Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

October 8, 2002
LIC-02-0108

U. S. Nuclear Regulatory Commission
ATTN.: Document Control Desk
Washington, DC 20555

References:
1. Docket No. 50-285
2. Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis
3. WCAP-15691, Joint Applications Report for Containment Integrated Leak Rate Test Interval Extension, Revision 3, August 2002

SUBJECT: Fort Calhoun Station Unit No. 1 License Amendment Request, Risk-Informed One-Time Increase in Integrated Leak Rate Test Surveillance Interval

Pursuant to 10 CFR 50.90, Omaha Public Power District (OPPD) hereby transmits an application for amendment to the Fort Calhoun Station Unit 1 (FCS) Operating License. Attachment 1 provides the No Significant Hazards Evaluation and the technical bases for this requested change to the Technical Specifications (TS). Attachments 2 and 3 contain marked-up and clean-typed Technical Specification pages reflecting the requested Technical Specification changes.

The proposed changes are submitted on a risk-informed basis as described in Reference 2. The proposed changes to extend the Integrated Leak Rate Test (ILRT) surveillance interval from 10 to 15 years are justified based on a combination of risk-informed analysis and assessment of the containment structural condition utilizing ILRT historical results and containment inspection programs. The risk aspects of the justification have been prepared by the Combustion Engineering Owners Group (CEOG) and are presented in Reference 3. Reference 3 was submitted to the NRC for review by CEOG letter CEOG-02-162 dated August 15, 2002. A brief description and history of the FCS ILRT testing results and the containment inspection program are discussed in Reference 3.

Reference 3 provides the risk-informed supporting analysis to demonstrate that the increase in risk of extending the ILRT interval from 10 to 15 years is insignificant. That analysis, done in conjunction with Reference 2, shows that the increase in total plant risk due to the extended ILRT interval is less than one half of one percent. The change in Large Early Release Fraction (LERF) is only 1.226E-9/year when the time interval is extended from 10 to 15 years.

A017

OPPD requests approval of the proposed amendment by July 12, 2003 to support the next FCS refueling outage in the Fall of 2003. OPPD requests 60 days to implement this amendment. No commitments are made to the NRC in this letter.

I declare under penalty of perjury that the foregoing is true and correct. (Executed on October 8, 2002)

If you have any questions or require additional information, please contact Dr. R. L. Jaworski at (402) 533-6833.

Sincerely,

D. J. Bannister
Manager Fort Calhoun Station

DJB/TRB/trb

Attachments:
1. Fort Calhoun Station's Evaluation
2. Markup of Technical Specification Pages
3. Clean-Typed Technical Specification Pages

c: E. W. Merschoff, NRC Regional Administrator, Region IV
A. B. Wang, NRC Project Manager
J. G. Kramer, NRC Senior Resident Inspector
Division Administrator - Public Health Assurance, State of Nebraska
Winston & Strawn

Attachment 1

Fort Calhoun Station's Evaluation For Risk-Informed One-Time Increase in Integrated Leak Rate Test Surveillance Interval

1.0 INTRODUCTION

This letter is a request to amend Operating License DPR-40 for the Fort Calhoun Station (FCS) Unit No. 1.

The proposed changes are submitted on a risk-informed basis as described in Reference 10.1. The proposed changes to extend the Integrated Leak Rate Test (ILRT) surveillance interval are justified based on a combination of risk-informed analysis and assessment of the containment structural condition utilizing ILRT historical results and containment inspection programs. The risk aspects of the justification have been prepared by the Combustion Engineering Owners Group (CEOG) and are presented in Reference 10.2. Reference 10.2 was submitted to the NRC for review by CEOG letter CEOG-02-162 dated August 15, 2002. A brief description and history of the FCS ILRT testing results and the containment inspection program are discussed in Reference 10.2.

Reference 10.2 provides the risk-informed supporting analysis to demonstrate that the increase in risk of extending the ILRT interval from 10 to 15 years is insignificant. That analysis, done in conjunction with Reference 2, shows that the increase in total plant risk due to the extended ILRT interval is less than one half of one percent. The change in Large Early Release Fraction (LERF) is only 1.226E-9/year when the time interval is extended from 10 to 15 years. Reference 10.2 demonstrates that, from a risk perspective, an extension in the interval out to 20 years has an insignificant impact on risk. This is consistent with the findings of Reference 10.3. This submittal requests only a one-time interval extension from 10 to 15 years.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed change is to add Technical Specification (TS) 5.19(4) to specify, "The first Type A test performed after the November 1993 Type A test shall be no later than November 2008."

