Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37384-2000

October 9, 2001

TVA-SQN-TS-01-10 10 CFR 50.90

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

Gentlemen:

In the Matter of) Docket No. 50-328
Tennessee Valley Authority)

SEQUOYAH NUCLEAR PLANT (SQN) - UNIT 2 - TECHNICAL SPECIFICATION (TS) CHANGE NO. 01-10, "ONE-TIME FREQUENCY EXTENSION FOR TYPE A TEST (CONTAINMENT INTEGRATED LEAK RATE TEST [CILRT])"

Reference: NRC letter to TVA dated February 5, 1996, "Issuance of Technical Specification Amendments for the Sequoyah Nuclear Plant, Units 1 and 2, (TAC Nos. M94239 and M94240)(TS 95-24)"

In accordance with the provisions of 10 CFR 50.4 and 50.90, TVA is submitting a request for an amendment to SQN License DPR-79 to change the TSs for Unit 2. The proposed change revises TS 6.8.4.h, "Containment Leakage Rate Testing Program," to allow a one-time 5-year extension to the current 10-year test interval for the performance-based leakage rate test program for 10 CFR 50, Appendix J, Type A tests (i.e., CILRTs).

The proposed change is submitted on a risk informed basis as described in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis." TVA performed a risk evaluation using Revision 1 of SQN's Probabilistic Safety Assessment (PSA). The conclusion of TVA's risk evaluation determined that a 5-year increase to

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the Type A test interval would result in a net increase in the Large Early Release Frequency (LERF) of less than 1.0E-7/reactor year (1.1 percent). In accordance with the guidance in NRC Regulatory Guide (RG) 1.174, this is considered non-risk significant. In addition, the net change from all releases (small, large, early and late) increases by 3.5E-7/reactor year or 1.6 percent and the population dose increases by 7.72 person-rem. Although no specific criteria is stated in RG 1.174 for "all releases" and dose, these increases are also "very small" and are considered to be non-risk significant. In addition to TVA's risk assessment, the proposed change is based on performance history from previous Type A tests and SQN's American Society of Mechanical Engineers (ASME) Section XI, Subsection IWE examination and inspection program.

TVA's application represents a cost beneficial licensing change. Performance of a Type A test imposes a significant expense to TVA (approximately $265,000) while the safety benefit of performing a test within 10 years versus 15 years is minimal.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the change is exempt from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). The SQN Plant Operations Review Committee and the SQN Nuclear Safety Review Board have reviewed this proposed change and determined that operation of SQN Unit 2, in accordance with the proposed change, will not endanger the health and safety of the public. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter to the Tennessee State Department of Public Health.

Enclosure 1 to this letter provides the description and evaluation of the proposed change. This includes TVA's determination that the proposed change does not involve a significant hazards consideration, and is exempt from environmental review. Enclosure 2 contains a copy of the appropriate TS page from Unit 2 marked up to show the proposed change. Enclosure 3 forwards the revised TS page for Unit 2 which incorporate the proposed change. Enclosure 4 contains the TVA evaluation of risk significance.

TVA's enclosed risk evaluation is consistent with similar assessments performed for New York Power Authority's Indian Point 3 Plant and Florida Power's Crystal River 3 Plant.

TVA requests NRC review and approval prior to the SQN Unit 2 Cycle 11 refueling outage (scheduled to begin in April 2001) to support TVA's schedule needs for this outage. Should you require additional information or clarification, please contact us as soon as possible.

No new commitments have been made as a result of this letter.

TVA requests that the revised TS be made effective within 45 days of NRC approval. This letter is being sent in accordance with NRC RIS 2001-05. If you have any questions about this change, please telephone me at (423) 843-7170 or J. D. Smith at (423) 843-6672.

Sincerely,

Pedro Salas
Licensing and Industry Affairs Manager

Subscribed and sworn to before me
on this 9th day of October

Peggy M. Billingsley
Notary Public

My Commission Expires October 9, 2002

Enclosures

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNIT 2
DOCKET NO. 328

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE NO. 01-10
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

I. DESCRIPTION OF THE PROPOSED CHANGE

TVA's proposed change revises SQN Unit 2 TS to include a one-time 5-year deferral of the Containment Integrated Leak Rate Test (CILRT), also referred to as the 10 CFR 50, Appendix J, Type A test.

SQN TS Section 6.8.4.h, "Containment Leakage Rate Testing Program," contains the general 10 CFR 50, Appendix J test and leakage requirements for the SQN containment structure. The SQN TS refers to requirements contained in 10 CFR 50, Appendix J, Option B and NRC Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak Test Program," dated September 1995. The RG endorses Nuclear Energy Institute (NEI) 94-01, Revision 0, entitled "Industry Guideline For Implementing Performance Based Option of 10 CFR 50, Appendix J," which requires that Type A tests be performed "at least once per 10-years based on acceptable performance history." TVA's proposed change requests, on a one-time basis, an extension to the current 10-year test interval to allow a 15-year test interval (i.e., extend up to 5 years from the spring 2002 to no later than spring 2007). Accordingly, SQN TS Section 6.8.4.h, "Containment Leakage Rate Testing Program," is revised to add the following provision:

"Performance of the spring 2002 containment integrated leakage rate (Type A) test may be deferred up to 5 years but no later than the spring 2007."

II. REASON FOR THE PROPOSED CHANGE

The last SQN Unit 2 Type A test was conducted in April 1992 during the Unit 2 Cycle 5 refueling outage. In accordance with the current SQN TS requirements, Unit 2 is required to perform the next 10-year CILRT during the upcoming Unit 2 Cycle 11 refueling outage (currently scheduled for spring 2002). The cost to TVA for performing a CILRT is substantial (estimated cost is $265,000) and involves approximately 36 hours of critical path time to perform the test. The reason for TVA's proposed change is to

defer the cost of this testing and to save critical path time during the upcoming SQN Unit 2 Cycle 11 refueling outage. Deferral of the Type A test to one of SQN's subsequent refueling outages will allow TVA to evaluate options for performing Type A testing during non-critical path schedules. In addition, deferral of the Type A test from the Unit 2 Cycle 11 refueling outage schedule will reduce the critical path time and provide an immediate cost savings to TVA in terms of replacement power. The total cost deferment is estimated to exceed one million dollars.

III. SAFETY ANALYSIS

Background

The SQN primary containment structure for Units 1 and 2 consists of a freestanding steel vessel with an ice condenser and a separate secondary containment that is a reinforced concrete shield building. The primary containment vessel consists of a cylindrical wall, a hemispherical dome, and a bottom liner plate encased in concrete. SQN Final Safety Analysis Report (FSAR) Figure 3.8.2-1 shows the outline and configuration of the containment vessel. Section 6.2.1 of the SQN FSAR describes SQN's containment design features.

