Attachment 1

St. Lucie Unit 2 Marked-Up Technical Specification Pages

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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 1 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** 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 deactivated automatic valves secured in their positions, except as provided in Table 3.6-2 of Specification 3.6.4.

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 ~~43.4~~ 41.8 psig and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

*In MODES 1 and 2, the RCB polar crane shall be rendered inoperable by locking the power supply breaker open.

**Except valves, blind flanges, and deactivated 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.

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, 0.50 percent by weight of the containment air per 24 hours at P_a, 41.8 ~~43.4~~ psig, or

2. Less than or equal to L_t, 0.35 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 20.9 ~~21.7~~ psig.

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

c. A combined bypass leakage rate of less of than or equal 0.12 L_a for all penetrations identified in Table 3.6-1 as secondary containment bypass leakage paths 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, or (c) with the combined bypass leakage rate exceeding 0.12 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 or equal to 0.60 L_a, and the bypass leakage rate to less than or equal to 0.12 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

SURVEILLANCE REQUIREMENTS (Continued)

shutdown at either P_a, ~~43.4~~ 41.8 psig or at P_t, ~~21.7~~ 20.9 psig during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet either .75 L_a or .75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either .75 L_a or .75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either .75 L_a or .75 L_t at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within 0.25 L_a or 0.25 L_t.

2. Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25% of the total measured leakage rate at P_a, ~~43.4~~ 41.8 psig or P_t, ~~21.7~~ 20.9 psig.

d. Type B and C tests shall be conducted with gas at P_a, ~~43.4~~ 41.8 psig at intervals no greater than 24 months except for tests involving:

1. Air locks,

2. Purge supply and exhaust isolation valves with resilient material seals.

e. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirements 4.6.1.7.3 and 4.6.1.7.4.

SURVEILLANCE REQUIREMENTS (Continued)

f. The combined bypass leakage rate shall be determined to be less than or equal to 0.12 L_a by applicable Type B and C tests at least once per 24 months except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a, ~~43.4~~ 41.8 psig during each Type A test.

g. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

h. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, ~~43.4~~ 41.8 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one containment air lock door inoperable*:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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*If the inner air lock door is inoperable, passage through the OPERABLE outer air lock door is permitted to effect repairs to the inoperable inner air lock door. No more than one airlock door shall be open at any time.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying the seal leakage is < 0.01 L_a as determined by precision flow measurement when the volume between the door seals is pressurized to greater than or equal to:

   1. For the personnel air lock, greater than or equal to P_a, 41.8 ~~43.4~~ psig for at least 15 minutes if not tested with the automatic tester.

   2. For the emergency air lock, greater than or equal to 41.8 ~~43.4~~ psig for at least 15 minutes.

b. By conducting overall air lock leakage tests at not less than P_a, 41.8 ~~43.4~~ psig, and verifying the overall air lock leakage rate is within its limit:

   1. At least once per 6 months,# and

   2. Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.*

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

#The provisions of Specification 4.0.2 are not applicable.

*This constitutes an exemption to Appendix J of 10 CFR 50.

Attachment 2

Safety Analysis

Introduction

A change is proposed to revise the St. Lucie Unit 2 Technical Specifications to change the Integrated Leak Rate Testing (ILRT) containment test pressure (P_a) to be equal to that of the Design Basis Accident (i.e. large break loss of coolant accident) instead of main steam line break.

Discussion

The proposed change to the St. Lucie Unit 2 Technical Specifications revises the calculated peak containment internal pressure value used for ILRT from that of the calculated maximum pressure resulting from the Main Steam Line Break (MSLB) event to the maximum calculated pressure for the Loss of Coolant Accident (LOCA) event. Use of the calculated containment pressure from the LOCA is consistent with the values used at other Pressurized Water Reactor power plants for ILRT.

The basis for the surveillance testing for measuring leak rates at the test pressure are contained in Appendix J of 10 CFR 50 as referenced in the Technical Specification Bases and Surveillance Requirements. Appendix J to 10 CFR 50, Section II., Explanation of Terms, Item I defines (P_a) (psig) as: "....the calculated peak containment internal pressure related to the design basis accident and specified either in the technical specification or associated bases."