3.0 BACKGROUND

FCS has implemented a Containment Leakage Rate Program closely following guidelines established by Reference 10.4. Using these guidelines and reviewing the past ILRT results has allowed the plant to establish a frequency of performance intervals of every ten years per 10CFR50, Appendix J, Option B. The most recent ILRT was performed in November 1993.

The CEOG has issued Reference 10.2. This document provides a risk-informed methodology for justifying modification of the plant licensing basis for pressurized water reactor containment ILRT intervals. In addition, Reference 10.2 includes a plant specific analysis, using the methodology outlined, for FCS in Appendix E. This report concludes the requested ILRT extension has a very small impact on the risk of events that may give rise to large early radionuclide releases. Any decrease in containment reliability due to

ILRT extension for the requested ILRT test interval modifications would result in a very small (negligible) impact on the large early release probability.

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

FCS was licensed for construction prior to May 21, 1971, and at that time committed to the preliminary General Design Criteria (GDC). These preliminary design criteria are contained in the FCS USAR Appendix G.

This activity complies with FCS Design Criterion 10, "Containment," which is similar to 10 CFR 50 Appendix A GDC 16, "Containment design." FCS Design Criterion 10 states that containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

This activity also complies with FCS Design Criterion 40, "Missile Protection," which is similar to 10 CFR 50 Appendix A GDC 4, "Environmental and dynamic effects design bases." FCS Design Criterion 40 states that protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

This activity also complies with FCS Design Criterion 49, "Containment Design Basis," which is similar to 10 CFR 50 Appendix A GDC 50, "Containment design basis." FCS Design Criterion 49 states that the containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

This activity also complies with FCS Design Criterion 50, "NDT Requirement for Containment Material," which is similar to 10 CFR 50 Appendix A GDC 51, "Fracture prevention of containment pressure boundary." FCS Design Criterion 50 states that principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperature under normal operating and testing conditions are not less than 30°F above nil ductility transition (NDT) temperature.

This activity also complies with FCS Design Criterion 54, "Containment Leakage Rate Testing," which is similar to 10 CFR 50 Appendix A GDC 52, "Capability for Containment Leakage Rate Testing." FCS Design Criterion 54 states that the containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period of time to verify its conformance with required performance.

This activity also complies with FCS Design Criterion 55, "Containment Periodic Leakage Rate Testing," which is similar to 10 CFR 50 Appendix A GDC 53, "Provisions for Containment Testing and Inspection." FCS Design Criterion 55 states that the containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.

All of these FCS Design Criteria will continue to be satisfied after the change allowing a one-time extension of the ILRT interval from 10 to 15 years.

5.0 TECHNICAL ANALYSIS

Evaluation

FCS has implemented a Containment Leakage Rate Program closely following guidelines established by Reference 10.4. Using these guidelines and reviewing the past ILRT results has allowed the plant to establish a frequency of performance intervals of every ten years per 10 CFR 50, Appendix J, Option B. The most recent ILRT was performed in November 1993.

The CEOG has issued Reference 10.2. This document provides a risk-informed methodology for justifying modification of the plant licensing basis for pressurized water reactor containment ILRT intervals. In addition, Reference 10.2 includes a plant specific analysis, using the methodology outlined, for FCS in Appendix E. This report concludes the requested ILRT extension has a very small impact on the risk of events that may give rise to large early radionuclide releases. Any decrease in containment reliability due to ILRT extension for the requested ILRT test interval modifications would result in a very small (negligible) impact on the large early release probability.

Risk Evaluation

Additional evaluation has been performed using a methodology similar to that already approved for the Crystal River Unit 3 Plant (CR3). This evaluation also concludes that the risk associated with the ILRT frequency extension is small and quantifiable.

