum internal pressure" (ILRT pressure) for the containment vessel is 39.6 psig at a coincident temperature of 264°F.

The calculated loss-of-coolant-accident (LOCA) blowdown transient analysis determined the peak containment pressure and temperature that containment would be exposed to following a design basis accident. This internal pressure, referred to as the "maximum calculated peak internal pressure" in Article NE-3112, is 37.2 psig as identified in Updated FSAR Table 6.2-1. The corresponding design temperature for the maximum calculated peak transient pressure is 258.6°F. This containment peak transient pressure was subsequently reanalyzed and calculated as 37.5 psig for the steam generator replacement, which is discussed below within Section 3.2. This value remains less than the original SER value of 38.4 psig and ILRT pressure specified in the UFSAR of 39.6 psig.

Following completion of the original containment vessel fabrication and post-weld heat treatment, pneumatic testing was performed in accordance with the applicable requirements of the ASME Code and ANS 7.60-1971 to demonstrate the structural integrity and leak-tightness of the completed containment vessel. Following completion of the containment overpressure testing and a second soap bubble visual inspection test, a leakage rate test of the vessel was performed. The Type A integrated leakage rate test was performed in accordance with 10 CFR 50 Appendix J-1973 and ANSI 45.4-1972. The starting test pressure was 41.3 psig to assure a continuous test pressure above 39.6 psig. Hourly readings were taken for 48 hours to monitor the decay of the test pressure. A summary technical report of this testing was generated which documented the analysis and interpretation of the leakage rate test data necessary to demonstrate the acceptability of the leakage rate testing in meeting acceptance criteria.

Subsection 3.8.2.1.14 of the Unit 1 Updated FSAR describes the post-operational testing and inspection of the containment vessel and penetrations as follows:

a. Leakage Rate Testing

"Periodic leakage rate tests of the containment vessel and leak tests of the testable penetrations will be conducted as described in Section 6.2 to verify their continued leak-tight integrity."

b. Surveillance of Structural Integrity

"A steel shell pressure containment vessel, designed, fabricated, inspected and pressure tested in accordance with the ASME Boiler and Pressure Vessel Code and protected by the concrete shield building will offer continued structural integrity over the life of the unit... Therefore it is contemplated that there will be no need for any special in-service test surveillance requirement other than visual inspection of the exposed interior and exterior surfaces of the containment vessel."

Section 6.2.1.4.2 of the Unit 1 Updated FSAR describes the periodic testing and inspection of the containment vessel which states the following:

a. Containment Vessel

"Periodic 'Type A, B and C' leakage rate tests are conducted as applicable in accordance with Appendix J, "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors," 10 CFR 50, and unit Technical Specifications to verify the continued leak-tight integrity of the containment."

Secondary System Design and Testing

Updated FSAR Section 10.1 indicates that the secondary system piping was designed in accordance with the requirements of ANSI/ANS B31.7 Class II, 1969. Post construction pressure testing performed on secondary system piping was consistent with the operating pressures for these systems, which would be significantly in excess of the containment test pressure required by Appendix J.

3.0 ANALYSIS OF EFFECTS ON SAFETY

3.1 ENGINEERING ANALYSIS OF TESTING REQUIREMENTS

Containment Vessel

The removal and reinstallation of the construction hatch closure cover will be performed under the St. Lucie ASME Section XI Program. ASME Section XI, IWE-4000, establishes requirements for repair and replacement activities for the containment

construction hatch cover. These Code requirements impose a number of welding and nondestructive examination criteria, including pressure testing in accordance with 10 CFR 50, Appendix J, as applicable for the type of repairs involved.

ASME Section XI, Subsection IWE-4000 and NEI 94-01 (per 10 CFR 50 Appendix J, Option B) require that any modification, component replacement (that is part of the primary containment boundary), or the resealing of a seal-welded door that was performed after the plant pre-operational leakage rate test shall be followed by either a Type A, Type B, or Type C test. These requirements are designed to ensure that any changes, including removal and replacement, are performed in a manner that will not affect the leak-tightness of the containment boundary.