The SQN TS (Section 6.8.4.h) establishes the requirements for implementing a program to perform containment leakage rate testing in accordance with 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B. The types of containment leakage tests include Type A (Containment Integrated Leakrate Test), Type B (local leakrate testing for containment penetrations, hatches, personnel air locks, electrical penetrations, etc.) and Type C (local leakrate testing for containment isolation valves). SQN's maximum allowable containment leakage rate is 1.0 L_a which is defined as 0.25 percent of the containment free air volume per day at an accident pressure of 12.0 pounds per square inch.

Implementation of 10 CFR 50, Appendix J, Option B

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage through the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage value specified in the SQN TSs (L_a). The limitation of containment leakage provides assurance that the containment would perform its design function following an accident.

The 10 CFR 50, Appendix J rule was revised (effective October 26, 1995) to allow licensees to choose containment leakage testing under Option A, "Prescriptive Requirements" or Option B, "Performance-Based Requirements." TVA requested a license amendment for SQN to allow implementation of Option B and was granted approval by NRC letter dated February 5, 1996. The SQN TS was subsequently revised to include Option B. The SQN TS revision included a reference to NRC RG 1.163 for performing Type A, B, and C testing. RG 1.163 specifies a method acceptable to NRC for complying with Option B by endorsing the use of NEI 94-01 and ANSI/ANS 56.8-1994, subject to specific regulatory positions in the RG.

Exceptions to the requirements of RG 1.163 are allowed by 10 CFR 50, Appendix J, Option B, Section V.B, "Implementation," which states:

> *The Regulatory Guide or other implementing document used by a licensee, or applicant for an operation license, to develop a performance based leakage-testing program must be included, by general reference, in the plant technical specifications. The submittal for technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide.*

Based on the provisions above, TVA is not required to file an exemption to 10 CFR 50, Appendix J, Option B.

The adoption of the Option B performance-based containment leakage rate testing program did not alter the basic method by which Appendix J leakage rate testing is performed, but it did alter the frequency of measuring primary containment leakage in Type A, B and C tests. Frequency is based upon an evaluation which looks at the "as found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained. The changes to the Type A test frequency did not directly result in an increase in containment leakage. Similarly, the proposed change to the Type A test frequency will not directly result in an increase in containment leakage.

The allowed frequency for testing was based upon a generic evaluation documented in NUREG-1493, "Performance-Based Containment Leakage-Test Program." Section 10.1.2 of this NUREG provided the following observations with regard to the Type A test frequency:

Reducing the frequency of Type A tests (ILRTs) from the current three per 10 years to one per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential leakage paths that cannot be identified by Type B and C testing and the leaks that have been found by Type A tests have been only marginally above the existing requirements.

Given the insensitivity of risk to containment leakage rate (Chapter 5) and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk.

The findings to date strongly support earlier indications that Type B and C testing can detect a very large fraction of containment leaks. The fraction of leaks that can be detected only by integrated containment leakage test is small, on the order of a few percent.

The proposed change is submitted on a risk informed basis as described in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis." TVA performed a risk evaluation using Revision 1 of SQN's Probabilistic Safety Assessment. TVA's evaluation is documented in a calculation (SQS20211) and is provided in Enclosure 4. The conclusion of TVA's risk evaluation determined that a 5-year extension of the Type A test interval would result in a net increase in the Large Early Release Frequency (LERF) of less than 1.0E-7/reactor year. In accordance with the guidance in RG 1.174, this is considered non-risk significant. In addition, the net change from all releases (small, large, early and late) increases by 3.5E-7/reactor year or 1.6 percent and the population dose increases by 7.72 person-rem. Although no specific criteria is stated in RG 1.174 for "all releases" and dose, these increases are also "very small" and are considered to be non-risk significant.

Current Test Interval Under Option B

The test frequency for Type A testing is stated in NEI 94-01, "Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per 10 years based on acceptable performance history. Acceptable performance history is defined as completion of

two consecutive periodic Type A tests where the calculated performance leakage rate was less than 1.0 La." Also included with NEI 94-01 is consideration of Plant-Specific Testing Program Factors described in Section 11.3. Based on SQN's Unit 2 test history and performance, SQN's current test interval is currently once every 10 years.

Test History Information

Previous Unit 2 Type A test results have shown leakage to be below the 1.0 L_a leakage limit. Margins to date from previous tests indicate at least 10 percent margin (worst case). Accordingly, the proposed extension of the Type A test for Unit 2 represents minimal risk for increased leakage. The risk is further minimized by continued 10 CFR 50, Appendix J Type B and Type C testing. SQN's inservice inspection (ISI) program and maintenance rule inspections provide additional confidence in containment structural integrity and leak tightness.

Containment Penetrations with Mechanical Bellows

The SQN containment penetration mechanical bellows are within the scope of containment inspection and Appendix J Type A, B or C leak testing and are two-ply laminated testable bellows. Each bellow is local leak rate tested (Type B) by pressurizing between the two plies. These bellows incorporate a screen mesh between the inner and outer plies to ensure separation is maintained. This design prevents a "pinch" from occurring at the folds and ensures that the entire space between the plies is pressurized and leak tested during Type B testing.

Following the issuance of NRC Information Notice 92-20, a representative sample of bellows was tested at TVA's SQN and Browns Ferry (identical design) plant sites to confirm adequate separation and communication exists across the entire testable volume. This test verified flow through the annulus between the plies of the bellows.

Option B of 10 CFR 50, Appendix J would allow extended test intervals up to 120 months for Type B components (bellows), based on acceptable performance. Due to industry concerns, SQN has limited extended test intervals for bellows to 60 months. Additionally, penetrations with bellows are tested on a staggered basis such that a portion are tested each refueling outage.

A review of TVA records since 1979 has revealed no failures of these bellow tests for either SQN Unit 1 or Unit 2.

Plant Operational Performance

During power operation, instrument air from air-operated valves is vented inside containment and provides pressurization of the containment structure. Instrumentation monitors containment pressure and annunciation is provided for conditions approaching the limits allowed by the TSs. This cycling of the containment pressure during operation amounts to periodic integrated pressure testing of the containment structure at low differential pressures. Although pressurization is not as significant as would be created during a design basis accident, pressurization of containment does provide assurance that the containment structure is leak tight. The periodic cycling of containment pressure also complements the visual inspection of interior and exterior boundaries in the containment structure that may be inaccessible for visual examination.

