DOCKET FILES

June 3, 1991

Docket Nos. 50-361
and 50-362

Mr. Harold B. Ray
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Southern California Edison Co.
Irvine Operations Center
23 Parker Street
Irvine, California 92718

Mr. Gary D. Cotton
Senior Vice President
Engineering and Operations
San Diego Gas & Electric Co.
101 Ash Street
San Diego, California 92112

Gentlemen:

SUBJECT: ISSUANCE OF AMENDMENT NOS. 93 AND 83 TO FACILITY OPERATING LICENSE NOS. NPF-10 AND NPF-15 FOR THE SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NOS. 2 AND 3 (TAC NOS. 80099 AND 80100)

The Commission has issued the enclosed Amendment Nos. 93 and 83 to Facility Operating License Nos. NPF-10 and NPF-15 for San Onofre Nuclear Generating Station, Unit Nos. 2 and 3. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated April 8, 1991, designated by you as PCN-351.

These amendments revise Surveillance Requirement 4.6.1.2.a and the associated Bases to permit the third Type A test of each 10-year inservice interval to be conducted during a separate plant outage from the 10-year plant Inservice Inspection.

A copy of our related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Original signed by:

Lawrence E. Kokajko, Project Manager
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 93 to NPF-10
2. Amendment No. 83 to NPF-15
3. Safety Evaluation

cc w/enclosures:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 93
License No. NPF-10

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Southern California Edison Company, San Diego Gas and Electric Company, the City of Riverside, California, and the City of Anaheim, California (licensees) (the licensee) dated April 8, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-10 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 93 , are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and must be fully implemented no later than 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James E. Dyer

James E. Dyer, Director
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 3, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 93

FACILITY OPERATING LICENSE NO. NPF-10

DOCKET NO. 50-361

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
3/4 6-2	3/4 6-2
B 3/4 6-1	B 3/4 6-1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 83
License No. NPF-15

1. The Nuclear Regulatory Commission (the Commission) has found that:

A. The application for amendment by Southern California Edison Company, San Diego Gas and Electric Company, the City of Riverside, California, and the City of Anaheim, California (licensees) (the licensee) dated April 8, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-15 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 83 , are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and must be fully implemented no later than 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James E. Dyer

James E. Dyer, Director
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 3, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 83

FACILITY OPERATING LICENSE NO. NPF-15

DOCKET NO. 50-362

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
3/4 6-2	3/4 6-2
B 3/4 6-1	B 3/4 6-1

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* listed in Sections A, B and C of Table 3.6-1 not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or automatic valves secured** in their positions, except as provided in Table 3.6-1 of Specification 3.6.3.

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

c. After each closing of each penetration subject to Type B testing, except containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at P_a 55.7 psig and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

*Except valves, blind flanges, and automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

**Locked, sealed or otherwise prevented from unintentional operation.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. Less than or equal to L_a, 0.10 percent by weight of the containment air per 24 hours at P_a, 55.7 psig, or

2. Less than or equal to L_t, 0.05 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 27.9 psig.

b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable, and the combined leakage rate for all penetrations and valves subject to Type B and C tests to less than 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4 - 1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at either P_a (55.7 psig) or at P_t (27.9 psig) during each 10-year service period. Prior to the Type A tests a visual inspection shall be conducted in accordance with Specification 4.6.1.6 to demonstrate the containment structural integrity.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR 50. The following exception, however, applies: The third Type A test of each 10-year inservice interval, need not be conducted when the unit is shutdown for the 10-year plant inservice inspection.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 5.0 psig, 2) the containment peak pressure does not exceed the design pressure of 60 psig during LOCA or steam line break conditions, and 3) the assumptions used for the initial conditions of the LOCA safety analysis remain valid.

The maximum peak pressure expected to be obtained from a LOCA or steam line break event is 55.7 psig. The limit of 1.5 psig for initial positive containment pressure will limit the total pressure to 57.2 psig which is less than the design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment average air temperature ensures that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a steam line break accident.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 55.7 psig in the event of a steam line break accident. The measurement of containment tendon lift off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, the chemical and visual examination of the sheathing filler grease, and the Type A leakage tests are sufficient to demonstrate this capability.

The surveillance requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Proposed Revision 3 to Regulatory Guide 1.35, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," April 1979; and Proposed Regulatory Guide 1.35.1, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," April 1979.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 93 AND 83 TO

FACILITY OPERATING LICENSE NOS. NPF-10 AND NPF-15

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NOS. 2 AND 3

DOCKET NOS. 50-361 AND 50-362

1.0 INTRODUCTION

By letter dated April 8, 1991, Southern California Edison Company (SCE or the licensee) requested changes to the Technical Specifications (TS) for Facility Operating License Nos. NPF-10 and NPF-15 that authorize operation of San Onofre Nuclear Generating Station, Unit Nos. 2 and 3 in San Diego County, California. The licensee has requested to revise TS 3/4 6.1.2, "Containment Leakage." Specifically, the licensee proposed to amend TS Surveillance Requirement 4.6.1.2.a and the associated Bases to permit the third Type A test of each 10-year inservice interval to be conducted during a separate plant outage from the 10-year plant Inservice Inspection.