The original Ebasco construction specification for the construction hatch required that the original design have sufficient material to allow the construction hatch cover to be removed and welded six times during the design life of the plant. The original construction specification for installation of the construction hatch, as well as the current ASME Section XI Code, requires that a volumetric examination be performed for the completed welding of the construction hatch cover.

As mentioned above, the removal and reinstallation of the construction hatch cover will be performed in accordance with ASME Section XI, Subsection IWE, which imposes repair or replacement requirements on this activity (Reference 13), including the volumetric examination of the completed welding. The ASME Section XI Code requires a pressure test after completion of installation of the system or component part. The St. Lucie plant will satisfy the pressure test requirements, as well as those of Appendix J, by performing a pneumatic test of the construction hatch-to-cover weld joint at or above the containment vessel internal design pressure of 39.6 psig. This pressure test for the construction hatch welding will be accomplished by construction of a test chamber around the weld joint between the construction hatch and its cover (Figure 4). The test chamber will be pressurized and a soap bubble test will be utilized to demonstrate that zero leakage occurs in the weld joint. The test pressure of 39.6 psig meets the criteria for testing in accordance with 10 CFR 50 Appendix J, Option B, under the criteria of ANSI/ANS 56.8. ANSI/ANS 56.8 specifies some of the acceptable methods for testing and requires that these tests be performed at a test pressure at least equal to the test pressure of at least 39.6 psig as specified in the UFSAR.

Containment structural integrity will be verified acceptable by the performance of Technical Specification 4.6.1.6. This specification relies on inspections performed in accordance with the Containment Leakage Rate Testing Program that is described in T.S. 6.8.4(h).

<u>10 CFR Part 50, Appendix J</u>

As described below, several NRC and industry documents provide guidance on compliance with 10 CFR 50, Appendix J.

a. Appendix J to 10 CFR 50 (Reference 3) specifies test requirements to ensure that: a) leakage does not exceed allowable rates specified in plant Technical Specifications; and b) integrity of the containment structure is maintained during its Service life. Two options are provided, "either of which can be chosen for meeting the requirements of this Appendix". As previously discussed, St. Lucie Unit 1 has adopted the rules specified in Option B for performance based testing. Option B references NRC Regulatory Guide 1.163 (Reference 6) for guidance on meeting Option B.

b. Regulatory Guide 1.163,"Performance-Based Containment Leak-Test Program," (Reference 6) briefly describes the regulatory position on performance based testing and endorses Nuclear Energy Institute (NEI) document NEI-94-01 Revision 0, (Reference 7) for methods acceptable to comply with the provisions of Option B, subject to four exceptions: a) establishment of test intervals to be based on NEI-94-01 and not the referenced ANSI/ANS 56.8 1994 standard; b) establishment of Type C testing intervals for certain components must be less than 30 months for purge valves and no greater than 60 months for other components; c) visual examination requirements of the containment vessel are required; and d) certain alternatives to Type C tests are not allowed. The first exception provides the method to justify a ten year interval for the ILRT, the remaining three exceptions do not affect this evaluation.

c. NEI-94-01,"Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," (Reference 7) was developed by an NEI Working Group and Task Force for implementation of Appendix J the alternative containment testing rule. The purpose of this document is to assist licensees in the implementation of Option B to minimize the redundant and overlapping engineering and evaluation efforts associated with regulatory requirements. Section 9 of NEI-94-01 describes the methodology for determining Type A, B, and C intervals. Establishment of intervals is documented in the Containment Leakage Rate Testing Program and is not within the scope of this evaluation.

As discussed in NEI 94-01, Section 9.2.4, test requirements for "repairs and modifications that affect the containment leakage integrity require leakage rate testing (Type A testing or local leakage rate testing) prior to returning the containment to operation." The construction hatch cover was tested as part of the original post construction structural integrity test at 49.5 psig and in seven subsequent ILRT's. The full penetration butt weld will be examined in accordance with ASME Section XI using volumetric examination methods. Based on the significant size of the butt weld (1-1/2") and the required non-destructive examination (NDE), no leakage through this welded joint is expected nor will any leakage be acceptable.