The introduction to Appendix J of 10 CFR 50 states, in part, that: "... primary reactor containments shall meet the containment leakage test requirements set forth in this appendix." The term primary reactor containment is defined in 10 CFR 50, Appendix J, Section II., Explanation of Terms, Item A as meaning the structure or vessel that encloses the components of the reactor coolant pressure boundary, and serves as an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.

The St. Lucie Unit 2 Technical Specifications Bases define the purpose of primary containment integrity as ensuring that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analysis. Subsection 6.2.1.1.1 of the St. Lucie Unit 2 Updated Final Safety Analysis Report (UFSAR) states that the containment is designed to provide protection to the

public from the consequences of a loss of coolant accident.

Criterion 16 of 10 CFR 50 Appendix A, Containment design, states in part that: "Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment ..."

Criterion 50 of 10 CFR 50 Appendix A, Containment design basis states in part that: "The reactor containment heat removal system shall be designed so that the containment structure, including access openings, penetrations, and the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss of coolant accident."

ANSI/ANS-56.8-1981 - Containment System Leakage Testing Requirements defines the primary reactor containment system as "the design feature, which acts as the principle leakage barrier, after the reactor coolant pressure boundary, to prevent the release under all conditions of design, of quantities of radioactive material that would have undue radiological effect on the health of the public."

These criteria for containment design and the definition of the purpose of the Integrated Leak Rate Test (ILRT) as presented in Appendix J of 10 CFR 50 define the purpose of the ILRT, to demonstrate that the reactor containment will maintain its integrity and not exceed the design leakage rate during the design basis accident which could release radioactivity to the environment. The accident which is considered for this event, as shown above, is the loss of coolant accident, since the potential radionuclide releases from this event are much more severe than for a main steam line break. Comparing St. Lucie Unit 2 UFSAR Table 15.0-4a and Table 15.6.6-12 shows that the LOCA thyroid dose is 675 times the MSLB thyroid dose and 2.6×10^5 times the MSLB whole body dose at the site boundary for the 0 to 2 hour dose calculation. The total event comparison shows an even larger difference.

The calculated pressure resulting from a loss of coolant accident should, therefore, be the accident peak pressure (P_a) used for the reactor containment testing requirements. The peak containment pressure calculated for the loss of coolant accident is given in St. Lucie Unit 2 UFSAR Table 6.2-2 as 41.8 psig. Therefore the value of (P_a) in the plant Technical Specifications should be revised to 41.8 psig.

Technical Specification 3/4 6.1 currently defines the value of containment pressure reached during an accident condition (P_a) as 43.4 psig. As defined in 10 CFR 50 Appendix J and in the St. Lucie Unit 2 Technical Specifications, the value of (P_a) also determines

the allowable leakage rate (L_a), viz., 0.5 weight percent of the containment air per 24 hours at a pressure (P_a). The value of (P_a) also determines a reduced test pressure, (P_t) (equal to one-half (P_a)) and its corresponding allowable leakage rate (L_t) 0.35 weight percent of the containment air per 24 hours at a pressure (P_t). These parameters are used throughout the Limiting Conditions for Operation and the Surveillance Requirements of 3/4.6.1.1, 3/4.6.1.2., and 3/4.6.1.3.

In part III.A.4 of 10 CFR 50, Appendix J, the test pressure for reduced pressure tests (P_t) is not less than 0.50 (P_a). Therefore, the use of a new value of (P_a) changes the value of (P_t) used in part a.2 of Technical Specification 3.6.1.2. to 20.9 psig.