American Society of Mechanical Engineers (ASME) Code Examination and Inspection (Subsection IWE)

TVA engineers and inspectors perform inspection activities on the containment structure to support performance of the required Type A test. SQN also performs containment inspections in accordance with the ASME Section XI IWE ISI program. The IWE program will continue to perform inspection activities on SQN Unit 2 containment through the proposed Appendix J test extension interval.

TVA has performed visual examinations of the Unit 2 metal containment in accordance with TVA's IWE program. To date, no major indications of containment degradation have been found. These periodic IWE inspections provide assurance that degradation of the containment structure will be detected and corrected before it can affect the structural integrity or leak tightness.

IV. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

TVA has concluded that operation of Sequoyah Unit 2, in accordance with the proposed change to the TS, does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

TVA's proposed revision to the Sequoyah Unit 2 technical specifications (TSs) adds notation to TS Section 6.8.4.h, "Containment Leakage Rate Testing Program," to allow a one-time 5-year extension to the current 10-year interval for 10 CFR 50, Appendix J, Type A testing.

A. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed extension to Type A testing does not increase the probability of an accident previously evaluated since the change is not a modification to plant systems, nor a change to plant operation that could initiate an accident.

TVA performed an evaluation of the risk significance for the proposed increase to the Sequoyah Unit 2 Type A test frequency. The results of the TVA evaluation indicate that the increase in Large Early Release Frequency (LERF) remains below the level of risk significance defined in NRC Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis." TVA's evaluation indicates that the increase in frequency for all releases (small, large, early and late) and the increase in radiation dose to the population is non-risk significant (3.5E-7/reactor year and 7.72 person-rem, respectively).

The proposed test interval extension does not involve a significant increase in the consequences of an accident because research documented in NUREG-1493 determined that generically, very few potential containment leakage paths fail to be identified by Type A tests. An analysis of 144 Type A test results, including 23 failures, found that no failures were due to containment liner breach. The NUREG concluded that reducing the Type A test frequency to once per 20 years would lead to an imperceptible increase in risk. Furthermore, the NUREG concluded that Type B and C

testing provides assurance that containment leakage from penetration leak paths (i.e., valves, flanges, containment air-locks) identify any leakage that would otherwise be detected by the Type A tests.

In addition to the NUREG conclusions, TVA's American Society of Mechanical Engineers (ASME) IWE program performs containment inspections periodically in order to detect evidence of degradation that may affect either the containment structural integrity or leak tightness. Accordingly, TVA's proposed extension of the Type A test interval does not increase the probability or consequences of an accident previously evaluated.

B. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change to extend the Type A test interval does not create the possibility of a new or different type of accident since there are no physical changes made to the plant. There are no changes to the operation of the plant that would introduce a new failure mode creating the possibility of a new or different kind of accident.

C. The proposed amendment does not involve a significant reduction in a margin of safety.

The proposed change to extend the Type A test interval will not significantly reduce the margin of safety. A generic study documented in NUREG-1493 indicates that extending the Type A leak test interval to 20 years would result in an imperceptible increase in risk to the public. The NUREG also found that, generically, the containment leakage rate contributes a very small amount to the individual risk and that the decrease in the Type A test frequency would have a minimal affect on risk because most potential leakage paths are detected by Type C testing.

Previous Type A leakage tests conducted on Sequoyah Unit 2 indicate that leakage from Unit 2 containment has been less than the 10 CFR 50, Appendix J leakage limit of 1.0 L_a. A review of previous Unit 2 Type A test results indicate at least a 10 percent margin exists below the 1.0 L_a leakage limit. These test results provide assurance that the proposed extension to the Type A test interval would not significantly reduce the margin of safety.

V. ENVIRONMENTAL IMPACT CONSIDERATION

The proposed change does not involve a significant hazards consideration, a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
SEQUOYAH PLANT (SQN)
UNIT 2

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE
MARKED PAGES

I. AFFECTED PAGE LIST

Unit 2

6-9

II. MARKED PAGES

See attached.

6.8.4 f. Radioactive Effluent Controls Program (Cont.)

of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the SITE BOUNDARY SHALL BE LIMITED to the following:

1. For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the total body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and
2. For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/year to any organ.

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radio-nuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50, and

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

g. Radiological Environmental Monitoring Program (DELETED)

h. Containment Leakage Rate Testing Program

Performance of the spring 2002 containment integrated leakage rate (Type A) test may be deferred up to 5 years but no later than the spring 2007.

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. Visual examination and testing, including test intervals and extensions, shall be in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 with exceptions provided in the site implementing instructions.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 12.0 psig.

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

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criteria 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 combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
SEQUOYAH PLANT (SQN)
UNITS 1 AND 2

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE
REVISED PAGES

I. AFFECTED PAGE LIST

Unit 2

Page 6-9

II. REVISED PAGES

See attached.

6.8.4 f. Radioactive Effluent Controls Program (Cont.)

of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the SITE BOUNDARY SHALL BE LIMITED to the following:

1. For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the total body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and
2. For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/year to any organ.

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radio-nuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50, and

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

g. Radiological Environmental Monitoring Program (DELETED)

h. 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. Visual examination and testing, including test intervals and extensions, shall be in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 with exceptions provided in the site implementing instructions. Performance of the spring 2002 containment integrated leakage rate (Type A) test may be deferred up to 5 years but no later than the spring 2007.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 12.0 psig.

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

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criteria 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 combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY
SEQUOYAH PLANT (SQN)
UNIT 2

TVA EVALUATION OF RISK SIGNIFICANCE FOR ONE-TIME EXTENSION OF CONTAINMENT INTEGRATED LEAK RATE TEST INTERVAL

QA Record

ORIGINAL

TVAN CALCULATION COVERSHEET/CCRIS UPDATE

REV 0 EDMS/RIMS NO.	EDMS TYPE:	EDMS ACCESSION NO (N/A for REV. 0)
B 87 010829 001	calculations(nuclear)	

Calc Title: EVALUATION OF THE RISK SIGNIFICANCE OF DECREASED CONTAINMENT INTEGRATED LEAK RATE TEST FREQUENCY

CALC ID	TYPE	PLANT	BRANCH	NUMBER	CUR REV	NEW REV
CURRENT	CN	SQN	NTB	SQS20211	0	
NEW	CN					

REVISION APPLICABILITY: Entire calc ☒ Selected pages ☐

ACTION	NEW ☒ REVISION ☐	DELETE ☐ RENAME ☐	SUPERSEDE ☐ DUPLICATE ☐	CCRIS UPDATE ONLY ☐ (D. V. & Approval Signatures Not Required)	No CCRIS Changes ☐ (For calc revision, CCRIS been reviewed and no CCRIS changes required)

