TSI 57

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 manual valves, blind flanges, or deactivated automatic valves secured in their closed 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 the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a, 48.1 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.1.d for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a;~~ Deleted

d. By performing containment leakage rate testing ~~, except for containment air locks,~~ in accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~ the Containment Leakage Rate Testing Program; and

e. By verifying containment structural integrity in accordance with the Containment Tendon Surveillance Program of Specification 6.8.5.c.

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* 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.

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CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with~~:~~

~~a.~~ ~~B~~oth doors closed except when the air lock is being used for normal transit entry and exits 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, 48.1 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, and

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.

CONTAINMENT SYSTEMS

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 that the seal leakage is less than 0.005 L_a as determined by precision flow measurements when measured for at least 30 seconds with the volume between the seals at a constant pressure of greater than or equal to 10 psig;~~

~~b. By conducting overall air lock leakage tests at not less than P_a, 48.1 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.~~ b. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

a. By verifying leakage rates in accordance with the Containment Leakage Rate Testing Program; and

~~#The provisions of Specification 4.0.2 are not applicable.~~
~~*This represents an exemption to Appendix J, Paragraph III.D.2.(b)(ii), of 10 CFR Part 50.~~

PROCEDURES AND PROGRAMS (Continued)

f. Radiological Environmental Monitoring Program (Continued)

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

INSERT A

6.8.5 The following programs, relocated from the Technical Specifications to FSAR Chapter 16, shall be implemented and maintained:

a. Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the WASTE GAS HOLDUP SYSTEM, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

1. The limits for concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM and a surveillance program to ensure the limits are maintained.

2. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY in the event of an uncontrolled release of the tanks' contents, consistent with Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases due to Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

3. A surveillance program to ensure that the quantity of radioactivity contained in the following outdoor liquid radwaste tanks, that are not surrounded by liners, dikes, or walls capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste system, is less than the amount that would result in concentrations less than the limits of 10 CFR Part 20.1 -20.602. Appendix B (redesignated at 56FR23391, May 21, 1991) at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA, in the event of an uncontrolled release of the tanks' contents:

   a. Reactor Makeup Water Storage Tank,
   b. Refueling Water Storage Tank,
   c. Condensate Storage Tank, and
   d. Outside temporary tanks, excluding demineralizer vessels and the liner being used to solidify radioactive waste.

INSERT A

g. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 48.1 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of the containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$;

   2) For each door, leakage rate is $\leq 0.005\ L_a$ when pressurized to $\geq$ 10 psig.

The provisions of Technical Specification 4.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

The provisions of Technical Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

ATTACHMENT 2

TECHNICAL SPECIFICATION CHANGES

(RE-TYPED)

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 manual valves, blind flanges, or deactivated automatic valves secured in their closed 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. Deleted.

d. By performing containment leakage rate testing in accordance with the Containment and Leakage Rate Testing Program.

e. By verifying containment structural integrity in accordance with the Containment Tendon Surveillance Program of Specification 6.8.5.c.

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* 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 AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with both doors closed except when the air lock is being used for normal transit entry through the containment, then at least one air lock door shall be closed.

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, and

   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.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By verifying leakage rates in accordance with the Containment Leakage Rate Testing Program; and

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

PROCEDURES AND PROGRAMS (Continued)

f. Radiological Environmental Monitoring Program (Continued)

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

g. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 48.1 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of the containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and C tests and $\leq$ 0.75 L_a for Type A tests;

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq$ 0.05 L_a when tested at $\geq P_a$;

2) For each door, leakage rate is $\leq$ 0.005 L_a when pressurized to $\geq$ 10 psig.

The provisions of Technical Specification 4.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

The provisions of Technical Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

PROCEDURES AND PROGRAMS (Continued)

6.8.5 The following programs, relocated from the Technical Specifications to FSAR Chapter 16, shall be implemented and maintained:

a. Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the WASTE GAS HOLDUP SYSTEM, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

1. The limits for concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM and a surveillance program to ensure the limits are maintained.

2. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY in the event of an uncontrolled release of the tanks' contents, consistent with Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases due to Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

3. A surveillance program to ensure that the quantity of radioactivity contained in the following outdoor liquid radwaste tanks, that are not surrounded by liners, dikes, or walls capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste system, is less than the amount that would result in concentrations less than the limits of 10 CFR Part 20.1 - 20.602, Appendix B (redesignated at 56FR23391, May 21, 1991) at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA, in the event of an uncontrolled release of the tanks' contents:

   a. Reactor Makeup Water Storage Tank,
   b. Refueling Water Storage Tank,
   c. Condensate Storage Tank, and
   d. Outside temporary tanks, excluding demineralizer vessels and the liner being used to solidify radioactive waste.

