Progress Energy

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 50.90

July 1, 2004
3F0704-05

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Crystal River Unit 3 – License Amendment Request (LAR) #283, Revision 0, Improved Technical Specification (ITS) 5.6.2.20, Containment Leakage Rate Testing Program, Integrated Leakage Rate Testing (ILRT) Surveillance Interval Extension

Reference: PEF to NRC letter, 3F0304-10, dated March 31, 2004, Crystal River Unit 3 – Letter of Intent to Submit a License Amendment Request to Extend Type A Integrated Leakage Rate Testing (ILRT) Surveillance Interval Referencing a New Method for Frequency Determination

Dear Sir:

Florida Power Corporation, doing business as Progress Energy Florida, Inc. (PEF), hereby submits License Amendment Request (LAR) #283, Revision 0, requesting a change to the Crystal River Unit 3 (CR-3) Facility Operating License in accordance with 10 CFR 50.90. LAR #283, Revision 0, proposes a revision to the CR-3 Improved Technical Specification (ITS) 5.6.2.20, "Containment Leakage Rate Testing Program," to allow for a 20-year Integrated Leak Rate Test (ILRT) interval.

As stated in the referenced letter, CR-3 is submitting this LAR as a pilot application to assist in developing a revision to Regulatory Guide (RG) 1.163 (1995). This amendment request is structured with the assumption that its review will be treated as a pilot application. This is a risk informed submittal based on NEI 94-01, Revision 1, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and EPRI Product No. 1009325, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals."

PEF is respectfully requesting NRC approval of this LAR by July 30, 2005 with 30 days for implementation.

No new regulatory commitments are made in this letter.

The Crystal River Unit 3 (CR-3) Plant Nuclear Safety Committee has reviewed this request and recommended it for approval.

Progress Energy Florida, Inc.
Crystal River Nuclear Plant
15760 W. Powerline Street
Crystal River, FL 34428

A001

If you have any questions regarding this submittal, please contact Mr. Sid Powell, Supervisor, Licensing and Regulatory Programs at (352) 563-4883.

Sincerely,

Dale E Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

DEY/lvc

Attachments:

A. Description of the Proposed License Amendment Request, Reason for Request, Technical Evaluation, Risk Evaluation, Conclusion and References
B. Regulatory Analysis
C. Proposed Revised Improved Technical Specification Pages - Strikeout/Shadowed
D. Proposed Improved Technical Specification Pages – Revision Bar Format
E. Risk Assessment For 20 year ILRT Interval

xc: NRR Project Manager
Regional Administrator, Region II
Senior Resident Inspector

STATE OF FLORIDA

COUNTY OF CITRUS

Dale E. Young states that he is the Vice President, Crystal River Nuclear Plant for Florida Power Corporation, doing business as Progress Energy Florida, Inc.; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 1st day of July, 2004, by Dale E. Young.

Janet Schroeder

Signature of ~~Notary Public~~
State of Florida

JANET SCHROEDER
MY COMMISSION # DD 128063
EXPIRES: June 20, 2006
Bonded Thru Notary Public Underwriters

(Print, type, or stamp Commissioned
Name of Notary Public)

Personally Known ✓ -OR- Produced Identification ______

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER - UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT A

LICENSE AMENDMENT REQUEST #283, REVISION 0

Description of the Proposed License Amendment Request, Reason for Request, Technical Evaluation, Risk Evaluation, Conclusion and References

1.0 DESCRIPTION OF THE PROPOSED CHANGE

The purpose of License Amendment Request (LAR) #283, Revision 0, is to propose a revision to the Crystal River Unit 3 (CR-3) Improved Technical Specification (ITS) 5.6.2.20, "Containment Leakage Rate Testing Program," to allow for a 20-year Integrated Leak Rate Test (ILRT) interval.

Currently, ITS 5.6.2.20 requires: "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 exception:

1. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the November 7, 1991 Type A test shall be performed no later than November 6, 2006."

CR-3 proposes to replace the above requirement as follows:

"NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the November 7, 1991 Type A test shall be performed no later than November 6, 2011."

2.0 REASON FOR REQUEST

The reason for this LAR is to reschedule the ILRT to coincide with the replacement of the CR-3 Once-Through Steam Generators currently planned for Refueling Outage 16 (2009).

3.0 TECHNICAL EVALUATION

NEI 94-01 Revision 1 (Draft), Section 9.2.3, specifies the requirements for extended Type A ILRT test intervals. It allows for an ILRT test frequency of at least once per 20 years based upon acceptable performance history. Acceptable performance history is defined as successful completion of two consecutive periodic Type A tests where the calculated performance leakage rate was less than 1.0 La. CR-3 has completed acceptable ILRTs in 1976 (preoperational), 1980, 1983, 1987, and 1991. Table 1 contains the ILRT (Type A) Performance History.

NEI 94-01 Revision 1 (Draft), Section 9.2.3.1, provides additional requirements for ILRT interval extensions beyond ten years. Type A testing (ILRT) intervals of up to 20 years are allowed by this document provided the continuing supplemental means of identifying potential containment degradation are performed.

1. A general visual examination of accessible interior and exterior surfaces of the containment for structural problems should be performed at each 1/3 of a ten-year

inspection interval. This examination may be performed in conjunction with or coordinated with the general visual examination of ASME B&PV Code Section XI, Subsections IWE and IWL.

CR-3 has completed general visual examinations in conjunction with the IWE and IWL examinations. Those results were satisfactory with no indications of significant degradation. Table 2 contains the visual examination history. Information regarding the CR-3 Inservice Inspection (ISI) Program, including IWE/IWL examinations, was previously provided in the CR-3 Response to NRC Request for Additional Information Re: Proposed License Amendment Request #267, Revision 2, "Containment Leakage Rate Testing Program" (TAC No. MB1349) dated July 16, 2001. These inspections will continue to be implemented in accordance with the ISI program.