UNITS	SYSTEMS	UNIDS
1 & 2	088	N/A

DCN EDC N/A	APPLICABLE DESIGN DOCUMENT(S) N/A	CLASSIFICATION
N/A		E

QUALITY RELATED?	SAFETY RELATED? (If yes, QR = yes)	UNVERIFIED ASSUMPTION	SPECIAL REQUIREMENTS AND/OR LIMITING CONDITIONS?	DESIGN OUTPUT ATTACHMENT?	SAR/TS AFFECTED
Yes ☐ No ☒	Yes ☐ No ☒	Yes ☐ No ☒	Yes ☐ No ☒	Yes ☐ No ☒	Yes ☐ No ☒

PREPARER PHONE NO	PREPARING ORG (BRANCH)	DESIGN VERIFICATION METHOD
843-8318	M/NE	Design Review

PREPARER SIGNATURE	DATE	CHECKER SIGNATURE	DATE
Christopher Carey	8/22/01	Anne E. Silber	8/22/01
DESIGN VERIFIER SIGNATURE	**DATE**	**APPROVAL SIGNATURE**	**DATE**
Anne E. Silber	8/22/01	John F. Thomas for	8/22/01

STATEMENT OF PROBLEM/ABSTRACT

This calculation determines the effect on release frequency and population dose as a result of a decrease in the frequency of performing containment integrated leak rate testing (ILRT). The results of this evaluation are to be used for justifying a change to the requirements for ILRT as administratively controlled by Technical Specification 6.8.4(h).

The effect of a decrease in the frequency of performing an ILRT is that the probability of a pre-existing leak in the containment shell increases. This results in an increase in the frequency of both large and small (fission product) releases to the environment which correlates to an increase in population dose. Revision 1 of the PSA is used to determine the increase in the frequency of large and small releases for ILRT frequencies of between 3/10 years and 1/20 years. Using information from Level III PSA analyses, the increase in population dose due to the increase in large and small release frequency due to a pre-existing leak is determined.

The results of this calculation demonstrate that decreasing the ILRT frequency from 1/10 ry to 1/15 ry results in an increase in:

- LERF of < 1.0E-7/ry,
- all releases (small, large, early and late) of about 3.49E-7/ry and
- the entire region population dose of 7.72 person-rem.

The increase in LERF is *very small* per RG 1.174 and the increase in all releases and the population dose are also *very small and non-risk-significant.*

MICROFICHE/EFICHE Yes ☐ No ☒ FICHE NUMBER(S)

☐ LOAD INTO EDMS AND DESTROY

☒ LOAD INTO EDMS AND RETURN CALCULATION TO CALCULATION LIBRARY. ADDRESS: OPS 1B-SQN

☐ LOAD INTO EDMS AND RETURN CALCULATION TO:

TVAN CALCULATION COVERSHEET/CCRIS UPDATE

CALC ID	TYPE	PLANT	BRANCH	NUMBER	REV
	CN	SQN	NTB	SQS20211	0

BLDG	ROOM	ELEV	COORD/AZIM	FIRM	Print Report Yes ☐
00	N/A	N/A	N/A	TVA	

CATEGORIES N/A

KEY NOUNS (A-add, D-delete)

ACTION (A/D)	KEY NOUN	A/D	KEY NOUN
A	PSA		
A	ILRT		
A	CONTAINMENT		
A	DOSE		

CROSS-REFERENCES (A-add, C-change, D-delete)

ACTION (A/C/D)	XREF CODE	XREF TYPE	XREF PLANT	XREF BRANCH	XREF NUMBER	XREF REV

CCRIS ONLY UPDATES:
Following are required only when making keyword/cross reference CCRIS updates and page 1 of form NEDP-2-1 is not included:

N/A	N/A	N/A	N/A
PREPARER SIGNATURE	DATE	CHECKER SIGNATURE	DATE

PREPARER PHONE NO. N/A	EDMS ACCESSION NO.

TVAN CALCULATION RECORD OF REVISION	
CALCULATION IDENTIFIER: SQS20211	
Title	EVALUATION OF THE RISK SIGNIFICANCE OF DECREASED CONTAINMENT INTEGRATED LEAK RATE TEST FREQUENCY
Revision No.	DESCRIPTION OF REVISION
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TVAN COMPUTER INPUT FILE
STORAGE INFORMATION SHEET

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Subject:
EVALUATION OF THE RISK SIGNIFICANCE OF DECREASED CONTAINMENT INTEGRATED LEAK RATE TEST FREQUENCY

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1.0 Purpose:

The purpose of this calculation is to determine the risk significance of a decrease in ILRT frequency. The effect of a decrease in the frequency of performing an ILRT is that the probability of a pre-existing leak in the containment shell increases. This results in an increase in the frequency of both large and small (fission product) releases to the environment which correlates to an increase in population dose. This calculation quantifies the increase in release frequency and population dose as a result of a decrease in the frequency of performing an ILRT.

2.0 References:

1. *Sequoyah Nuclear Plant Probabilistic Safety Assessment*, Revision 1 Report, (B38 960806800).

2. NUREG-1493, *Performance-Based Containment Leak-Test Program*, September, 1995.

3. Indian Point 3 Nuclear Power Plant, Docket No. 50-286, License No. DPR-64, *Supplemental Information Regarding Proposed Change to Section 6.14 of the Administrative Section of the Technical Specifications*, January 18, 2001.

4. NUREG/CR-4551, Volume 5, Revision 1, Part 1, *Evaluation of Severe Accident Risks: Sequoyah, Unit 1*, December, 1990.

5. Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis.*

3.0 Design Input Data: None.

4.0 Assumptions: None

5.0 Requirements/Limiting Conditions: None.

6.0 Computations and Analyses:

6.1 Effect of ILRT Frequency on the Probability of a Preexisting Containment Leak

The effect of a decrease in the frequency of performing an ILRT is that the probability of a preexisting leak in the containment shell increases. The fault tree for small and large containment isolation failures used in the PSA (from reference 1) accounts for the following failures to isolate containment:

- a failure of instrumentation to generate a containment isolation signal along with failure of the operator to manually initiate this action,
- a containment penetration failing to isolate as the result of the failure of the inboard and outboard isolation valves to close or
- the existence of a preexisting leak in the containment.