The purpose of this evaluation is to present a plant-specific analysis using a methodology similar to that already approved for the Crystal River 3 (CR3) application (Reference 10.5). Note that OPPD concludes that the methodology applied in Reference 10.2 to be reasonable and consistent with good practice in risk-informed evaluations. The results of the Reference 10.2, which represents the use of a best-estimate approach to establish the probability of the small isolation failures of interest, demonstrates an even better risk justification of the request. The previously approved methodology utilizes a 95^{th} percentile estimate of the probability of the small isolation events and the results reflect a somewhat greater impact of the change on overall risk. Other differences between the methodologies will be described in the body of the evaluation below. The change is demonstrated to be risk insignificant in both methodologies.

Both of the methodologies followed the same general approach to the evaluation of the risk of the interval extension. There were differences in the approaches in the assumptions and in the development of a probability estimate for the release class 3 events. The methodologies:

- Both utilize the EPRI TR-104285 (Reference 10.6) release classes to categorize the various containment failure scenarios.
- Both establish the plant-specific frequencies for each EPRI release class.
- Both define estimated leakage for each release class.
- Both quantify the risk for each release class by multiplying the class frequency times the assumed leakage.
- Both evaluated a baseline case (3 tests in 10 years), a current case (1 test in 10 years), and the proposed case (1 test in 15 years).

Table 1 summarizes the treatment of each of the EPRI Release Classes and provides a summary of some of the differences between the Reference 10.2 and the CR3 methodologies.

Table 1
EPRI Release Class Definitions

Release Class	Description	CR3 Submittal	CEOG (Table E2-2 of Reference 10.2)
1	No containment failure	Frequency reduced as Class 3 increases; leakage magnitude increases to 2 L_a	Frequency reduced with Class 3 increase; considered leakage of L_a
2	Large isolation failures	No change from baseline consequence measures; considered leakage of 35 L_a	No change from baseline consequence measures; considered leakage of 1000 L_a
3	Isolation failures	3a: small leaks, 10 L_a, non-LERF 3b: large leaks, 35 L_a, LERF Probability derived using 95th %-ile χ^2 distribution of NUREG-1493 data	3a: small leaks, 25 L_a, non-LERF 3b: large leaks, 1000 L_a, LERF Probability derived using log-normal distribution of NUREG-1493 data
4,5	Other small isolation failures (LLRT)	No change from baseline consequence measures; not analyzed	No change from baseline consequence measures; not analyzed
6	Other isolation failures	No change from baseline consequence measures; considered leakage of 35 L_a	No change from baseline consequence measures; considered leakage of 350 L_a
7	Induced failures	No change from baseline consequence measures; considered leakage of 100 L_a	No change from baseline consequence measures; considered leakage of 2800 L_a
8	Bypass	Characterized by SGTR scenario – not impacted by ILRT extension	Characterized by SGTR and ISLOCA – not impacted by ILRT extension

Evaluation of Baseline ILRT Interval

The plant-specific evaluation of risk for the baseline case ILRT interval for FCS is presented in Table 2. The release frequencies for the Class 2, 6, 7, and 8 bins are taken from Reference 10.2, which had compiled these data based on the FCS Probabilistic Risk Assessment (PRA). As noted in Table 1, the risk associated with the Class 4 and 5 bins is not impacted by the ILRT interval and is not analyzed here. The release frequencies for the Class 3a and 3b bins are determined based on the previously approved methodology (see next paragraph). The release frequency for Class 1 is the value of core damage frequency (CDF) reduced by the frequencies of the Class 3a and 3b scenarios. (Note Reference 10.2 had utilized a value of CDF representative of sequences in which the containment remains intact. This value was approximately 36% of total CDF. The previously approved methodology used total CDF. Total CDF will be used in this plant-specific evaluation.)

The Class 3a and 3b frequencies in the previously approved methodology were determined based on a 95th percentile χ^2 distribution of the NUREG-1493 data. For the baseline ILRT interval (3 tests in 10 years), this resulted in a frequency for Class 3a of 0.064 (Reference 4) times CDF and a frequency for Class 3b of 0.021 (Reference 5) times CDF. These frequencies are used in the FCS evaluation presented in Table 2. Note the total CDF for FCS is 1.34E-05 per year and the intact containment release frequency is 4.87E-06 per year based on the current plant risk model.