ATTACHMENT 3

SAFETY EVALUATION

SAFETY EVALUATION

This license amendment requests a revision to Technical Specification 5.3.1, "Fuel Assemblies" to allow the use of an alternate zirconium based fuel cladding material, ZIRLO, for Callaway Plant. Limited substitution of fuel rods by ZIRLO filler rods would also be permitted if justified by a cycle specific reload analysis.

Union Electric is planning to load fuel with ZIRLO cladding during the Callaway Plant refuel outage currently scheduled to begin in October 1996. Therefore, Union Electric respectfully requests that the NRC Staff review and approve this license amendment request no later than June 1, 1996, so that the amendment is in place prior to receipt of new fuel with ZIRLO cladding. A similar request to allow the use of ZIRLO clad fuel rods and ZIRLO filler rods has been submitted by Commonwealth Edison for Byron and Braidwood Nuclear Power Stations.

Changing to ZIRLO cladding is the first phase of a transition to higher burnup fuel. Future core designs may feature longer cycles, higher capacity factors, and ultimately, higher discharge burnups. Using higher discharge burnup in the reactor core design reduces the number of fuel assemblies required per reload. Union Electric will save money by paying less for fuel fabrication and by using less spent fuel storage space. In order to support the required fuel enrichment and burnups, ZIRLO cladding must be used to maintain fuel integrity. The transition cannot be made until all the assemblies in the core have ZIRLO cladding and proper NRC approval of the remaining changes, such as increased discharge burnup limit, is obtained.

Background for the Current Specification

Technical Specification 5.3.1 requires fuel rods to be clad with Zircaloy-4. Fuel rods may be substituted by filler rods consisting of Zircaloy-4 or stainless steel, or by vacancies if justified by a cycle specific reload analysis.

The fuel system is designed so that there will not be damage as a result of normal operation and anticipated operational occurrences, fuel damage during postulated accidents would not be severe enough to prevent control rod insertion when it is required, and core cooling will always be maintained, even after severe postulated accidents. The design thereby meets the related requirements of 10 CFR 50.46, 10 CFR 50, Appendix A and K, 10 CFR 100, and General Design Criteria 10, 27, 35.

Impact of the Proposed Changes

In Federal Register Volume 57, Number 169, dated August 31, 1992, the NRC published amended regulations to reduce the regulatory burden on nuclear licensees. The NRC revised the acceptance criteria in 10 CFR 50.44 and 10 CFR 50.46 relating to evaluations of emergency core cooling systems and combustible gas control applicable to Zircaloy clad fuel to include ZIRLO clad fuel. ZIRLO is a preferred cladding material since it provides a significant improvement in corrosion margin and fuel integrity. The NRC noted that the revision to include ZIRLO as an acceptable zirconium based cladding material will reduce the licensee burden and will not reduce the protection of the public health or safety.

This change is consistent with 10 CFR 50.44 and 10 CFR 50.46. The change is also consistent with NRC approved topical report, WCAP-13060, "Westinghouse Fuel Assembly Reconstitution Evaluation Methodology," which meets the intent of Supplement 1 of Generic Letter 90-02, "Alternative Requirements for Fuel Assemblies in the Design Section of Technical Specifications." In addition, NUREG-1431, "Standard Technical Specifications for Westinghouse Plants," specifically includes ZIRLO as an acceptable cladding material.

An analysis of the safety implications is provided in an NRC letter to Westinghouse dated July 1, 1991, "Acceptance for Referencing of Topical Report WCAP-12610, 'Vantage+ Fuel Assembly Reference Core Report' (TAC No. 77258)." The report supports the following conclusions:

(1) The mechanical design bases and limits for the ZIRLO clad fuel assembly design are the same as those for the previously licensed Zircaloy-4 clad fuel assembly design, except that Zirlo cladding improves corrosion performance.

(2) The neutronic evaluations have shown that ZIRLO clad fuel nuclear design bases are satisfied and that key safety parameter limits are applicable. The nuclear design models and methods accurately describe the behavior of ZIRLO clad fuel.

(3) The thermal and hydraulic design basis for ZIRLO clad fuel is unchanged.