The first two containment isolation failures listed above are identified by ESFAS testing or stroke testing containment isolation valves, respectively. The existence of a leak in a containment penetration is identified by either a local leak rate test (LLRT) or an integrated leak rate test (ILRT). The existence of a leak in the containment shell is identified by an ILRT. The decrease in the frequency of conducting ILRTs increases the probability of a preexisting leak in containment, but does not affect the probability of the other containment isolation failure mechanisms listed above.

For a component that does not change state, the failure probability of the component (Q) is given by:

$$Q = \lambda * (T/2 + TM)$$

where,

λ = the failure rate
T = the test interval and
TM = the PSA mission time

Since T > year and TM ~ several days, the failure probability for a pre-existing containment leak is approximately:

$$Q = \lambda T/2$$

As discussed above, the existence of a leak in a containment penetration is identified by either a LLRT or an ILRT. The probability of a preexisting leak in a containment penetration can be rewritten as:

$$Q_p = \lambda_p\{P(LLRT)T_{LLRT} + P(ILRT/\overline{LLRT})T_{ILRT}\}/2$$

where,

Q_p	=	the probability of a preexisting leak in a containment penetration,
λ_p	=	the rate of occurrence of a containment penetration leak,
P(LLRT)	=	the probability of detecting a pre-existing containment penetration leak with a LLRT,
T_{LLRT}	=	the test interval for the LLRT,
$P(ILRT/\overline{LLRT})$	=	the probability of detecting a pre-existing leak with an ILRT given it was not detected with a LLRT and
T_{ILRT}	=	the test interval for the ILRT

As described in reference 2, LLRTs are performed prior to the ILRT so detecting a preexisting containment penetration leak during an ILRT is contingent upon not detecting it during a LLRT. Since all preexisting containment penetration leaks are detected by either a LLRT or a subsequent ILRT it follows that:

$$P(LLRT) + P(ILRT/\overline{LLRT}) = 1.0$$

Reference 2 determined that: $P(ILRT/\overline{LLRT}) = 0.03$; and therefore, $P(LLRT) = 0.97$ so the probability of a preexisting containment penetration leak is given by:

$$Q_p = \lambda_p\{0.97T_{LLRT} + 0.03T_{ILRT}\}/2$$

As discussed above, the existence of a leak in the containment liner is only identified by an ILRT. The probability of a preexisting leak in the containment liner can be written as:

$$Q_l = \lambda_l T_{ILRT}/2$$

where,

Q_l	=	the probability of a preexisting leak in the containment liner and
λ_l	=	the rate of occurrence of a containment liner leak

Therefore, the probability of a preexisting containment leak, Q, is equal to:

$$Q = \lambda_p\{0.97T_{LLRT} + 0.03T_{ILRT}\}/2 + \lambda_l T_{ILRT}/2$$

The remaining parameters in the above equation are the λ's -the failure rates. The failure rate of the containment liner is expected to be comparable to the failure rate of a storage tank rupture. The storage tank rupture failure rate distribution used in the PSA (reference 1) has a mean failure rate of 2.52E-8/hr (ZTTK1B) or 1.84E-05/month. The mean value of the failure rate as opposed to the 95th percentile (9.40E-8/hr) is used since the containment vessel is designed and tested as a pressure retaining membrane and due to the lack of any corrosion mechanisms. These factors tend to reduce

the expected failure rate for the containment vessel so the mean value for the failure rate of storage tank ruptures represents the 95th percentile failure rate for the containment vessel.

Previous analyses (references 3) have used values of Q of 0.064 for preexisting small containment leaks and 0.021 for preexisting large containment leaks. These failure probabilities are based on the 95th percentile of the χ^2 distribution using 0 large containment leaks and 4 small containment leaks in 144 tests. These failure probabilities are based on data previous to 1995 so they correspond to a 3-in-10 year ILRT test frequency and all containment penetrations being subjected to a LLRT once per refueling cycle. Using a LLRT interval of 18 months and a ILRT test interval of 40 months (3-in-10 years), the values λ_p are calculated from the below equations:

small: $0.064 = \lambda_p\{0.97T_{LLRT} + 0.03T_{ILRT}\}/2 + (1 - 0.021/0.064)(1.84E\text{-}05)T_{ILRT}/2 \Rightarrow \lambda_p = 6.83E\text{-}03$

large: $0.021 = \lambda_p\{0.97T_{LLRT} + 0.03T_{ILRT}\}/2 + (0.021/0.064)(1.84E\text{-}05)T_{ILRT}/2 \Rightarrow \lambda_p = 2.24E\text{-}03$

The increase in the probability of a preexisting small and large containment leak is given in Table-1 as the ILRT test interval is varied from 40 months to 20 years:

Table-1

ILRT Test Interval (years)	ILRT Test Interval (months)	Relative Probability of a Preexisting small leak (basis 0.064)	Relative Probability of a Preexisting large leak (basis 0.021)
3-1/3	40	1.00	1.00
10	120	1.14	1.14
15	180	1.24	1.24
20	240	1.34	1.35

6.2 Effect of a Preexisting Containment Leak on Releases:

The level I portion of the PSA (reference 1) determines the frequency of accident scenarios or sequences which result in damage to the core. In addition, the level I portion of the PSA determines the state or condition of the plant for the sequences which result in core damage (CD). Key information about the state of the plant determined for each CD sequence is:

- RCS pressure,
- the availability of secondary heat removal,
- if the RWST has been injected into containment,
- the availability of containment sprays and
- if the containment is isolated or bypassed.

The above described key information results in various combinations of plant conditions (referred to as plant damage states, PDS). Every CD sequence which has a frequency greater than a selected frequency is assigned to a PDS. The sum of the frequencies of all CD sequences assigned to a given

PDS yields the frequency of the PDS. The level I portion of the PSA reports the frequency of PDS which have a frequency greater than 1.0E-11.

The key information from the PDS for determining the effect of the increased probability of a containment leak is the state of containment. The PDS characterize the containment as being either intact, having a small or large isolation failure (hole) or as being bypassed by a small or large leak (SGTR or ISLOCA). This information is used to characterize the fission product release from containment and is presented in Table-2.

The PDS can be thought of as initiating events to the level II portion of the PSA. Rather than analyzing all 79 PDS in the level II portion of the PSA (see Table-2), the PDS are combined into 17 Key Plant Damage States (KPDS) as summarized in Table 3.