Table 2
FCS Risk Evaluation
of Baseline ILRT Interval

Class	Frequency (per reactor-year)	Release (person-rem)	Risk (person-rem/year)
1	FREQ(intact)-FREQ(3a)-FREQ(3b) = 3.73E-06 (Reference 10.5)	L_a = 3.77E+03 (Reference 10.2)	0.01
2	7.43E-08	35 L_a = 1.32E+05	0.01
3a	0.064 x CDF = 8.58E-07	10 L_a = 3.77E+04	0.03
3b	0.021 x CDF = 2.81E-07	35 L_a = 1.32E+05	0.04
6	0.00E-00	35 L_a = 1.32E+05	0.00
7	5.90E-06	100 L_a = 3.77E+05	2.22
8	2.53E-06	2.54E+06 (Reference 10.2)	6.43
		Total Risk	8.74

In Reference 10.1, a risk contribution of the intact containment sequences (i.e., Classes 1, 3a, and 3b) was determined. Using the previously approved methodology, the risk contribution due to the ILRT Type A testing was considered to be due to the Class 3a and 3b scenarios. From Table 2, it can be seen that the risk contribution associated with the ILRT testing interval considering Classes 3a and 3b is:

$$\% \text{ Risk} = [(\text{Risk}_{\text{Class 3a}} + \text{Risk}_{\text{Class 3b}}) / \text{Total Risk}] \times 100$$

$$= [(0.03 + 0.04) / 8.74] \times 100$$

$$= 0.79\%$$

In Reference 10.2, it was also assumed that the Class 2, 3b, 6, 8, and the early Class 7 scenarios could lead to large early releases and thus, contribute to LERF. The previously approved methodology focused only on the Class 3b scenario, which is the only one affected by the consideration of the ILRT interval. As the parameter of concern in the evaluation is ΔLERF, and because Class 3b is the only class affected by the interval extension, ΔLERF is compared on a consistent basis in both methodologies. Thus, for this evaluation the baseline LERF is the Class 3b frequency, or 2.81E-07 per year.

Risk Evaluation of the Current ILRT Interval (1 in 10 years)

This evaluation of the "once in 10 years" interval will be performed using the same approach as taken above for the baseline case. The frequencies for all release classes, except Class 1, 3a, and 3b, are unaffected by the change in the interval and remain as in Table 2. And the releases for all of the classes are the same as those shown in Table 2 for the baseline case.

The increased probability of not detecting excessive leakage in a Type A test directly impacts the frequencies of the Class 3 events. Based on the previously approved methodology, the Class 3a and 3b frequencies are determined by simply multiplying the baseline frequency by a factor of 1.1. With this change in the Class 3 frequencies, the Class 1 frequency is also adjusted to preserve the total CDF. The evaluation of the current interval is presented in Table 3.

Table 3
FCS Risk Evaluation
of Current ILRT Interval

Class	Frequency (per reactor-year)	Release (person-rem)	Risk (person-rem/year)
1	FREQ(intact)-FREQ(3a)-FREQ(3b) = 3.61E-06 (Reference 10.5)	L_a = 3.77E+03 (Reference 10.2)	0.014
2	7.43E-08	35 L_a = 1.32E+05	0.010
3a	1.1 x 0.064 x CDF = 9.43E-07	10 L_a = 3.77E+04	0.036
3b	1.1 x 0.021 x CDF = 3.10E-07	35 L_a = 1.32E+05	0.041
6	0.00E-00	35 L_a = 1.32E+05	0.000
7	5.90E-06	100 L_a = 3.77E+05	2.224
8	2.53E-06	2.54E+06 (Reference 10.2)	6.426
		Total Risk	8.750

As was noted above for the baseline evaluation:

- the risk contribution due to the Type A test interval is [(0.036 + 0.041) / 8.750] x 100, or 0.87%.
- the LERF for the current interval evaluation is the Class 3b frequency, or 3.10E-07 per year.

Risk Evaluation of the Proposed ILRT Interval (1 in 15 years, one-time)

This evaluation of the "once in 15 years" interval is performed using the same approach as taken above for the baseline case. The frequencies for all release classes, except Class 1, 3a, and 3b, are unaffected by the change in the interval and remain as in Table 2. The releases for all of the classes are the same as those shown in Table 2 for the baseline case.

The increased probability of not detecting excessive leakage in a Type A test directly impacts the frequencies of the Class 3 events. Based on the previously approved methodology, the Class 3a and 3b frequencies are determined by simply multiplying the baseline frequency by a factor of 1.15. With this change in the Class 3 frequencies, the Class 1 frequency is also adjusted to preserve the total CDF. The evaluation of the current interval is presented in Table 4.