In accordance with NEI 94-01, Section 9.2.4, "local leakage rate testing" will be performed. The local leakage testing will consist of installation of a leak chase, pressurizing to at least 39.6 psig, and application of a soap bubble solution (snoop) to test the butt weld from the opposite side (Figure 4). This method of testing is advantageous because it definitively tests the specific weld in question with an acceptance criterion of "zero leakage."

d. Generally, a soap bubble (snoop) test is not considered an acceptable method of quantifying leakage, as discussed in Reference 10. This is because the correlation of soap bubbles to a leakage rate is not well defined. The exception to this is if the leakage criterion is set to "zero leakage", then the method provides quantifiable results - no leakage at all (Reference 10). Although leakage is acceptable in terms of the overall containment testing program (i.e., sum of all leakage paths), this particular test will have a strict criterion of "zero leakage." Based on the above, the use of a local leak chase meets the requirements 10 CFR 50 Appendix J, Option B.

Secondary System Testing

In terms of containment isolation features, the secondary side system consisting of main feedwater and main steam piping are normally considered part of the passive closed system inside containment which is an extension of the containment structure. During an ILRT, the secondary side system is vented to atmosphere in order to provide a pressure differential of 39.6 psig or greater from the containment atmosphere to the outside atmosphere through the main steam piping system.

The secondary side of the replacement steam generators have been hydro tested to 1.25 x 1000 = 1250 psia as part of the fabricator's program. Once the piping to and from the steam generators is welded in place, the welds will be examined and a pressure test performed in accordance with ASME Section XI. This test will be at pressures approximately twenty times higher than that of an ILRT. Although the leak test is in a direction reverse to that of a LOCA environment, the test is acceptable because of the high pressure during the leak test. Section 9.2.1 of NEI 94-01 and Section 6.2 of ANSI/ANS 56.8 allow licensees to utilize reverse testing if justified. For these welded connections, no leakage will be acceptable. This is consistent with interpretations made by other plants. Accordingly, as discussed in Reference 11, this testing is considered acceptable to meet 10 CFR 50 Appendix J.

Welded Attachment Pads

For pads welded to the containment vessel, test requirements subsequent to repairs and modifications are addressed in NEI 94-01, Section 9.2.4. For welds of attachments to the surface of steel pressure-retaining boundary, Type A (ILRT) testing may be deferred to the next scheduled ILRT. Accordingly, the installation of pads to the containment without performance of an ILRT this outage meets the requirements of Appendix J testing.

3.2 STEAM GENERATOR REPLACEMENT EVALUATION

The evaluation of the Steam Generator Replacement Project (SGRP) activities is addressed in the SGRP Stand Alone Safety Evaluation. This report contains the two separate 50.59 evaluations for SGRP activities. FPL's Evaluation (Reference 14) contains an evaluation of construction activities and plant changes as a result of the implementation effort. The second FPL evaluation (Reference 15) contains an evaluation of the replacement steam generator. These two evaluations do not address the specific type of testing that would be required to satisfy Appendix J requirements. The use of a local leak rate test for the containment construction hatch as discussed within this evaluation will not affect the conclusions previously reached in the two SGRP evaluations identified above.

A containment transient response evaluation for the worst case LOCA was performed as part of the evaluations performed for the SGRP. The replacement steam generators contain slightly more primary side mass than the original steam generators. This results in a slight increase in the peak containment pressure following the worst case LOCA. Calculations performed with an increased primary side mass and energy corresponding to the replacement steam generators show that the blowdown results in a peak containment pressure of 37.5 psig. This is an increase of 0.3 psig over the current containment analysis of record (37.2 psig) shown in Updated FSAR Table 6.2-1. The SGRP peak containment transient pressure of 37.5 psig is less than the peak pressure of 38.4 psig originally accepted by the NRC in the original plant SER (Reference 9) and remains below the originally specified ILRT test pressure of at least 39.6 psig. Therefore, the proposed use of a local leak rate test pressure of 39.6 psig will still bound the new containment peak transient pressure calculated for the Steam Generator Replacement Project.