Review of operating experience (OE) such as NRC Information Notice 2004-09: Corrosion of Steel Containment and Containment Liner, is performed in accordance with procedure CAP-NGGC-0202, Operating Experience Program. Evaluations performed per this procedure determine OE applicability and follow-up actions to reduce the probability of occurrence of similar events.

2. Qualitative low pressure containment monitoring and trending should be performed to provide a gross indication of the potential existence of excessive containment to atmosphere leak path(s). Low pressure monitoring variables may include gas makeup rates for inerted containments and containment venting frequencies for containments with internal compressed air bleeds performed to maintain containment positive pressure within acceptable limits. Establishing historical trends and comparing existing trends of gas makeup or venting frequency over a long period (several months) will provide a gross indication on containment to atmosphere leak-tightness.

   CR3's Reactor Building (RB) pressure has been essentially constant for the last 3 years with no indication of a gross containment to atmosphere leak path. The RB pressure data was graphed and indicated no adverse trend. It was determined that the equalization of Reactor Building pressure is primarily performed during reactor startups, and normally only performed during unit operation if there is RCS leakage. The leak tightness of containment is evidenced by relatively constant containment pressure (with daily fluctuations due to atmospheric pressure and temperature changes) and RB pressure equalization not being a routine activity during unit operation. It can be concluded that there is not an excessive containment to atmosphere leak path(s). Changes to OP-417, "Containment Operating Procedure" have been incorporated to monitor the containment makeup and venting frequency. The Control Room will log and notify the Leak Rate Program Manager of the start/stop times, beginning/ending RB pressure, and the reason for RB equalization.

TABLE 1

ILRT (Type A) Performance History

ILRT Completion Date (Refuel Outage)	**Report Submitted to NRC (FPC letter number)**	**As-Found Leakage Rate (95% upper confidence limit)**		
		Total Time Analysis (% wt./day)	Mass Point Analysis (% wt./day)	* Adjusted Values
1976 (pre-operational)				
6/30/80 (2R)	10/30/80 (3F1080-09)	n/a	0.142	
7/11/83 (4R)	10/11/83 (3F1083-13)	0.179	0.148	
11/15/87 (6R)	3/21/88 (3F0388-16)	0.137	0.107	0.147 (Total Time Analysis)
11/07/91 (8R)	1/29/92 (3F0192-16)	0.0986	0.1014	0.111 (Mass Point Analysis)

These results are below the as-found acceptance criteria of 1.0La (0.25% wt./day), and the as-left acceptance criteria of 0.75La (0.1875% wt./day).

* In 1997, it was discovered that several penetrations were not properly vented during past ILRTs. Those penetrations were LLRT'd, and their test leakages were added to the last two ILRTs (as Type C penalties). The ILRT results were adjusted and transmitted by CR-3 to NRC letter, 3F0897-11, dated August 12, 1997. Reference: LER 50-302/97-014-00, Restart Issue R-10 (OP-15A & 16A).

TABLE 2

General Visual Examination History

1994 – External general visual inspection was completed. Report indicates that containment is in good sound condition with no indication of major faults. (SP-182, RAN 90013-0109).

1997 – Interior inspection was performed in conjunction with Restart Issue M-12. The RB Liner was showing areas of paint flaking, peeling, and general areas of pitting and corrosion. The degradation appeared to have occurred at the interface of the caulk and the RB liner (there was essentially no metal loss below the 95 foot elevation). The liner was recoated by Work Request (WR) 346262. The cork and caulk were installed per WR 345306. It was concluded that the structural integrity of the liner has not been degraded beyond its design margin.

1998 – External inspection performed. A few minor cracks and spalls were found during the inspection, which were typical for a facility of this age. No concrete problems were observed that impacted the design function or integrity of the Reactor Building. (Special Report 97-09, 3F0198-07 dated January 8, 1998).

1999 – Interior inspections were completed. All results were satisfactory. (CR-3 – 90-Day ISI Summary Report, 3F0200-04 dated February 9, 2000).

2001 – Exterior inspection was coordinated with IWL. The complete primary containment pressure boundary (concrete containment) was inspected. The inspections included VT-3C visual method and further investigated with VT-1C for any suspect conditions. Results were evaluated and dispositioned by the Responsible Professional Engineer (Structural) as satisfactory. (CR-3 – 90-Day ISI Summary Report, 3F0102-04, dated January 22, 2002).

2003 – Interior inspection was coordinated with IWE Containment Liner (MC) general visual inspections. This work was comprised of general visual inspection of all accessible metal containment liner plates, penetration and associated attachments, and any emergent issues arising from these inspections. The results were evaluated and dispositioned by the program Responsible Professional Engineer (Structural) as satisfactory. (CR-3 – 90-Day ISI Summary Report, 3F0204-04, dated February 2, 2004).

4.0 SUMMARY OF METHODOLOGY AND PLANT SPECIFIC RISK EVALUATION

The basis for extending the ILRT test interval is provided in Section 11.0 of NEI 94-01, Revision 1. Assessments of the risk impact of extending the ILRT testing interval up to twenty years was recently completed in 2003. Section 11.0 of the revised NEI 94-01 Guidelines addresses the objective of the work concluded in 2003 and published as EPRI Product No. 1009325, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals." The objective of this product was to perform a generic risk impact assessment for optimized ILRT intervals of up to twenty years, utilizing current industry performance data and risk-informed guidance, primarily NRC Regulatory Guide 1.174. The assessment includes current containment leak rate and performance information obtained through an NEI industry wide survey conducted in 2001. The data gathered indicated that there were no failures that could result in a risk-significant early release.