Table 4
FCS Risk Evaluation
of Proposed ILRT Interval

Class	Frequency (per reactor-year)	Release (person-rem)	Risk (person-rem/year)
1	FREQ(intact)-FREQ(3a)-FREQ(3b) = 3.56E-06	L_a = 3.77E+03 (Reference 10.2)	0.013
2	7.43E-08	35 L_a = 1.32E+05	0.010
3a	1.15 x 0.064 x CDF = 9.86E-07	10 L_a = 3.77E+04	0.037
3b	1.15 x 0.021 x CDF = 3.24E-07	35 L_a = 1.32E+05	0.043
6	0.00E-00	35 L_a = 1.32E+05	0.000
7	5.90E-06	100 L_a = 3.77E+05	2.224
8	2.53E-06	2.54E+06 (Reference 10.2)	6.426
Total Risk			8.754

As was noted above for the baseline evaluation:

- the risk contribution due to the Type A test interval is [(0.037 + 0.043) / 8.754] x 100, or 0.91%.
- the LERF for the current interval evaluation is the Class 3b frequency, or 3.24E-07 per year.

Conditional Containment Failure Probability

Another parameter of interest in evaluating the risk impact of a change to the ILRT interval is the conditional containment failure probability (CCFP). In the Reference 10.2 methodology, ΔLERF was considered to be directly related to ΔCCFP. The results using that approach were a ΔCCFP of 0.05% due to the proposed interval compared to the current interval, and 0.11% due to the change to the proposed interval compared to the baseline case. In the previously approved methodology that was used in the plant-specific evaluation developed in this submittal, CCFP was defined as:

CCFP = 1 – (frequency of no containment failure sequences / CDF)

CCFP = 1 – [freq (Cl1) + freq (Cl3a)]/CDF

Further, the sequences representing no containment failure were considered to be the Class 1 and 3a events. Thus, using this approach and the information from Tables 2, 3, and 4, the ΔCCFP between the current ILRT interval and the proposed ILRT interval may be derived by:

$\Delta CCFP_{c\ to\ p}$ = {[freq (Cl1) + freq (Cl3a)]$_c$ - [freq (Cl1) + freq (Cl3a)]$_p$}/ CDF

= {[3.614E-06 + 9.434E-07] - [3.557E-06 + 9.862E-07]} / 1.34E-05

= 0.0011, or 0.11%

Similarly, the impact of the proposed ILRT interval compared with the baseline ILRT interval is given by:

$\Delta CCFP_{b\ to\ p}$ = {[freq (Cl1) + freq (Cl3a)]$_b$ - [freq (Cl1) + freq (Cl3a)]$_p$}/ CDF

= {[3.728E-06 + 8.576E-07] - [3.557E-06 + 9.862E-07]} / 1.34E-05

= 0.0032, or 0.32%

Summary

A summary of the risk evaluation of the ILRT interval changes using the previously approved methodology is presented in Table 5.

Reference 10.1 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reference 10.1 defines very small changes in risk as resulting in increases of CDF below 1E-06/year and increases in LERF below 1E-07/year. Since the ILRT does not impact CDF, the relevant metric is LERF. Calculating the increase in LERF involves determining the impact of the ILRT interval on the leakage probability.

Table 5
Summary of Results of ILRT Interval
Risk Evaluation (Using Previously Approved Approach)

ILRT Interval	ILRT Risk Contribution	LERF (per year)	ΔLERF from baseline (per year)	ΔLERF from current (per year)
baseline (3 in 10 years)	0.79%	2.81E-07	—	—
current (1 in 10 years)	0.87%	3.10E-07	2.81E-08	—
proposed (1 in 15 years)	0.91%	3.24E-07	4.221E-08	1.407E-08

For comparison purposes, the evaluation results from Reference 10.2, derived using differences in assumptions and methodology, are presented in Table 6.