4.0 EFFECT ON TECHNICAL SPECIFICATIONS

The construction hatch cover was tested as part of the original post construction structural integrity test at 1.25 X 39.6 = 49.5 psig and in seven subsequent ILRTs. The full penetration butt weld will be examined in accordance with ASME Section XI using volumetric examination methods. Although some leakage would be acceptable in terms of the overall containment testing program (that is, the sum of all leakage paths), the containment leakage test proposed for the reinstallation of the construction hatch will have a strict criterion of zero leakage in order to satisfy the repair and replacement requirements. Based on the above, the use of a local leak chase meets the requirements of Appendix J testing.

Consequently, the design margins identified for containment in the St. Lucie Containment Leakage Rate Testing Program, referenced in plant Technical Specifications (Reference 2), will be restored after the weld repairs are completed on the construction hatch cover and secondary system piping within containment. The proposed testing activities will have no effects on plant Technical Specifications, and the testing program using the pressure of 39.6 psig for the containment vessel will not require any changes to Technical Specifications in order to implement the testing approach outlined in this evaluation.

5.0 CONCLUSIONS

This evaluation serves to document the testing requirements from 10 CFR 50, Appendix J, which are applicable to the Unit 1 steel containment pressure vessel following the replacement of the two steam generators scheduled for the Fall 1997 refueling/steam generator replacement outage (SL1-15). The testing approach for the Unit 1 containment that is discussed above has been evaluated against the facility change criteria established by 10 CFR 50.59 to ensure compliance with NRC regulatory criteria.

This evaluation concludes that the proposed methodology for testing the containment vessel and secondary system piping is an acceptable alternatives to a full ILRT and complies with the provisions of 10 CFR 50, Appendix J, plant Technical Specifications, and the Updated FSAR. The testing approach as evaluated against the criteria of 10 CFR 50.59 determined that the proposed testing approach does not involve an unreviewed safety question, or require changes to Technical Specifications. Therefore, this testing approach can be implemented without prior NRC approval pursuant to the requirements of 10 CFR 50.59.

6.0 REFERENCES

1. St Lucie Unit 1 Updated FSAR, Section 3.8.2, "Containment Structure," and Section 6.2.1,"Containment Functional Design."

2. St Lucie Unit 1 Technical Specifications 3.6.1.1, 3.6.1.2, 3.6.1.3, 3.6.1.6, 3.6.6.3, and 6.8.4, Amendment No.149, transmitted by NRC letter dated February 10, 1997.

3. Title 10 Code of Federal Regulations (CFR) Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Effective Date October 26, 1995.

4. FPL letter L-96-244, "St. Lucie Unit 1, Proposed License Amendment - Implementation of 10 CFR 50 Appendix J, Option B," dated October 28, 1996.

5. St. Lucie Plant Administrative Procedure ADM 68.01, "Containment Leakage Rate Testing Program," Revision 0.

6. NRC Regulatory Guide 1.163, "Performance Based Containment Leak-Test Program," dated September 1995.

7. Nuclear Energy Institute Guideline NEI 94-01, "Industry Guideline For Implementing Performance-Based Option Of 10 CFR Part 50, Appendix J," Revision 0, dated July 26, 1995.

8. ANSI/ANS-56.8-1994, "American National Standard For Containment System Leakage Testing Requirements."

9. AEC St. Lucie 1 Safety Evaluation (SER), "Safety Evaluation of the St. Lucie Plant Unit No.1," dated November 8, 1974, May 9, 1975, and March 1, 1976.