An expert panel was convened to determine risk-significant large failure magnitude and frequency. The expert panel considered defense-in-depth approaches such as alternative inspections to supplement the existing testing programs. Once the expert panel determined large failure magnitude and frequency, the risk impact was determined for two representative plants, a PWR and a BWR. The hypothetical representative plants were assigned a Core Damage Frequency (CDF) equal to 1E-4 per year. This CDF value is assumed to include external events and other hazards.

The risk assessment (EPRI Product No. 1009325) demonstrated that the change in population dose is very small (less than 0.02%). The change in Large Early Release Frequency (LERF) is less than 1E-07. This value meets the RG 1.174 acceptance guidelines for very small changes. NEI 94-01, Revision 1, concludes that those results confirm NUREG-1493 conclusions regarding risk in extending ILRT intervals up to twenty years using current regulatory guidance and risk-informed concepts.

CR-3 Risk Evaluation

Attachment E contains Calculation No. P-04-0001 which assesses the risk associated with implementing a 20-year ILRT interval at CR-3 following the methodology in EPRI Product No. 1009325. The calculation shows that the reduction in Type A ILRT testing (once in 20 years) results in a change in LERF of 9.18E-09 per year. Comparing this risk to the acceptance criteria of RG 1.174, the change is in the "very small region." The calculation also shows a low change in population dose (2.18E-03 person-rem/year or 0.59%) and a low change in conditional containment failure probability of 0.12%.

Quality of the Crystal River Unit 3 PSA

The models used for this application were generated using updated Individual Plant Examination (IPE) models developed in response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," and associated supplements.

The original development work was a level one Probabilistic Risk Assessment (PRA) study completed in 1987 (Crystal River Unit 3 Probabilistic Risk Assessment, Florida Power Corporation, Science Applications Intl. Corporation, July 1987), which was submitted to the NRC and reviewed by Argonne National Laboratory (NUREG/CR-5245). This study was subsequently updated for the Generic Letter 88-20 IPE submittal to include a level two containment analysis and an internal flooding analysis.

Revisions to the models have been made to maintain the models consistent with plant design changes and operational changes. These current changes have been made by individuals knowledgeable in risk assessment techniques and methods, and reviewed by plant Engineering and Operations personnel familiar with the plant design and operation. The current Probabilistic Safety Assessment (PSA) model and the risk assessment performed for this application have been documented as a calculation.

Current administrative controls include written procedures and review of all model changes, data updates, and risk assessments performed using PSA methods and models. Risk assessments are performed by a PSA engineer, reviewed by another PSA engineer, and approved by the PSA Supervisor or designee. Procedures, PSA model documentation, and associated records for applications of the PSA models are controlled documents.

Since the submittal of the original PRA study in 1987, the PSA models have been maintained consistent with the current plant configuration such that they are considered "living" models which reflect the as-built, as-operated plant. The PSA models are updated for different reasons, including plant changes and modifications, procedure changes, accrual of new plant data, discovery of modeling errors, and advances in PSA technology. The update process ensures that the applicable changes are implemented and documented in a timely manner so that risk analyses performed in support of plant operations reflect the current plant configuration, operating philosophy, and transient and component failure history. The PSA maintenance and update process is described in administrative procedure ADM-NGGC-0004, "Updates to PSA Models." Model updates are performed at a frequency dependent on the estimated impact of the accumulated changes. Guidance to determine the need for a model update is provided in the procedure. Prior to startup from a refueling outage, known outstanding changes, including identified model errors and enhancements, are reviewed, and either model changes are implemented, or the outstanding item is dispositioned to be deferred for a future model update.

<u>PSA Software</u>

Computer programs that process PSA model inputs are verified and validated in accordance with administrative procedure CSP-NGGC-2505, "Software Quality Assurance and Configuration Control of Business Computer Systems." This procedure provides for software verification and validation to ensure the software meets the software requirement specifications and functional requirements, and typically includes a comparison of results generated to the results generated from previously approved software.

Validation requirements for each quality related PSA computer program are documented in the Software Life Cycle document, which consists of a Software Verification/Validation Plan (SVVP) and Report (SVVR). These requirements include the method of validation, the frequency of validation, the documentation required and the acceptance criteria. Actual validation benchmark problems can exercise more than one program, but a separate SVVR must be submitted for each program. Each SVVP and SVVR is reviewed, and then approved by the software owner, who is the PSA Supervisor. Software validation tests both the software and the hardware. Validation tests are also performed following any significant change in the hardware, operating system, or program, or if the validation period established in the SVVP procedure expires.

Model Changes Since Submittal of the IPE

Since the submittal of the IPE, there has been several significant plant design changes incorporated into the PSA model which have resulted in a reduction in the Core Damage Frequency (CDF). Updates have been made to plant-specific data (through 1999) and initiating events data, as well as updates to the methods used for human reliability, common cause, internal flooding and level two analyses.

As of the date of this submittal, there are no outstanding or planned plant changes requiring a change to the PSA model which would affect the conclusions of the analysis in Attachment E.

PSA Reviews

As discussed above, the original CR-3 PRA study was reviewed by Argonne National Laboratory as documented in NUREG/CR-5245. For the IPE submittal, multiple levels of review were used, including an assessment by Engineering and Operations personnel familiar with the plant design and operation. Subsequent revisions to the PSA models were performed by qualified individuals with knowledge of PSA methods and plant systems. Involvement by Engineering and Operations personnel in providing input and reviewing the results was obtained, when required, based on the scope of the changes being implemented.