Table 6
Summary of Results of ILRT Interval
Risk Evaluation (using Reference 10.2 approach)

ILRT Interval	ILRT Risk Contribution	LERF	□LERF from baseline	□LERF from current
baseline (3 in 10 years)	0.45%	2.672E-06	—	—
current (1 in 10 years)	0.86%	2.674E-06	1.635E-09	—
proposed (1 in 15 years)	1.16%	2.675E-06	2.862E-09	1.226E-09

Conclusion

The risk associated with extending the ILRT interval is quantifiable. OPPD has utilized two alternate methodologies to quantify the risk and evaluate the proposed change in the ILRT interval to 15 years. Both methodologies demonstrate the risk associated with the extension of the interval is small. On this basis, OPPD requests approval of a one-time extension of the FCS ILRT interval to 15 years.

6.0 REGULATORY ANALYSIS

The proposed changes are submitted on a risk-informed basis as described in Reference 10.1. The proposed changes to extend the Integrated Leak Rate Test (ILRT) surveillance interval are justified based on a combination of risk-informed analysis and assessment of the containment structural condition utilizing ILRT historical results and containment inspection programs. The risk aspects of the justification have been prepared by the Combustion Engineering Owners Group (CEOG) and are presented in Reference 10.2. Reference 10.2 was submitted to the NRC for review by CEOG letter CEOG-02-162 dated August 15, 2002.

Reference 10.2 provides the risk-informed supporting analysis to demonstrate that the increase in risk of extending the ILRT interval from 10 to 15 years is insignificant. That analysis, done in conjunction with Reference 2, shows that the increase in total plant risk due to the extended ILRT interval is less than one half of one percent. The change in LERF is only 1.226E-9/year when the time interval is extended from 10 to 15 years. Reference 10.2 demonstrates that, from a risk perspective, an extension in the interval out to 20 years has an insignificant impact on risk. This is consistent with the findings of Reference 10.3. This submittal requests only a one-time interval extension from 10 to 15 years. This complies with the regulatory requirements in FCS Design Criteria 10, 40, 49, 50, 54, and 55 by continuing to prevent damage to the containment structure.

In conclusion, based on the considerations discussed above, (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.

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

OPPD has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change adds a one-time extension to the current surveillance interval to the current surveillance interval for Type A testing (ILRT). The current test interval of 10 years, based on performance history, would be extended on a one-time basis to 15 years from the last Type A test. The proposed extension to Type A testing cannot increase the probability of an accident previously evaluated since the containment Type A test is not a modification, nor a change in the way that plant systems, structures, or components are operated, and is not an activity that could lead to equipment failure or accident initiation. The proposed change does not involve a significant increase in the consequences of an accident since research in Reference 10.3 has found that generically very few potential leaks are not identified in Type B and C tests. Reference 10.3 concluded that an increase in the test interval to 20 years resulted in an imperceptible increase in risk. FCS provides a high degree of assurance through testing and inspection that the containment will not degrade in a manner only detectable by Type A testing. Inspections required by ASME code and the Maintenance Rule are performed in order to identify indications of containment degradation that could affect leak tightness. Type B and C testing required by 10 CFR 50, Appendix J are not affected by this proposed extension to the Type A test interval and will continue to identify containment penetration leakage paths that would otherwise require a Type A test.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed change adds a one-time extension to the current surveillance interval to the current surveillance interval for Type A testing (ILRT). The change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. This change will not alter assumptions made in the safety analysis and licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

Response: No

The proposed change will not result in operation of the facility involving a significant reduction in a margin of safety. The proposed change adds a one-time extension to the current interval for Type A testing. The current test interval of 10 years, based on performance history, would be extended on a one-time basis to 15 years from the last Type A test. Reference 10.3 has found that generically very few potential leaks are not identified in Type B and C tests. Reference 10.3 concluded that an increase in the test interval to 20 years resulted in an imperceptible increase in risk. Furthermore, the extended test interval would have a minimal effect on such risk since Type B and C testing detect over 95 percent of potential leakage paths. A plant specific risk calculation, as part of Reference 10.2, on this topic obtained results consistent with the generic conclusions of Reference 10.3. The overall increase in risk contribution was determined as 0.31%.

Based on the above, OPPD concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

8.0 ENVIRONMENTAL CONSIDERATION

The proposed change adds a one-time extension to the current surveillance interval to the current surveillance interval for Type A testing (ILRT). The current test interval of 10 years, based on performance history, would be extended on a one-time basis to 15 years from the last Type A test. The changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

- As demonstrated in Section 7.0, the proposed amendment does not involve a significant hazards consideration.