10. USNRC, Nuclear Reactor Regulation, letter (C. Y. Shiraki, Div Reactor Projects - III/IV) to Commonwealth Edison Co. (D. L. Farrar), "Issuance of Exemptions From the Requirements of 10 CFR Part 50, Appendix J - Zion Nuclear Power Station, Unit Nos. 1 and 2," dated December 28, 1995. (Page 6 Item 1)

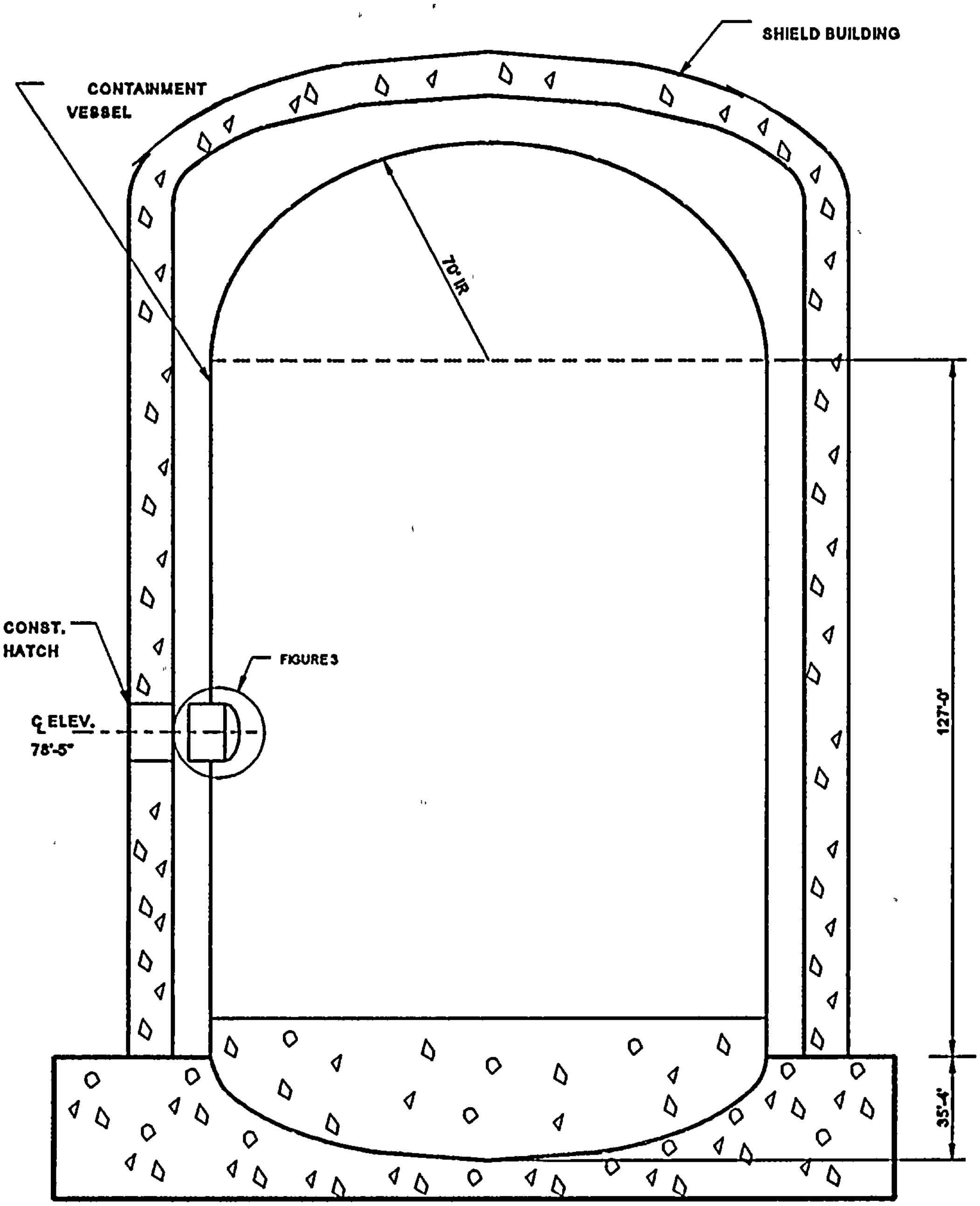
11. NRC Nuclear Reactor Regulation, Internal Memorandum (G. M. Holahan, Div. of System Technology, to J. A. Zwolinski, Division of Reactor Projects III, IV, and V), "Region III Request for Position on Leakage Out of Containment Through PWR Steam Generator Secondary Side During Containment Integrated Leakage Rate Test," dated February 1, 1991. (Page 3 Response to Question 3)