The CR-3 PSA model and documentation was subjected to the industry peer certification review process in September 2001. In preparation for this review, an external consultant was hired to develop system notebook documentation. This required a review of the system models against plant drawings and procedures, and identification of any inconsistencies with the models. Items identified from this review were considered and dispositioned. The internal flooding and common cause failure analyses were updated to current industry methodologies and data sources. An internal review of the PSA model elements and their corresponding documentation was conducted to ensure the model and documentation reflected the plant design.

- The industry peer certification review was conducted by a diverse group of PSA engineers from other Babcock & Wilcox (B&W) plants, industry PSA consultants familiar with the B&W plant design, and a representative from the Institute of Nuclear Power Operations (INPO). The certification review covered all aspects of the PSA model and the administrative processes used to maintain and update the model. This review generated specific recommendations for model changes to correct errors, as well as guidance for improvements to processes and methodologies used in the of plant-specific thermal-hydraulic analyses which provide the basis for accident sequences, system success criteria, and timing for operator actions;
- Revision of accident sequence logic for steam generator tube rupture (SGTR) and anticipated transient without scram (ATWS) mitigation;
- Development of an initiating event to address the loss of all raw water pumps (loss of ultimate heat sink);
- Update of the interfacing systems loss of coolant accident (ISLOCA) analyses;
- Update of the human reliability analysis including the dependency analysis for multiple operator action responses to an event, and
- Update of the level two analysis.

Issues involving model documentation are being addressed as each individual PSA document is reviewed and approved under Progress Energy corporate procedures. Other changes involving guidance documents and administrative processes used for model updates are planned to be addressed by Progress Energy corporate procedures, once the peer review process has been completed for all PSA models (including the Robinson Nuclear Plant, Brunswick Nuclear Plant, and Harris Nuclear Plant). The issues identified by the peer review in these areas have been reviewed and determined not to have any impact on this submittal. Thus, deferral of completion of these items is acceptable for this application of the PSA model. All other peer review items which impact the PSA model have been addressed and are reflected in this submittal.

At the time of the peer review, the level two model was not yet completed, and only a preliminary draft version, along with the original IPE level two results, were available for review. The level two model is now complete, and the findings identified from the peer certification review of the preliminary results and the IPE model have been addressed.

The quality of the CR-3 PRA was recently assessed by the NRC staff. The Safety Evaluation for License Amendments #207, issued June 13, 2003, and License Amendment #212, issued May 18, 2004, indicate that the risk analysis provided by CR-3 as part of those License Amendment Requests was of sufficient quality and satisfied the intent of the applicable sections of RG 1.177, 1.174 and the Standard Review Plan (SRP).

5.0 CONCLUSION

Based on the information provided in this submittal, PEF concludes that the predicted risk due to the proposed change is within the acceptance criteria of RG 1.174 and RG 1.177, and therefore acceptable.

6.0 REFERENCES

1. NEI 94-01, "Nuclear Energy Institute Industry Guideline For Implementing performance-Based Option of 10 CFR 50, Appendix J," Revision 0, July 26, 1995.

2. Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995.

3. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.

4. Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk Informed Decisionmaking: Technical Specifications," August 1998.

5. Draft Revision 1 to NEI 94-01, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, and Final (pre-publication) EPRI Report, Product No. 1009325, Risk Impact Assessment of Extended Integrated Leak Rate Intervals, Accession Number ML041240409, dated April 27, 2004.

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER - UNIT 3

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ATTACHMENT B

LICENSE AMENDMENT REQUEST #283, REVISION 0

Regulatory Analysis

REGULATORY ANALYSIS

License Amendment Request (LAR) #283, Revision 0, proposes a revision to Crystal River Unit 3 (CR-3), Improved Technical Specification (ITS) 5.6.2.20, "Containment Leakage Rate Testing Program," to allow for a 20-year Integrated Leak Rate Test (ILRT) interval.

CR-3 proposes to replace the current Containment Leakage Rate Testing Program interval requirement with the following:

"NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the November 7, 1991 Type A test shall be performed no later than November 6, 2011."

No Significant Hazards Evaluation

(1) Does not involve a significant increase in the probability or consequences of an accident previously analyzed.

The proposed revision to the CR-3 ITS is to change the current interval for Type A (ILRT) testing to an interval of up to 20 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 testing extension is not a modification to plant systems, nor a change to plant operation that could initiate an accident. The proposed extension to Type A testing does not involve a significant increase in the consequences of an accident since research documented in NUREG-1493 found that, generically, very few potential containment leakage paths fail to be identified by Type B and C tests. Additionally, EPRI Product No. 1009325, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," concluded that there is very little risk associated with extension of Type A testing intervals up to twenty years.

CR-3 provides a high degree of assurance, through testing and inspection, that the containment will not degrade in a manner detectable only by Type A testing. Inspections required by the Maintenance Rule and American Society of Mechanical Engineers (ASME) code are performed in order to identify indications of containment degradation that could affect leak tightness. Type B and C testing, required by the CR-3 ITS, will identify any containment opening, such as valves, that would otherwise be detected by the Type A tests. A CR-3 Type A test extension will not represent a significant increase in the consequences of an accident. Other defense in depth measures include monitoring of the containment makeup and venting frequency.

(2) Does not create the possibility of a new or different kind of accident from any accident previously analyzed.

The proposed extension to Type A testing cannot create the possibility of a new or different type of accident since there are no physical changes being made to the plant.

There are no changes to the operation of the plant that could introduce a new failure mode creating the possibility of a new or different kind of accident.