(4) The methods and computer codes used in the analysis of the non-loss of coolant accident (LOCA) licensing basis events are valid for ZIRLO clad fuel, and all licensing basis criteria will be met.

(5) The large break LOCA evaluation model was modified to reflect the behavior of the ZIRLO clad material during a loss of coolant accident. It is concluded that the revised evaluation model satisfies the intent of 10 CFR 50.46 and Appendix K of 10 CFR 50. There is no significant impact on typical large break LOCA analysis results for the ZIRLO model revisions.

In addition, bounding large break LOCA rod heatup cases were evaluated for Callaway and all acceptance criteria were met, including those in 10 CFR 50.46. Adequate margin to the peak clad temperature limit of 2200 °F is maintained.

The effect of ZIRLO on a locked rotor transient was evaluated for Callaway and all acceptance criteria were met and adequate margin to the peak clad temperature limit of 2700 °F is maintained.

The rod control cluster assembly (RCCA) ejection event was analyzed at hot full power and hot zero power. The analysis demonstrated that any consequential damage to the core or the reactor coolant system would not prevent long term core cooling and that the offsite dose would remain within the guidelines of 10 CFR 100. WCAP-12610 includes results of sensitivity analyses performed by Westinghouse that demonstrate that the impact of ZIRLO on RCCA ejection event analyses results in an insignificant change (very small benefit) in both the fraction of fuel melted at the hot spot as well as the peak fuel stored energy.

WCAP-13060 delineates the methodology used to evaluate applicable design criteria associated with reconstituted fuel assemblies that have solid filler rods replacing uranium filled fuel rods. Evaluations and analyses of fuel assembly reconstitution will be performed on a cycle specific basis whenever reconstituted fuel assemblies are used in the reactor core. The WCAP includes proposed technical specification changes based on the WCAP conclusions and the guidelines of Generic Letter 90-02.

Fuel configuration, size, enrichment and cladding material shall be limited to those designs that have been analyzed with applicable NRC-approved codes and methods and shown by test or cycle specific reload analyses to comply with all safety design bases. The use of ZIRLO fuel cladding or filler rods will be justified by a cycle specific reload analysis in accordance with NRC approved applications of fuel rod configuration. The justification of the core analysis methods must address the effect on core-wide analyses of permissible core configurations with the reconstituted fuel.

Evaluation

This license amendment requests a revision to Technical Specification 5.3.1, "Fuel Assemblies" to allow the use of an alternate zirconium based fuel cladding material, ZIRLO, for Callaway Plant. Limited substitution of fuel rods by ZIRLO filler rods would also be permitted if justified by a cycle specific reload analysis.

The proposed changes to the TS do not involve an unreviewed safety question because operation of Callaway Plant with this change would not:

1. Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report.

   The methodologies used in the accident analysis remain unchanged. The proposed changes do not change or alter the design assumptions for the systems or components used to mitigate the consequences of an accident. Use of ZIRLO fuel cladding does not adversely affect fuel performance or impact nuclear design methodology. Therefore accident analyses are not impacted.

   The operating limits will not be changed and the analysis methods to demonstrate operation within the limits will remain in accordance with NRC approved methodologies. Other than the changes to the fuel assemblies, there are no physical changes to the plant associated with this technical specification change. A safety analysis will continue to be performed for each cycle to demonstrate compliance with all fuel safety design bases.

VANTAGE 5 fuel assemblies with ZIRLO clad fuel rods meet the same fuel assembly and fuel rod design bases as other VANTAGE 5 fuel assemblies. In addition, the 10 CFR 50.46 criteria are applied to the ZIRLO clad rods. The use of these fuel assemblies will not result in a change to the reload design and safety analysis limits. Since the original design criteria are met, the ZIRLO clad fuel rods will not be an initiator for any new accident. The clad material is similar in chemical composition and has similar physical and mechanical properties as Zircaloy-4. Thus, the cladding integrity is maintained and the structural integrity of the fuel assembly is not affected. ZIRLO cladding improves corrosion performance and dimensional stability. No concerns have been identified with respect to the use of an assembly containing a combination of Zircaloy-4 and ZIRLO clad fuel rods. Since the dose predictions in the safety analyses are not sensitive to fuel rod cladding material, the radiological consequences of accidents previously evaluated in the safety analysis remain valid.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility for an accident or malfunction of equipment of a different type than any previously evaluated in the Safety Analysis Report.