Table-2

PDS	Frequency	Intact[1]		small bypass[2]		large bypass[3]		small isolation failures[4]		large isolation failures[5]	
		analyzed in level II[6]	not analyzed in level II[7]	analyzed in level II[6]	not analyzed in level II[7]	analyzed in level II[6]	not analyzed in level II[7]	analyzed in level II[6]	not analyzed in level II[7]	analyzed in level II[5]	not analyzed in level II[7]
FCI	1.08E-05	1.08E-05									
ENI	9.71E-06	9.71E-06									
LCI	6.58E-06	6.58E-06									
EIB	2.95E-06			2.95E-06							
BCI	2.60E-06	2.60E-06									
FNI	2.20E-06	2.20E-06									
FGI	1.35E-06	1.35E-06									
ENB	8.25E-07			8.25E-07							
LNI	6.79E-07	6.79E-07									
GNI	4.88E-07	4.88E-07									
DCI	4.62E-07	4.62E-07									
HCI	4.18E-07	4.18E-07									
ENS	1.43E-07							1.43E-07			
LGI	1.14E-07		1.14E-07								
BGI	1.11E-07		1.11E-07								
KNI	9.46E-08		9.46E-08								
AGI	5.38E-08		5.38E-08								
EGI	4.73E-08	4.73E-08									
FCS	4.26E-08							4.26E-08			
HNI	4.18E-08	4.18E-08									
HGI	4.00E-08		4.00E-08								
ATV	3.51E-08					3.51E-08					
KNS	2.74E-08								2.74E-08		
LCS	2.60E-08								2.60E-08		
GNS	2.51E-08							2.51E-08			
ETL	2.33E-08									2.33E-08	

Table-2

PDS	Frequency	Intact[1]		small bypass[2]		large bypass[3]		small isolation failures[4]		large isolation failures[5]	
		analyzed in level II[6]	not analyzed in level II[7]	analyzed in level II[6]	not analyzed in level II[7]	analyzed in level II[6]	not analyzed in level II[7]	analyzed in level II[6]	not analyzed in level II[7]	analyzed in level II[6]	not analyzed in level II[7]
FPL	1.49E-08									1.49E-08	
BCS	1.04E-08								1.04E-08		
FCB	9.35E-09			9.35E-09							
LPL	8.95E-09										8.95E-09
FNS	8.59E-09							8.59E-09			
CNI	8.48E-09		8.48E-09								
BEI	7.44E-09		7.44E-09								
BNI	5.41E-09		5.41E-09								
CNS	5.09E-09								5.09E-09		
KGI	4.77E-09		4.77E-09								
FGS	4.57E-09								4.57E-09		
BPL	3.59E-09										3.59E-09
KTL	3.42E-09										3.42E-09
GTL	3.13E-09									3.13E-09	
LNS	2.55E-09								2.55E-09		
FTL	2.45E-09										2.45E-09
BII	2.26E-09		2.26E-09								
ANS	2.20E-09								2.20E-09		
DGI	1.86E-09		1.86E-09								
DCS	1.78E-09								1.78E-09		
FEI	1.66E-09		1.66E-09								
FRL	1.38E-09										1.38E-09
HCS	1.30E-09								1.30E-09		
DNI	1.19E-09		1.19E-09								
HEI	1.16E-09		1.16E-09								
LEI	8.67E-10		8.67E-10								
LTL	8.34E-10										8.34E-10

Table-2

PDS	Frequency	Intact[1]		small bypass[2]		large bypass[3]		small isolation failures[4]		large isolation failures[5]	
		analyzed in level II[6]	not analyzed in level II[7]	analyzed in level II[6]	not analyzed in level II[7]	analyzed in level II[6]	not analyzed in level II[7]	analyzed in level II[6]	not analyzed in level II[7]	analyzed in level II[6]	not analyzed in level II[7]
CTL	6.38E-10										6.38E-10
HCB	6.01E-10				6.01E-10						
DPL	5.99E-10										5.99E-10
HPL	4.01E-10										4.01E-10
FII	3.92E-10		3.92E-10								
ANI	3.88E-10		3.88E-10								
BGS	3.70E-10								3.70E-10		
GGI	3.60E-10		3.60E-10								
HII	3.40E-10		3.40E-10								
ATL	2.78E-10										2.78E-10
LGS	2.76E-10								2.76E-10		
LII	2.52E-10		2.52E-10								
JCI	2.14E-10		2.14E-10								
AGS	1.99E-10								1.99E-10		
EGS	1.47E-10								1.47E-10		
FNB	1.36E-10				1.36E-10						
BRL	1.25E-10										1.25E-10
GNB	1.18E-10				1.18E-10						
LRL	8.87E-11										8.87E-11
HGS	8.64E-11								8.64E-11		
ARL	7.22E-11										7.22E-11
HNS	7.00E-11								7.00E-11		
ERL	5.34E-11										5.34E-11
CGI	2.84E-11		2.84E-11								
BES	2.19E-11								2.19E-11		
FGB	1.78E-11				1.78E-11						

Table-2

PDS	Frequency	Intact[1]		small bypass[2]		large bypass[3]		small isolation failures[4]		large isolation failures[5]	
		analyzed in level II[6]	not analyzed in level II[7]	analyzed in level II[6]	not analyzed in level II[7]	analyzed in level II[6]	not analyzed in level II[7]	analyzed in level II[6]	not analyzed in level II[7]	analyzed in level II[6]	not analyzed in level II[7]
Totals		3.54E-05	4.51E-07	3.79E-06	8.72E-10	3.51E-08	0.00E+00	2.19E-07	8.25E-08	4.14E-08	2.29E-08
CDF	4.00E-05	Total Intact[8]	3.59E-05	Total small bypass[8]	3.79E-06	Total large bypass	3.51E-08	Total small Isolation Failure[8]	3.02E-07	Total Large Isolation Failure[8]	6.42E-08
								small dependent isolation failures[9]	1.50E-07	large dependent isolation failures[10]	6.57E-09
								small preexisting leaks[11]	2.56E-06	large preexisting leaks[11]	8.41E-07

Notes:

1. PDS which end in I
2. PDS which end in B
3. PDS which end in V
4. PDS which end in S
5. PDS which end in L
6. These are the PDS which are evaluated in the level II portion of the PSA (see Table 4.6-1 of reference 1).
7. These PDS are not evaluated in the level II portion of the PSA, but are used in this analyses for characterizing releases from containment.
8. Sum of the analyzed and unanalyzed PDS

Table-2

9. These are the accident sequences where there is a failure to isolate a small containment penetration. The sum of the PDS for the column titled *small isolation failures*, include accident sequences for which a small isolation failure has occurred due to either a failure to isolate a small containment penetration or a small preexisting leak. The probability of a small preexisting leak used in the PSA is 3.80E-03 (basic event CNTLK1_PREEXISTS). Therefore total frequency of small containment penetration isolation failures is calculated as: 3.02E-07 - 3.80E-03*CDF= 3.02E-07 - 3.80E-03*4.00E-05 = 1.50E-07/year.