12. NRC Information Notice 97-29, "Containment Inspection Rule," dated May 30, 1997.

13. SGRP Safety Evaluation ENG-PSL-SEMP-94-034, "Code Reconciliation for SGRP," Revision 3, dated September 3, 1997.

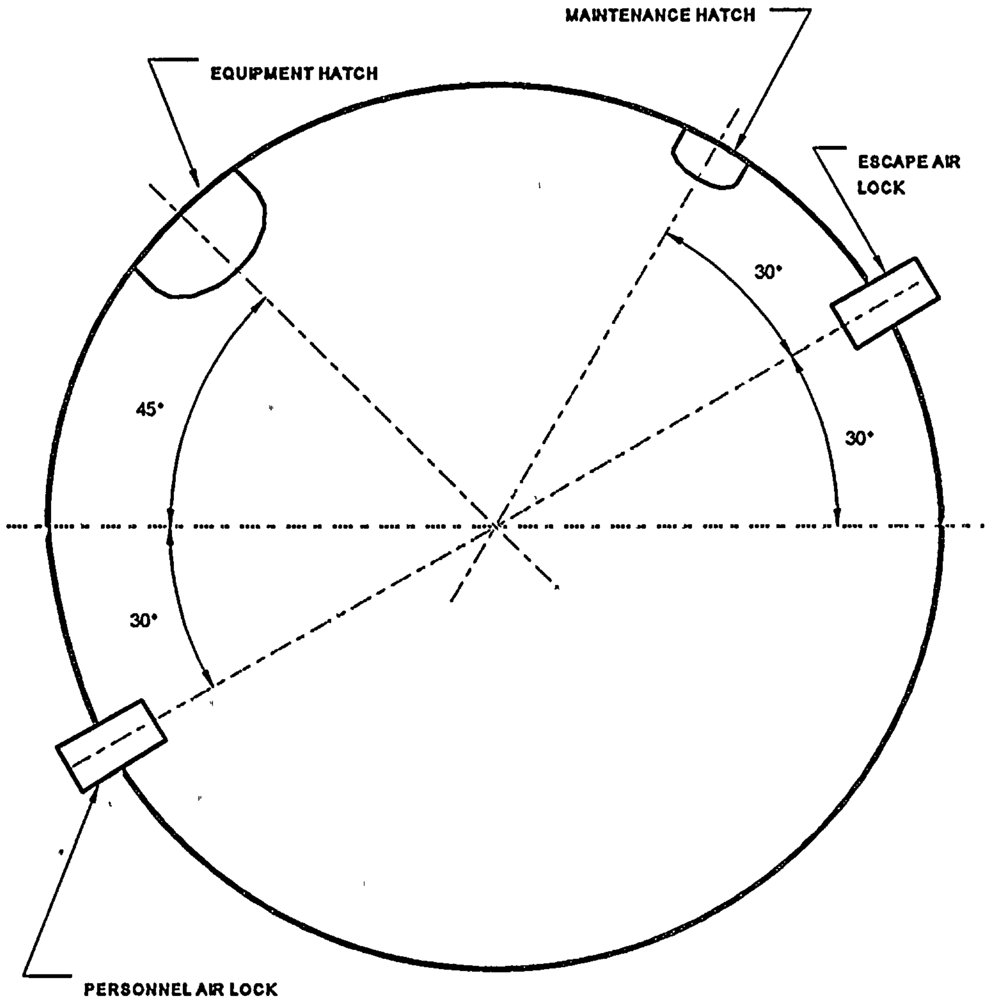
14. SGRP Safety Evaluation ENG-PSL-SEMP-94-033, "Safety Evaluation for the PSL-1 Steam Generator Replacement Project," Revision 6, dated July 17, 1997.