VANTAGE 5 fuel assemblies with ZIRLO clad fuel rods satisfy the same design bases as those used for other VANTAGE 5 fuel assemblies. All design and performance criteria continue to be met and no new failure mechanisms have been identified. The ZIRLO cladding material offers improved corrosion resistance and structural integrity.

The proposed changes do not affect the design or operation of any system or component in the plant. The safety functions of the related structures, systems or components are not changed in any manner, nor is the reliability of any structure, system or component reduced. The changes do not affect the manner by which the facility is operated and do not change any facility design feature, structure or system. No new or different type of equipment will be installed. Since there is no change to the facility or operating procedures, and the safety functions and reliability of structures, systems or components are not affected, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Reduce the margin of safety as defined in the basis for any technical specification.

   Use of ZIRLO cladding material does not change the VANTAGE 5 reload design and safety limits. The use of these fuel assemblies will take into consideration the normal core operating conditions allowed in the Technical Specifications. For each cycle reload core, the fuel assemblies will be evaluated using NRC-approved reload design methods, including consideration of the core physics analysis peaking factors and core average linear heat rate effects.

   The use of Zircaloy-4, ZIRLO or stainless steel filler rods in fuel assemblies will not involve a significant reduction in the margin of safety because analyses using NRC-approved methodologies will be performed for each configuration to demonstrate continued operation within the limits that assure acceptable plant response to accidents and transients. These analyses will be performed using NRC-approved methods that have been approved for application to the fuel configuration.

Conclusion

Given the above discussions as well as those presented in the Significant Hazards Consideration, the proposed change does not adversely affect or endanger the health or safety of the general public or involve an unreviewed safety question.

ATTACHMENT 4

SIGNIFICANT HAZARDS EVALUATION

SIGNIFICANT HAZARDS EVALUATION

This license amendment requests a revision to Technical Specification (TS) 3/4.6.1.1, "Containment Integrity," and 3/4.6.1.3, "Containment Air Locks" to implement performance based leakage rate testing as permitted by 10 CFR 50, Appendix J. TS 6.8.4 would be revised by the addition of the leakage rate testing program description. These changes support the implementation of performance based testing allowed by Appendix J, Option B for Type A, B and C containment leak rate testing.

This proposed change is consistent with the revision to 10 CFR 50, Appendix J as noticed in 60 FR 49495 dated September 26, 1995. A similar request to partially implement Option B for Type B and C testing has been submitted by Georgia Power Company for Vogtle Electric Generating Plant.

Background

The purpose of Appendix J leak test requirements as stated in the introduction to 10 CFR 50, Appendix J is to "assure that (a) leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values as specified in the technical specifications or associated bases and (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of containment, and systems and components penetrating primary containment."

A revision to 10 CFR 50, Appendix J was issued on September 26, 1995 in Federal Register Volume 60, No. 186. The revision establishes Option B - Performance-Based Requirements, for conducting integrated leak rate tests and local leak rate tests. Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, was issued and endorses, with exceptions, NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0.

The NRC Staff issued the revised 10CFR 50, Appendix J as part of the initiative to eliminate requirements that are marginal to safety. This effort is discussed in SECY-94-036, "Staff Plans for Revising 10 CFR 50, Appendix J, Containment Leakage Testing, and for Handling Exemption Requests," dated February 17, 1994; and SECY-94-090, "Institutionalization of Continuing Program for Regulatory Improvement," dated March 31, 1994.

Appendix J, as revised by Option B, establishes new performance-based requirements and criteria for periodic leak rate testing. With Option B, the schedule requirements for integrated leak rate tests and local leak rate tests will be based upon the previous test results. NEI 94-01 was developed to provide guidance to implement Option B and the justification for extended test intervals is based on performance history and risk insights. Regulatory Guide 1.163, which endorses NEI 94-01, Revision 0, with exceptions, provides specific guidance on developing a performance-based leakage test program, acceptable leakage rate test methods, procedures, and analyses that may be used to implement the requirements and criteria of Option B. The Callaway Containment Leakage Rate Testing Program would implement performance-based testing as allowed by Option B of 10 CFR 50, Appendix J.

Justification

The proposed change to TS 3/4.6.1.1, 3/4.6.1.3 and 6.8.4g would revise or support the implementation of performance-based leakage rate testing, instead of paraphrasing Appendix J as is done in the present TS. There are no changes to the test type, test methodologies or test acceptance criteria, only the required frequency of tests would be affected. These changes will allow Union Electric to implement the recent revision to 10 CFR 50, Appendix J.