10. These are the accident sequences where there is a failure to isolate a large containment penetration. The sum of the PDS for the column titled *large isolation failures*, include accident sequences for which a large isolation failure has occurred due to either a failure to isolate a large containment penetration or a large preexisting leak. The frequency of a large preexisting leak used in the PSA is 1.44E-03 (basic event CNTLK1_PREEXISTL). Therefore, total frequency of large containment penetration isolation failures is calculated as: 6.42E-08 - 1.44E-03*CDF = 6.42E-08 - 1.44E-03*4.00E-05 = 6.57E-09/year.

11. Since the PSA used a smaller value for the frequency of small preexisting containment leaks than determined in Section 6.1, the frequency for small preexisting containment leaks is calculated as the product of the probability of a small preexisting containment leak and CDF. From Section 6.1, the probability of a small preexisting containment leak for the 3 in 10 year ILRT test interval is 0.064. Therefore, the total frequency for small preexisting containment leaks is calculated as: 0.064*4.00E-05 = 2.56E-06/year.

12. Since the PSA used a smaller value for the frequency of large preexisting containment leaks than determined in Section 6.1, the frequency of a large preexisting containment leaks is calculated as the product of the probability of a large preexisting containment leak and CDF. From Section 6.1, the probability of a large preexisting containment leak for the 3 in 10 year ILRT test interval is 0.021. Therefore, the total frequency for large preexisting containment leaks is calculated as: 0.021*4.00E-05 = 8.41E-07/year.

Table-3

KPDS	Frequency	Description
FCI	1.08E-05	
EIB	3.05E-06	Includes FCS, FCB, ETL, GTL and FPL
ENIYA	2.91E-06	0.3*ENI (see Table 4.6-3 of reference 1)
ENIYB	5.83E-07	0.06*ENI (see Table 4.6-3 of reference 1)
ENIYN	6.21E-06	0.64*ENI (see Table 4.6-3 of reference 1)
FNI	2.20E-06	
BCI	2.60E-06	
ENB	1.00E-06	Includes GNS, ENS, FNS
FGI	1.35E-06	
LCI	7.04E-06	Include DCI
GNI	4.88E-07	
HCI	4.18E-07	
ATV	3.51E-08	
HNI	4.18E-08	
EGI	4.73E-08	
LNIYA	3.53E-07	0.52*LNI (see Table 4.6-3 of reference 1)
LNIYC	3.26E-07	0.48*LNI (see Table 4.6-3 of reference 1)

In addition to the previously discussed causes of containment isolation failure, there are additional containment failures that result from the progression of the accident. These failures are identified in the level II portion of the PSA. The level II portion of the PSA (reference 1) determines the frequency of accident scenarios or sequences which result in containment failures. In addition, the level II portion of the PSA determines the plant/conditions in containment for the sequences which result in containment failure (CF). Key information about the state of the plant determined for each CF sequence is:

- RCS pressure at the time of vessel failure,
- time and size and location of the containment failure or bypass,
- containment spray operation and ice condenser function and
- ex-vessel debris cooling.

The above described key information results in various source term characteristics (referred to as key release categories, KRC). Every CF sequence which has a frequency greater than a selected frequency is assigned to a KRC. The sum of the frequencies of all CF sequences assigned to a given KRC yields the frequency of the release category. The level II portion of the PSA reports the frequency of KRC which have a frequency greater than 1.0E-11.

The KRC characterize the releases from containment as being either early or late and as being either small or large. The KRC for small and large early releases are due to either the containment isolation failures, preexisting leaks or bypasses previously identified in the level I portion of the PSA or due to severe accident progression (e.g. a large containment failure due to a hydrogen explosion). The KRC for late releases (either small or large) are due solely to severe accident progression. The frequency

of the KRC and their characterization are presented in Table 4. The characterization of the KRC is consistent with Table 4.9-3 of reference 1.

Table-4

KRC	Frequency	Intact	Small Early CF & Bypass	Large Early CF & Bypass	Late[1]
R21	1.43E-05	1.43E-05			
R22	6.99E-06	6.99E-06			
R17L	4.50E-06				4.50E-06
R20	4.05E-06		4.05E-06[4]		
R17U	1.87E-06				1.87E-06
R11I	6.16E-06				6.16E-06
R17LU	4.44E-07				4.44E-07
R01DI	2.42E-07			2.42E-07	
R04IF	1.15E-07			1.15E-07	
R03IF	5.88E-08			5.88E-08	
R19	3.51E-08			3.51E-08	
R11IF	5.78E-07				5.78E-07
R01IF	2.85E-08			2.85E-08	
R02IF	2.63E-08			2.63E-08	
R03I	5.93E-08			5.93E-08	
R04UIF	1.55E-08			1.55E-08	
R03	8.89E-09			8.89E-09	
R01I	8.83E-09			8.83E-09	
R18	9.75E-09			9.75E-09	
R04	3.34E-09			3.34E-09	
R03UIF	3.08E-09			3.08E-09	
R01UIF	7.68E-10			7.68E-10	
R05LIF	5.80E-10		5.80E-10		
R06IF	5.23E-10		5.23E-10		
R05LI	2.27E-10		2.27E-10		
R05IF	6.29E-11		6.29E-11		
R05I	1.88E-11		1.88E-11		
R06LIF	2.62E-11		2.62E-11		
Total	3.95E-05	2.13E-05	4.05E-06	6.15E-07	1.36E-05
Small early releases due to severe accident progression[2]					4.27E-08
Large early releases due to severe accident progression[3]					5.39E-07

Table-4

Notes:

1. Large and small source term release is not distinguished.
2. This is calculated as the total of the column *Small Early CF & Bypass* minus the total from the columns *small bypass - analyzed in level II* and *small isolation failures - analyzed in level II* from Table-2.
3. This is calculated as the total of the column *Large Early CF & Bypass* minus the total from the columns *Large bypass - analyzed in level II* and *Large isolation failures - analyzed in level II* from Table-2.
4. This is the total frequency of SGTR releases which is slightly higher than listed in Table-2 since this includes severe accident induced SGTR failures.

Using the results from Tables-1, -2 & -4 the effect of increasing the ILRT test frequency are summarized in Table-5.