Use of (P_a) as defined above for the LOCA versus the MSLB does not affect the Technical Specifications governing containment integrity: Technical Specification 3/4 6.1.4, Internal Pressure; 3/4.6.1.5, (containment) Air Temperature; and 3/4 6.1.6, Containment Vessel Structural Integrity; are still in effect and the maximum containment design pressure of 44.0 psig is not exceeded by these changes. Furthermore, Technical Specification 3/4.6.1.6 considers the higher MSLB pressure (43.4 psig) in making the determination of containment structural integrity. St. Lucie Unit 2 UFSAR Section 3.8.2.7. states that an overload pressure test of 50 psig was conducted on the St. Lucie Unit 2 containment vessel, verifying its ability to accept an MSLB. This section also discusses the initial leak rate test, at the design pressure of 44 psig, conducted on the containment vessel. These tests verified the vessel's ability to accept design conditions.

Testing at the lower pressure will not result in any changes in the St. Lucie Unit 2 accident analyses nor will it result in significant changes in the radiological doses calculated as a result of these accident analyses. The radiological consequences for a main steam line break inside containment remain bounded by the radiological consequences of a main steam line break outside containment discussed in the St. Lucie Unit 2 UFSAR Section 15.1.5.1.

Attachment 3

Determination of No Significant Hazards Consideration

The standards used to arrive at a determination that a request for amendment involves no significant hazards consideration are included in the Commission's regulation, 10 CFR 50,92, which state that no significant hazards considerations are involved if the operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed as follows:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment change does not involve a significant increase in the probability or consequences of an accident previously evaluated because the primary reactor containment is designed to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from a loss of coolant accident. This meets the requirement of 10 CFR 50 Appendix A Criterion 50 for the containment to retain its integrity during a loss of coolant accident. Satisfactory leak rate testing at the value of the peak calculated containment pressure following a loss of coolant accident provides the assurance required by the design basis 3/4.6.1.1 - Containment Integrity, that any release of radioactive materials will be restricted to that assumed in the safety analysis. The probability or consequences of an accident are not significantly increased because there is no change to the containment design basis nor the ability of the containment to perform the required function of preventing the release of radioactivity to the environment.

(2) Use of the modified specification would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The use of this modified specification cannot create the possibility of a new or different kind of accident from any previously evaluated since the design features of the primary reactor containment as required by Criterion 16 of 10 CFR 50 Appendix A are not altered. The testing at the calculated

peak accident pressure during a loss of coolant accident demonstrates that the reactor containment and associated systems provide an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment. No new failure mode is introduced due to the change of test pressure, since the assurance of integrity at the calculated accident pressure is maintained by testing at the value.

(3) Use of the modified specification would not involve significant reduction in a margin of safety.

The existing test pressure using the calculated peak pressure value for the main steam line break (MSLB) tests the containment to a higher pressure than is required. This could be considered as providing a greater margin of safety since testing at approximately 104% of calculated accident peak value insures that the actual pressure stress is accommodated. However, there is no requirement for the test pressure to be higher than the calculated design basis accident peak pressure. For purposes of protection from radiological releases, the peak loss of coolant accident (LOCA) pressure is appropriate and the peak MSLB pressure is not appropriate. As shown in St. Lucie Unit 2 Tables 15.0-4a and 15.6.6-12 the radiological consequences of a LOCA produce a minimum of 675 times the site boundary dosage as the inside containment MSLB. Article NE-3112.1 - Design Pressure, of 1986 ASME Boiler and Pressure Vessel Code, states that the design internal pressure shall not be less than 100% of the maximum containment internal pressure under conditions for which the containment function is required, i.e., LOCA event. Prior to being placed in service, the containment was successfully tested by being pressurized to 50 psig which verified the design pressure of 44 psig.

Therefore, the modified specification which establishes the peak accident containment pressure corresponding to the actual accident requiring containment integrity does not reduce any margin of safety. The containment vessel structural integrity requirement of Technical Specification 3/4.6.1.6 is met in that the design pressure of 44 psig is greater than 100% of the 41.8 psig which is the condition for which the containment function is required. Furthermore, this Technical Specification considers the higher MSLB pressure (43.4 psig) in the determination of containment structural integrity.

Based on the above, we have determined that the proposed amendment does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the probability of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety; and therefore does not involve a significant hazards consideration.

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