- The proposed amendment does not result in a significant change in the types or increase in the amounts of any effluents that may be released off-site. Also, the TS change does not introduce any new effluents or significantly increase the quantities of existing effluents. As such, the change cannot significantly affect the types or amounts of any effluents that may be released off-site.

- The proposed amendment does not result in a significant increase in individual or cumulative occupational radiation exposure. The proposed change does not result in any physical plant changes. No new surveillance requirements are anticipated as a result of these changes that would require additional personnel entry into radiation controlled areas. Therefore, the amendment has no significant affect on either individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

9.0 PRECEDENCE

Several requests have already been approved by the NRC for the one-time surveillance interval extension to 15 years for the Type A test. The proposed change is similar to the recently approved requests by Waterford 3 (February 14, 2002) and Calvert Cliffs (May 1, 2002).

10.0 REFERENCES

10.1 Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis

10.2 WCAP-15691, Joint Applications Report for Containment Integrated Leak Rate Test Interval Extension, Revision 3, August 2002

10.3 NUREG-1493, Performance Based Containment Leak-Test Program

10.4 NEI 94-01, Industry Guideline for Implementing Performance Based-Option of 10 CFR 50 Appendix J

10.5 FPC Letter to USNRC, 3F0601-06, June 20, 2001, Crystal River-Unit 3 – License Amendment Request #267, Revision 2, "Supplemental Risk-Informed Information in Support of License Amendment Request #267".

ATTACHMENT 2

Markup of Technical Specification Page

5.0 ADMINISTRATIVE CONTROLS

5.18 Process Control Program (PCP) (Continued)

a. Shall be documented and records of reviews performed shall be retained as required by the Quality Assurance Program. This documentation shall contain:

   1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   2. A determination that the change will maintain the overall conformance of the solidified waste program to existing requirements of federal, state, or other applicable regulations.

b. Shall become effective after the review and acceptance by the Plant Review Committee and the approval of the plant manager.

c. Temporary changes to the PCP may be made in accordance with Technical Specification 5.8.2.

d. Shall be submitted to the Nuclear Regulatory Commission in the form of a complete, legible copy of the entire PCP as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the PCP was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed and shall indicate the date (e.g., month/year) the change was implemented.

5.19 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," as modified by the following exceptions:

(1) If the Personnel Air Lock (PAL) is opened during periods when containment integrity is not required, the PAL door seals shall be tested at the end of such periods and the entire PAL shall be tested within 14 days after RCS temperature T_{cold} > 210°F.

(2) Type A tests may be deferred for penetrations of the steel pressure retaining boundary where the nominal diameter does not exceed one inch.

(3) Elapsed time between consecutive Type A tests used to determine performance shall be at least 24 months or refueling interval.

(4) The first Type A test performed after the November 1993 Type A test shall be no later than November 2008.

The containment design accident pressure (P_a) is 60 psig.

ATTACHMENT 3

Clean-Typed Technical Specification Page

6.0 **ADMINISTRATIVE CONTROLS**
5.18 Process Control Program (PCP) (Continued)

a. Shall be documented and records of reviews performed shall be retained as required by the Quality Assurance Program. This documentation shall contain:

   1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   2. A determination that the change will maintain the overall conformance of the solidified waste program to existing requirements of federal, state, or other applicable regulations.

b. Shall become effective after the review and acceptance by the Plant Review Committee and the approval of the plant manager.

c. Temporary changes to the PCP may be made in accordance with Technical Specification 5.8.2.

d. Shall be submitted to the Nuclear Regulatory Commission in the form of a complete, legible copy of the entire PCP as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the PCP was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed and shall indicate the date (e.g., month/year) the change was implemented.

5.19 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," as modified by the following exceptions:

(1) If the Personnel Air Lock (PAL) is opened during periods when containment integrity is not required, the PAL door seals shall be tested at the end of such periods and the entire PAL shall be tested within 14 days after RCS temperature $T_{cold} > 210°F$.

(2) Type A tests may be deferred for penetrations of the steel pressure retaining boundary where the nominal diameter does not exceed one inch.

(3) Elapsed time between consecutive Type A tests used to determine performance shall be at least 24 months or refueling interval.

(4) The first Type A test performed after the November 1993 Type A test shall be no later than November 2008.

The containment design accident pressure (P_a) is 60 psig.