(3) *Does not involve a significant reduction in the margin of safety.*

The proposed extension to Type A testing will not significantly reduce the margin of safety. The NUREG-1493 generic study of the effects of extending containment leakage testing found that a 20-year extension in Type A leakage testing resulted in an imperceptible increase in risk to the public. Additionally, EPRI Product No. 1009325, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," concluded that there is very little risk associated with extension of Type A testing intervals up to twenty years. The containment inspections being performed in accordance with the ASME Code, Section XI, and the Maintenance Rule provide a high degree of assurance that the containment will not degrade in a manner that it is only detectable by Type A testing.

Applicable Regulatory Requirements

PEF has evaluated the Regulatory Requirements applicable to the proposed changes to ITS 5.6.2.20 which include: 10 CFR 50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, 10 CFR 50.55a, Codes and Standards and 10 CFR Part 50, Appendix J. PEF has determined that the proposed change does not require any exemptions or relief from regulatory requirements other that the changes requested to ITS 5.6.2.20.

Environmental Impact Evaluation

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not:

(i) involve a significant hazards consideration,

(ii) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and

(iii) result in a significant increase in individual or cumulative occupational radiation exposure.

PEF has reviewed proposed License Amendment Request #283, Revision 0, and concludes it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with this request.

PROGRESS ENERGY FLORIDA, INC.

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ATTACHMENT C

LICENSE AMENDMENT REQUEST #283, REVISION 0

Proposed Revised Improved Technical Specification Pages - Strikeout/Shadowed

5.6 Procedures, Programs and Manuals

5.6.2.19 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

c. The reactor vessel pressure and temperature limits, including those for heatup and cooldown rates, shall be determined so that all applicable limits (e.g., heatup limits, cooldown limits, and inservice leak and hydrostatic testing limits) of the analysis are met.

d. The PTLR, including revisions or supplements thereto, shall be provided upon issuance for each reactor vessel fluency period.

5.6.2.20 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 exception:

1. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the November 7, 1991 Type A test shall be performed no later than November 6, 20~~06~~11.

The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 54.2 psig. The containment design pressure is 55 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.25% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

1. 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 Type C Tests and $\leq 0.75\ L_a$ for Type A Tests.

2. Air lock testing acceptance criteria are:

   a. Overall air lock leakage range is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   b. For each door, leakage rate is $\leq 0.01\ L_a$ when tested at $\geq$ 8.0 psig.

   The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

   The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

(continued)

PROGRESS ENERGY FLORIDA, INC.

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ATTACHMENT D

LICENSE AMENDMENT REQUEST #283, REVISION 0

Proposed Improved Technical Specification Pages
- Revision Bar Format

5.6 Procedures, Programs and Manuals

5.6.2.19 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

c. The reactor vessel pressure and temperature limits, including those for heatup and cooldown rates, shall be determined so that all applicable limits (e.g., heatup limits, cooldown limits, and inservice leak and hydrostatic testing limits) of the analysis are met.

d. The PTLR, including revisions or supplements thereto, shall be provided upon issuance for each reactor vessel fluency period.

5.6.2.20 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, Revision 1, "Performance-Based Containment Leak Test Program," dated September 1995, as modified by the following exception:

1. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the November 7, 1991 Type A test shall be performed no later than November 6, 2011.

The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 54.2 psig. The containment design pressure is 55 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.25% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

1. 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 Type C Tests and $\leq 0.75\ L_a$ for Type A Tests.

2. Air lock testing acceptance criteria are:

   a. Overall air lock leakage range is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   b. For each door, leakage rate is $\leq 0.01\ L_a$ when tested at ≥ 8.0 psig.

   The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

   The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

(continued)

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER - UNIT 3

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ATTACHMENT E

LICENSE AMENDMENT REQUEST #283, REVISION 0

Risk Assessment For 20 Year ILRT Interval

SYSTEM #	N/A
CALC. SUB-TYPE	N/A
PRIORITY CODE	4
QUALITY CLASS	Nonsafety

NUCLEAR GENERATION GROUP

P-04-0001
(CALCULATION #)

RISK ASSESSMENT FOR 20 YEAR ILRT INTERVAL
(Title including structures, systems, components)

☐ BNP UNIT ____

☒ CR3 ☐ HNP ☐ RNP ☐ NES ☐ ALL

APPROVAL

REV	PREPARED BY	REVIEWED BY	SUPERVISOR
0	Signature D. Mi[illegible]	Signature Andrew J. Howe	Signature BAMorgen
	Name David N. Miskiewicz	Name Andrew J. Howe	Name Bruce A. Morgen
	Date 6/15/04	Date 6-16-04	Date 6-18-04

(For Vendor Calculations)

Vendor ____________________ Vendor Document No. ____________

Owner's Review By ____________________ Date ____________

LIST OF EFFECTIVE PAGES

PAGE	REV	PAGE	REV	ATTACHMENTS		
i-v	0			Number	Rev	Number of Pages
1-10	0					
				AMENDMENTS		
				Letter	Rev	Number of Pages

Rev. #	Revision Summary (list of ECs incorporated)
0	Original.