Table-5

Class	Description	ILRT Frequency (ILRT/month)			
		1/40	1/120	1/180	1/240
1	Containment Intact[2]	1.84E-05	1.80E-05[1]	1.76E-05	1.73E-05
2	Small Containment Penetration Isolation Failures[3]	1.50E-07			
3	Large Containment Penetration Isolation Failures[3]	6.57E-09			
4	Small Early Containment Failures Due Severe Accident Progression[3]	4.27E-08			
5	Large Early Containment Failures Due to Severe Accident Progression[3]	5.39E-07			
6	Late Containment Failures (Small & Large) Due to Severe Accident Progression[3]	1.36E-05			
7	Small Containment Bypasses[3]	4.05E-06			
8	Large Containment Bypasses[3]	3.51E-08			
9	Small Preexisting Leaks[4]	2.56E-06	2.91E-06	3.17E-06	3.43E-06
10	Large Preexisting Leaks[4]	8.41E-07	9.58E-07	1.05E-06	1.13E-06
	CDF (sum of all classes)	4.02E-05	4.02E-05	4.02E-05	4.02E-05
	LERF (sum of classes 3, 5, 7, 8 & 10)	5.47E-06	5.59E-06	5.68E-06	5.76E-06
	Change in LERF (based on a 1/120 month ILRT frequency)	-1.17E-07	0.00E+00	8.79E-08	1.76E-07
	All Releases (sum of classes 2 through 10)	2.18E-05	2.22E-05	2.26E-05	2.29E-05
	Change in All Releases (based on a 1/120 month ILRT frequency)	-4.65E-07	0.00E+00	3.49E-07	6.98E-07
	Change in All Releases (% based on a 1/120 month ILRT frequency)	-2.1	0.0	1.6	3.1

Table 5

Notes:

1. Base value calculated as the sum of *intact frequency* from Table-2 and the *intact -not analyzed in level-2*, the total *small isolation failures -analyzed in level II*, the total *large isolation failures - analyzed in level II* from Table-2. Minus the sum of the *small dependent isolation failures*, the *large dependent isolation failures*, the *small preexisting leaks* and the *large preexisting leaks* from Table-2.

2. Calculated as the base value (1.84E-5) plus class 9 & 10 for the ILRT/40 month column minus class 9 & 10 from the column of interest.

3. Invariant to changes in ILRT frequency.

4. See Table-1 for the multiplier used on the frequency of these leaks given in Table-2 (e.g., for a ILRT/120 months, the multiplier for small preexisting leak frequency is 1.14 and the base frequency is 2.56E-06).

6.3 Effect of a Preexisting Containment Leak on Population Dose

The release classes determined in Section 6.2 are assigned a leakage rate in Table-6, consistent with reference 3.

Table-6

Class	Description	Maximum Leak Rate (in L_a)[1]
1	Containment Intact	2
2	Small Containment Penetration Isolation Failures	35
3	Large Containment Penetration Isolation Failures	35
4	Small Early Containment Failures Due Severe Accident Progression	100
5	Large Early Containment Failures Due to Severe Accident Progression	100
6	Late Containment Failures (Small & Large) Due to Severe Accident Progression	100
7	Small Containment Bypasses[2]	N/A
8	Large Containment Bypasses[2]	N/A
9	Small Preexisting Leaks	10
10	Large Preexisting Leaks	35

Notes:

1. L_a is 0.25%/day

2. These sequences involve containment bypasses so their leak rate is not quantified in terms of L_a. These sequences are not effected by changes in ILRT frequency.

The dose to the surrounding population from severe accidents was determined for SQN in reference 4. The results of that study are summarized in Table-7.

Table-7

Risk Measure[1]	Dose (person-rem/ry)
Population Dose 50 miles	12
Population Dose Entire Region[2]	81[(3)]

Notes:
1. From Table 5.1-1 of reference 4.
2. The entire region is the area within 1000 miles of SQN (Section 4.2 of reference 4).
3. This is the same value used in reference 2.
4. The reference 4 study determined the CDF for SQN to be 5.60E-5/ry.

The population dose for the entire region is used in reference 2 as the dose from a leak rate of 1 L_a. Consistent with reference 3, the population dose is increased linearly with L_a to determine the population dose for a given class of containment releases. The effect on population dose as ILRT frequency is decreased is calculated in Table-8.

Table-8

Class	Description	Population Dose at an ILRT Frequency (1/month) of:[1]			
		1/40	1/120	1/180	1/240
1	Containment Intact	5.33E+01	5.20E+01	5.10E+01	5.00E+01
2	Small Containment Penetration Isolation Failures	7.58E+00			
3	Large Containment Penetration Isolation Failures	3.33E-01			
4	Small Early Containment Failures Due Severe Accident Progression	6.18E+00			
5	Large Early Containment Failures Due to Severe Accident Progression	7.79E+01			
6	Late Containment Failures (Small & Large) Due to Severe Accident Progression	1.96E+03			
7	Small Containment Bypasses	Not Quantified			
8	Large Containment Bypasses	Not Quantified			
9	Small Preexisting Leaks	3.71E+01	4.21E+01	4.59E+01	4.97E+01
10	Large Preexisting Leaks	4.26E+01	4.85E+01	5.30E+01	5.74E+01
	Total Dose (person-rem)[2]	2.19E+03	2.19E+03	2.20E+03	2.21E+03
	Change in Population Dose for Entire Region[3]	-9.62	0.00	7.22	14.43

Table-8

Notes:

1. This is calculated as the product of the frequency of the described sequences from Table-5, the magnitude of the release (in La) from Table-5 and dose to the entire region which is calculated as the dose to the entire region which is calculated based on the information in Table-7, specifically, 81 person-rem/ry divided by the CDF of 5.60E-05 or 1.45E6 person-rem.

2. Sum of classes 1 through 6, 9 & 10.

3. Change based on an ILRT/120 months.

7.0 Summary of Results:

1. LERF increases by 8.79E-8/ry when the frequency of an ILRT is decreased from 1/10 ry to 1/15 ry.

2. All releases (small, large, early and late) increases by 3.49E-7/ry or 1.6% when the frequency of an ILRT is decreased from 1/10 ry to 1/15 ry.

3. Population dose increases by 7.72 person-rem when the frequency of an ILRT is decreased from 1/10 ry to 1/15 ry.

8.0 Supporting Graphics: None.

9.0 Conclusions:

1. The increase in LERF when the frequency of an ILRT is decreased from 1/10 ry to 1/15 ry is less than 1.0E-7/ry which is considered *a very small increase in LERF* per Regulatory Guide 1.174 (reference 5).

2. The increase in the frequency of all releases (small, large, early and late) and the increase in population dose when the frequency of an ILRT is decreased from 1/10 ry to 1/15 ry are about 2% & <1%, respectively. Although no specific criteria is stated in RG 1.174, these increases are also *very small* and non-risk-significant.

10.0 Appendices and Attachments: None.