15. SGRP Safety Evaluation ENG-PSL-SEMP-94-026, "Stand Alone Safety Evaluation (SASE) Volume 2 - Steam Generator Equivalency Report (SGER)," Revision 1, dated June 3, 1997.



CROSS SECTION OF CONTAINMENT VESSEL
AND SHIELD BUILDING

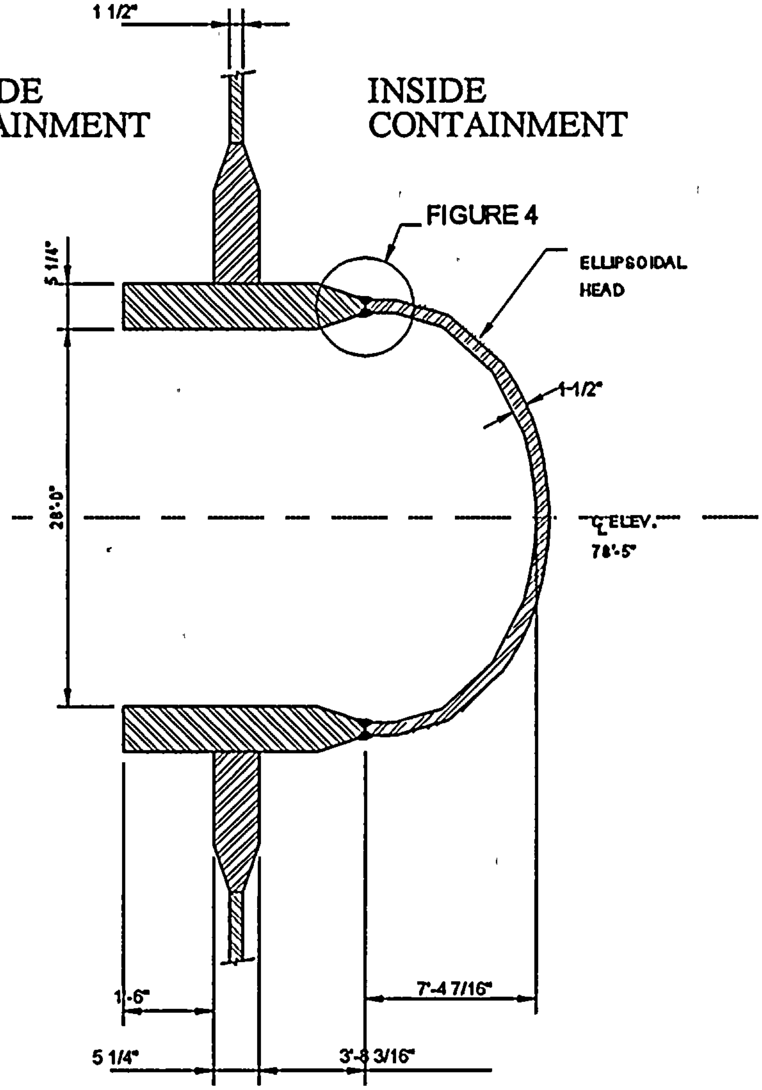
Figure 1



PLAN VIEW OF UNIT 1