Document Indexing Table

Document Type (e.g. CALC, DWG, TAG, PROCEDURE, SOFTWARE)	**ID Number** (e.g., Calc No., Dwg. No., Equip. Tag No., Procedure No., Software name and version)	**Function** (i.e. IN for design inputs or references; OUT for affected documents)	**Relationship to Calc.** (e.g. design input, assumption basis, reference, document affected by results)	**Action** (specify if Doc. Services or Config. Mgt. to Add, Deleted or Retain) (e.g., CM Add, DS Delete)
CALC	P-02-0001, Rev.1	IN	Design input, Reference	DS Add
CALC	F-01-0001, Rev. 2	IN	Design input, Reference	DS Add

Record of Lead Review

Design P-04-000X **Revision** 0

The signature below of the Lead Reviewer records that:
- the review indicated below has been performed by the Lead Reviewer;
- appropriate reviews were performed and errors/deficiencies (for all reviews performed) have been resolved and these records are included in the design package;
- the review was performed in accordance with EGR-NGGC-0003.

☐ **Design Verification Review** ☒ **Engineering Review** ☐ **Owner's Review**
- ☐ Design Review
- ☐ Alternate Calculation
- ☐ Qualification Testing

☐ **Special Engineering Review** ______

☐ YES ☐ N/A **Other Records are attached**

Andrew Howe	Andrew Howe	PSA	6-16-2004
Lead Reviewer (print)	**(sign)**	**Discipline**	**Date**

Item No.	Deficiency	Resolution
	No deficiencies	

FORM EGR-NGGC-0003-2-5
This form is a QA Record when completed and included with a completed design package. Owner's Reviews may be processed as stand alone QA records when Owner's Review is completed

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Record of Interdisciplinary Reviews

PART I — DESIGN ASSUMPTION / INPUT REVIEW: APPLICABLE ☐ Yes ☒ No

The following organizations have reviewed and concur with the design assumptions and inputs used in this calculation:

	Name	Signature	Date
Systems Engineering			
Operations			
Other			

PART II — RESULTS REVIEW:

The following organizations are aware of the impact of the results of this calculation (on designs, programs and procedures):

	Name	Signature	Date
Systems Engineering ☐ Yes ☒ NO			
Comments:			
Operations ☐ Yes ☒ NO			
Comments:			
Engineering Services	Tim Howard	Tim Howard	6/15/04
Comments:			
Licensing	Loretta V Cecilia	Loretta V Cecilia	6/21/04
Comments:			

TABLE OF CONTENTS

1.0 Purpose

This calculation assesses the risk associated with implementing a 20 year ILRT interval at CR3 based on the methodology presented in EPRI Technical Report 1009325. This assessment will be used to support a regulatory submittal, and will be based on elements of RG 1.174.

2.0 References

1. CR3 calculation P-02-0001, Rev.1, "CR3 PSA - Model of Record - MOR03a", June 2004
2. CR3 calculation F-01-0001, Rev.2, "Evaluation of Risk Significance of ILRT Extension", June 2001
3. EPRI Technical Report 1009325, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals", December 2003
4. NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", July 1998
5. CR3 Improved Technical Specifications, Through Amendment 212
6. NUREG 1493, "Performance Based Containment Leak-testing Programs", September 1995
7. EPRI Technical Report 105396, "PSA Applications Guide", August 1995
8. NEI 94-01, Rev.1 Draft
9 NRC Regulatory Guide 1.177, "An Approach for Plant Specific, Risk Informed Decision Making: Technical Specifications", July 1998

3.0 Design Inputs

The inputs for this analysis are contained in the references listed in section 2.0.

The EPRI methodology (Ref. 3) defines 8 accident classes for use in the analysis as presented below. Two of the classes (3a and 3b) are specifically created to assess the risk of ILRT intervals.

Class No.	Description
1	Containment intact
2	Large containment isolation failures
3a	Small pre-existing leak in containment
3b	Large pre-existing leak in containment
4	Small isolation failure – failure to seal – (Type B test)
5	Small isolation failure – failure to seal – (Type C test)
6	Containment isolation failures (dependent failures personnel errors)
7	Severe accident phenomena included failures (early and late containment failures)
8	Containment bypass (SGTR, MSIV leakage and ISLOCA)

For each class, a base frequency and population dose are determined by binning the level 2 PRA release category/containment end-state results. The following table contains the CR3 PRA Model of Record (Ref. 1) level 2 results, with the associated EPRI accident class.

Containment End State		Frequency	EPRI Accident Class
LERF	Early Failure	6.34E-09	7
	Isolation (lg)	0.00E+00	2
	Bypass (sm)	3.14E-07	8
	Bypass (lg)	5.14E-08	8
Non-LERF	Late Failure	6.47E-08	7
	Isolation (sm)	3.41E-07	2
	Failure (IVR)	2.55E-07	7
Intact	Intact	6.55E-06	1
	Total	7.58E-06	

Reference 2 determined population doses based on a level 3 PRA study performed for CR3 and supporting analysis. This data is still considered valid for the current assessment. The following table presents the data for CR3 by EPRI accident class.

EPRI Accident Class	Population Dose (Person-Rem) (Ref. 2)
1	987
2	658000
3a	na
3b	na
4	na
5	na
6	na
7	197000
8	202000

Based on Reference 5, section 5.6.2.20, the acceptance criteria for containment leakage is 0.25 weight percent per day (La).

4.0 Assumptions

1. The risk associated with the ILRT interval has no impact on CDF.
2. Failures detectable only by ILRT can be large enough to contribute to LERF.
3. The population dose rates used in Reference 2 are applicable for this assessment.
4. ILRT interval does not have a significant impact on external events such as fire.
5. The probability of a pre-existing leak in containment is a function of the time between surveillance tests.
6. The probability of leakage at 100 La from Reference 3, is the probability of a leak greater than or equal to 100 La based on the write-up provided in section 6.2 of EPRI report.