Implementation of the Containment Integrated Leakage Rate Program would allow the integrated leak rate test presently scheduled for Refuel 8 to be rescheduled, since the criteria established by Appendix J, Option B, which requires only one integrated leak rate test in 10 years is presently satisfied by past integrated leak rate test results. Additionally, Type B and C tests presently scheduled for Refuel 8 could also be evaluated for rescheduling, since they may also meet the criteria for test frequency extension. Adoption of the new performance-based leakage rate testing program will result in significant dollar and radiation exposure savings since unnecessary testing can be eliminated.

Additional Information

License Amendment No. 98 and an exemption from the requirements of 10 CFR 50, Appendix J, Section III.D.1.(a) were granted for Callaway Plant on April 5, 1995 and April

4, 1995, respectively. The license amendment and exemption provided relief from the requirements to perform the overall integrated containment leakage rate test at intervals of 40 plus or minus 10 months. The approval of the license amendment and exemption allowed the schedule for the third Type A test to be extended to Refuel 8. However, with the adoption of 10 CFR 50, Appendix J, Option B, the overall integrated containment leakage rate test scheduled for Refuel 8 will be rescheduled, based upon past performance history of Type A tests performed at Callaway Plant, using the criteria provided in NEI 94-01, Revision 0.

Evaluation

This license amendment requests a revision to Technical Specification (TS) 3/4.6.1.1, "Containment Integrity," and 3/4.6.1.3, "Containment Air Locks" to implement performance-based leakage rate testing as permitted by 10 CFR 50, Appendix J. TS 6.8.4 would be revised by the addition of the leakage rate testing program. These changes support the implementation of performance-based testing allowed by Appendix J, Option B for Type A, B and C containment leak rate testing.

The proposed changes to the TS do not involve a significant hazards consideration because operation of Callaway Plant with this change would not:

1. Involve a significant increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report.

   The proposed changes to TS 3/4.6.1.1 and 3/4.6.1.3 and the program addition to TS 6.8.4g have no affect on plant operation. The proposed changes only provide mechanisms within TS for implementing a performance-based methodology for determining the frequency of leak rate testing, as allowed by the NRC. The test type, method, and acceptance criteria will not be changed. Containment leakage will continue to be maintained within the required limits. Based on industry and NRC evaluations performed in support of developing Option B, these changes potentially result in a minor increase in the consequences of an accident previously evaluated due to the increased testing intervals. However, the proposed changes do not result in an increase in the core damage frequency since the containment system is used for mitigation purposes only.

Directly referencing the Containment Leakage Rate Testing Program for Containment ILRT and LLRT requirements does not involve any modification to plant equipment or affect the operation or design basis of the containment. Leakage rate testing is not a precursor to or an initiating event for any accident.

Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously evaluated in the Safety Analysis Report.

   The proposed changes only allow for implementation of 10 CFR 50, Appendix J, Option B and do not involve any modifications to any plant equipment or affect the operation or design basis of the containment. The proposed changes do not affect the response of the containment during a design basis accident.

3. Involve a significant reduction in a margin of safety.

   The proposed changes do not affect or change a safety limit, any limiting condition for operation or affect plant operations. The changes only implement the Appendix J, Option B test frequencies that have been determined by NRC not to involve a safety concern. The testing methods, acceptance criteria and bases are not changed and still provide assurance that the containment will provide its intended function.

Conclusion

Given the above discussions as well as those presented in the Safety Evaluation, the proposed change does not adversely affect or endanger the health or safety of the general public or involve a significant hazards consideration.

ATTACHMENT 5

ENVIRONMENTAL CONSIDERATION

ENVIRONMENTAL CONSIDERATION

This license amendment requests a revision to Technical Specification (TS) 3/4.6.1.1, "Containment Integrity," and 3/4.6.1.3, "Containment Air Locks" to implement performance based leakage rate testing as permitted by 10 CFR 50, Appendix J. TS 6.8.4 would be revised by the addition of the leakage rate testing program description. These changes support the implementation of performance based testing allowed by Appendix J, Option B for Type A, B and C containment leak rate testing.

The proposed amendment involves changes with respect to the use of facility components located within the restricted area, as defined in 10 CFR 20, and changes surveillance requirements. Union Electric has determined that the proposed amendment does not involve:

(1) A significant hazard consideration, as discussed in Attachment 4 of this amendment application;

(2) A significant change in the types or significant increase in the amounts of any effluents that may be released offsite;

(3) A significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets 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 enviromental assessment need be prepared in connection with the issuance of this amendment.