CONTAINMENT AND MAJOR PENETRATIONS

DWG

Figure 2



CONSTRUCTION HATCH
THROUGH CONTAINMENT VESSEL

Figure 3

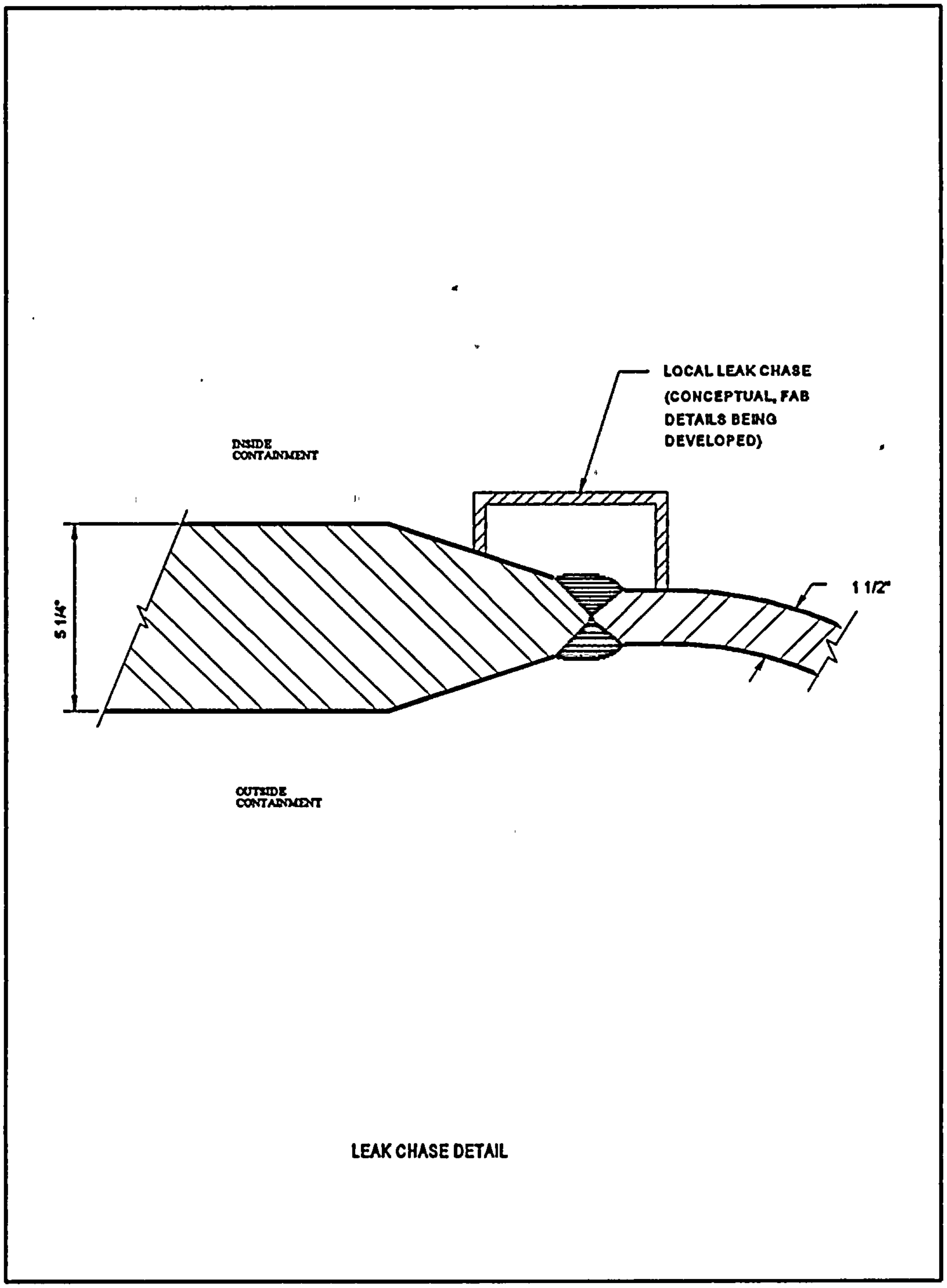

LOCAL LEAK CHASE
(CONCEPTUAL, FAB
DETAILS BEING
DEVELOPED)
INSIDE
CONTAINMENT
5 1/4"
1 1/2"
OUTSIDE
CONTAINMENT
LEAK CHASE DETAIL

Figure 4