5.0 Calculation/Analysis Details

5.1 Baseline CDF and Dose

Based on the information discussed in Section 3.0, the following table provides the relevant baseline data to perform the risk assessment. Consistent with the EPRI methodology (Ref.3), accident classes 4, 5, and 6 are not affected by the optimization of the ILRT interval and therefore are not required for the evaluation.

Accident Class	Base Frequency	Population Dose (Person-Rem)
1	6.55E-06	987
2	3.41E-07	658000
3a		
3b		
7	3.26E-07	197000
8	3.65E-07	202000
Totals	7.58E-06	

The base frequency values are entered in Summary Table 5.3 in the column labeled "5.1".

5.2 EPRI Methodology

The EPRI methodology presents the evaluation as a nine step process. The details are provided in section 7 of reference 3.

5.2.1 Step 1: Determination of Containment Leakage Rates (EPRI Step 2)

In the determination of the leakage rates that correspond to the 3a and 3b accident classes, previous submittals (Ref. 2) have chosen values of 10La to represent small containment leakage and 35La to represent large containment leakage. The EPRI methodology (Ref. 3) proposes a value of 100La to conservatively represent large early release frequency.

LERF is defined by the EPRI PSA Applications Guide (Ref. 6) based on an "unscrubbed containment failure pathway of sufficient size to release the contents of the containment within one hour..". Based on the allowable leakage (La) of 0.25 weight% per day for CR3, the expected leakage rate would be need to be 9600 La to release one volume in one hour, or 400 La to release one containment volume per day. Based on this, 100 La can be considered very conservative for treatment as a LERF contributor.

5.2.2 Step 2: Adjusted Baseline Risk Determination (EPRI Step 1)

In this step the risk is determined for original ILRT interval of 3 tests in 10 years. Reference 3 uses an expert elicitation process to develop probabilities of occurrence for a range of leak sizes. Based on Table 6-1 of reference 3 the probability of a small leakage pathway (accident class 3a) is 3.88E-03 based on an La of 10, and a large pre-existing leak in containment (accident class 3b) is 2.47E-04 based on an La of 100.

The resulting frequency of accident class 3a is equal to:

Frequency of Class 3a = Class 1 Freq. * Class 3a leakage probability

= 6.55E-06 per year * 3.88E-03

= 2.54E-08 per year

The resulting frequency of accident class 3b is equal to

Frequency of Class 3b = Class 1 Freq. * Class 3b leakage probability

= 6.55E-06 per year * 2.47E-04

= 1.62E-09 per year

Subtracting these values from the intact class (accident class 1) provides conservatism of total CDF.

Adjusted Class 1 Frequency = Class 1 Freq. – (Class 3a + Class 3b) Freq.

= 6.55E-06 per year – (2.54E-08 + 1.62E-09) per year

= 6.52E-06 per year

The other accident classes are not affected.

The above frequencies are entered in Summary Table 5.3 in the column labeled with "5.2.2".

5.2.3 Step 3: Develop the Baseline Population Dose

As discussed in section 3 the Class 1 dose is 987 person-REM. The associated doses for Class 3a and Class 3b based on the respective La values of 10 and 100 are:

Class 3a Dose = Class 1 dose * 10 = 987 person-REM * 10 = 9870 person-REM

Class 3b Dose = Class 1 dose * 100 = 987 person-REM * 100 = 98700 person-REM

These results are entered in Summary Table 5.3 along with the dose data in section 5.1 in the column labeled with "5.2.3".

5.2.4 Step 4: Determine Baseline Dose Rate

The dose rate is calculated by multiplying the dose (5.2.3) by the adjusted frequency of occurrence of the accident class (5.2.2). The results are provided in the Summary Table 5.3 in the column labeled with "5.2.4".

5.2.5 Step 5: Determine Change in Pre-Existing Leak Probability & Revised Frequencies

The change in the pre-existing leak probability is based on the assumption that the pre-existing leak is a function of the time between surveillance tests (ILRT). The method used in NUREG 1493 and endorsed by Reference 3 states that relaxing the ILRT frequency from three in 10 years to one in ten years will increase the average time that a leak that is detectable only by ILRT goes undetected from 18 to 60 months (1/2 the surveillance interval), a factor of 60/18 = 3.33 increase.

Therefore, relaxing the ILRT testing interval from three in 10 years to one in 20 years will increase the average time that a leak that is detectable only by ILRT goes undetected from 18 to 120 months (1/2 the surveillance interval), a factor of 120/18 = 6.67 increase.

For a 20 year interval, the revised frequencies of a Class 3a and Class 3b leaks will be based on an increase by a factor of 6.67, and the revised Class 1 frequency will decrease by an equivalent amount to conservatively preserve the total CDF.

The results are entered in Summary Table 5.3, in the columns labeled with "5.2.5" for each extended interval evaluated.

5.2.6 Step 6: Determine Revised Population Dose Rate

The revised population dose rate for a 20 year ILRT interval is calculated by multiplying the base population dose (5.2.3) by the revised frequencies (5.2.5). The results for each accident class are summed to obtain the Revised Total Dose Rate.

The results are entered in Summary Table 5.3, in the columns labeled with "5.2.6" for each extended interval evaluated.

5.2.7 Step 7: Determine Percentage Increase In Population Dose

The percentile change in dose rate is calculated for a given ILRT interval by subtracting the total baseline dose rate (5.2.4) from the total interval dose rate (5.2.6) and dividing by the baseline.

The results are entered in Summary Table 5.3, in the row labeled with "5.2.7".

5.2.8 Step 8: Evaluate Change In LERF

The risk associated with extending the ILRT interval involves a potential that a core damage event that normally would result in only a small radioactive release from containment could result in a large release due to an undetected leak path existing during the extended interval. This potential is conservatively represented in accident Class 3b, as discussed in 5.2.1. The change in LERF is calculated for a given ILRT interval by subtracting the adjusted baseline Class 3b frequency (5.2.2) from the interval Class 3b frequency (5.2.5).