2.0 EVALUATION

Appendix J requires that a set of three Type A tests be performed during each 10-year service period with the third test being conducted when the plant is shut down for the 10-year plant inservice inspection. The proposed TS change would eliminate the requirement of conducting the third Type A test of a 10-year service period during the shutdown for the 10-year unit inservice inspection.

The purpose for requiring the third Type A test during shutdown for the 10-year plant inservice inspection is to assure that the three Type A tests are not bunched together during the first 90 months of the 10-year operation cycle. Requiring the third Type A test during the 10-year plant inservice inspection assures that the three Type A tests are evenly spaced over the 10-year interval.

The licensee has requested an amendment that would eliminate conducting the third Type A test in a 10-year service period during the unit shutdown for the 10-year inservice inspection (ISI). For example, the third Type A test for

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the first 10-year service period for San Onofre 2 is scheduled for the San Onofre Unit 2 Cycle 6 refueling outage in August 1991. The licensee contends that, because the 10-year ISI has been extended beyond 1991, (Cycle 7 refueling outage) the inspection is not necessary for the Unit 2 Cycle 6 refueling outage and, therefore, must be uncoupled from the third Type A test in each 10-year service period which is required by Appendix J. (A similar situation exists for Unit 3). To perform a fourth Type A test during the same shutdown as the 10-year plant ISI (Cycle 7 refueling outage) would only satisfy the Technical Specification requirement to perform a Type A test during the same shutdown for the 10-year plant ISI. Additionally, performing a fourth containment ILRT, for the sole purpose of being done during the same outage as the 10-year ISI, would not necessarily enhance the purpose, or provide further assurance of containment integrity, above that which has already been demonstrated. Moreover, the licensee intends to conduct the three Type A tests at 40±10 month intervals during each 10-year service period.

Additionally, not extending the inservice inspection would impose hardship on the licensee with little or no increase in the level of quality of safety. This inspection is not related to containment integrity requirements of Appendix J. The purpose of the Appendix J test program is to ensure that leakage through the primary reactor containment and systems and the components penetrating primary containment does not exceed allowable leakage rate values. The purpose of the ASME Code Section XI inservice inspection program is to ensure that structural integrity of Class 1, 2, and 3 components is maintained in accordance with ASME Code requirements. Therefore, the proposed separation has no safety consequences because the requirements on containment integrity in Appendix J and the TSs, and on structural integrity of Class 1, 2, and 3 components in the ASME Code are not being changed by the proposed change to TS 4.6.1.2.a.

The staff has considered the amendment request for uncoupling the third Type A test of each 10-year service period from the 10-year unit ISI and concludes it is justified on the grounds that the third Type A test within each 10-year service period and the 10-year ISI may be scheduled separately, and the safe operation of San Onofre Unit Nos. 2 and 3 does not require that the two tests be conducted in the same outage. The licensee is still required to conduct the 10-year ISI in accordance with Section XI of the ASME Code.

Therefore, based upon the information presented above, the staff finds the amendment request acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the California State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Lawrence E. Kokajko

Date: June 3, 1991

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* listed in Sections A, B and C of Table 3.6-1 not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or automatic valves secured** in their positions, except as provided in Table 3.6-1 of Specification 3.6.3.

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

c. After each closing of each penetration subject to Type B testing, except containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at P_a 55.7 psig and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

* Except valves, blind flanges, and automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

**Locked, sealed or otherwise prevented from unintentional operation.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. Less than or equal to L_a, 0.10 percent by weight of the containment air per 24 hours at P_a, 55.7 psig, or

2. Less than or equal to L_t, 0.05 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 27.9 psig.

b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable, and the combined leakage rate for all penetrations and valves subject to Type B and C tests to less than 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4 - 1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at either P_a (55.7 psig) or at P_t (27.9 psig) during each 10-year service period. Prior to the Type A tests a visual inspection shall be conducted in accordance with Specification 4.6.1.6 to demonstrate the containment structural integrity.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR 50. The following exception, however, applies: The third Type A test of each 10-year inservice interval, need not be conducted when the unit is shutdown for the 10-year plant inservice inspection.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 5.0 psig, 2) the containment peak pressure does not exceed the design pressure of 60 psig during LOCA or steam line break conditions, and 3) the assumptions used for the initial conditions of the LOCA and safety analysis remain valid.

The maximum peak pressure expected to be obtained from a LOCA or steam line break event is 55.7 psig. The limit of 1.5 psig for initial positive containment pressure will limit the total pressure to 57.2 psig which is less than the design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment average air temperature ensures that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a steam line break accident.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 55.7 psig in the event of a steam line break accident. The measurement of containment tendon lift off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, the chemical and visual examination of the sheathing filler grease, and the Type A leakage tests are sufficient to demonstrate this capability.

The surveillance requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Proposed Revision 3 to Regulatory Guide 1.35, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," April 1979; and Proposed Regulatory Guide 1.35.1, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," April 1979.