The results are entered in Summary Table 5.3, in the row labeled with "5.2.8".

5.2.9 Step 9: Evaluate Conditional Containment Failure Probability (CCFP)

The conditional containment failure probability (CCFP) is defined as the probability of containment failure given the occurrence of a core damage accident. CCFP is calculated for a given ILRT interval by subtracting the interval Class 1 frequency (5.2.5) divided by the total CDF (5.2.5) from 1.0.

CCFP = 1 – (Interval Class 1 frequency / total CDF for all Classes)

The results are entered in Summary Table 5.3, in the row labeled with "5.2.9".

5.3 Summary Table

			ILRT Interval									
			3 per 10 Years (Baseline)				1 per 10 Years			1 per 20 Years		
Accident Class	Description	5.1 Base Frequency	5.2.2 Adjusted Base Frequency	5.2.3 Population Dose (person-rem)	5.2.4 Population Dose Rate (Person-rem/rx-yr)	Percent of Total Dose Rate	5.2.5 Frequency	5.2.6 Population Dose Rate (Person-rem/rx-yr)	Percent of Total Dose Rate	5.2.5 Frequency	5.2.6 Population Dose Rate (Person-rem/rx-yr)	Percent of Total Dose Rate
1	Containment Intact	6.55E-06	6.52E-06	987	6.44E-03	1.743%	6.46E-06	6.38E-03	1.722%	6.37E-06	6.29E-03	1.692%
2	Large containment isolation failures	3.41E-07	3.41E-07	658000	2.24E-01	60.763%	3.41E-07	2.24E-01	60.617%	3.41E-07	2.24E-01	60.407%
3a	Small pre-existing leak in containment	N/A	2.54E-08	9870	2.51E-04	0.068%	8.47E-08	8.36E-04	0.226%	1.70E-07	1.67E-03	0.450%
3b	Large pre-existing leak in containment	N/A	1.62E-09	98700	1.60E-04	0.043%	5.39E-09	5.32E-04	0.144%	1.08E-08	1.07E-03	0.287%
4	Small isolation failure - failure to seal (Type B test)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
5	Small isolation failure - failure to seal (Type C test)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
6	Containment isolation failures (dependent failures personnel errors)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
7	Severe accident phenomena induced failures (early and late containment failures)	3.26E-07	3.26E-07	197000	6.43E-02	17.401%	3.26E-07	6.43E-02	17.359%	3.26E-07	6.43E-02	17.299%
8	Containment bypass (SGTR, MSIV leakage and ISLOCA)	3.65E-07	3.65E-07	202000	7.38E-02	19.981%	3.65E-07	7.38E-02	19.933%	3.65E-07	7.38E-02	19.864%
Totals		7.58E-06	7.58E-06	1.17E+06	3.70E-01	100.000%	7.58E-06	3.71E-01	100.000%	7.58E-06	3.72E-01	100.000%
Change in Dose Rate			N/A				8.95E-04			2.18E-03		
% Change in Dose Rate (5.2.7)			N/A				0.24%			0.59%		
dLERF (5.2.8)			N/A				3.77E-09			9.18E-09		
CCFP (5.2.9)			13.64%				13.69%			13.76%		
dCCFP			NA				0.05%			0.12%		

5.4 Sensitivity to Population Dose

A sensitivity case was run using the population doses from EPRI report (Ref. 3) for the hypothetical plant application. These doses are significantly higher than those used for the CR3 analysis.

EPRI Accident Class	Population Dose (Person-Rem) (Ref. 2)	Population Dose (Person-Rem) (Ref. 3)
1	9.87E+02	9.79E+05
2	6.58E+05	4.90E+08
3a	na	
3b	na	
4	na	
5	na	
6	na	
7	1.97E+05	1.37E+09
8	2.02E+05	2.52E+09

Applying the higher doses to the assessment process discussed in section 5.2 yields the following results compared with the actual CR3 results:

Figure of Merit	CR3 Results (Section 5.3)	Sensitivity Results
Change in Dose Rate	2.18E-03	2.16E+00
% Change in Dose Rate	0.59%	0.08%
dLERF	9.18E-09	9.18E-09
CCFP	13.76%	13.76%
dCCFP	0.12%	0.12%

There is no impact to LERF or CCFP. This is because these values are based on the frequencies only, and are not directly impacted by dose in this analysis. There is an increase in Dose rate proportional to the increase in dose. This is logical, since areas with higher populations can impact more people. Finally, the higher doses yielded a lower percent change in dose rate. This is due to the much greater total dose rate in the base case.

6.0 Results / Conclusions

This analysis is consistent with previous assessments which demonstrate that the quantitative risk of extending the ILRT interval is very low. The ILRT has no impact on CDF and is one of several tests and inspections used to ensure containment integrity.

Using relatively conservative inputs for the leakage and timing associated with a large early release, the reduction in Type A ILRT test results in a change in LERF of 9.18E-09 per year based on a 20 year interval at CR3. Comparing this risk to the acceptance criteria of Regulatory Guide 1.177 (5.0E-08), and Regulatory Guide 1.174 (1.0E-07 per year), the change is "small", or in the "very small" region.

Other figures-of-merit that support the low risk conclusion are the low change in population dose (2.18E-03 person-rem/yr or 0.59%), and the low change in conditional containment failure probability (0.12%).

Based on the sensitivity case run in section 5.4, the assumption used for population dose does not impact the conclusions of this evaluation.