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

September 12, 2002

TVA-SQN-TS-00-06 10 CFR 50.90

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

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

SEQUOYAH NUCLEAR PLANT (SQN) - UNITS 1 AND 2 - TECHNICAL SPECIFICATION (TS) CHANGE NO. 00-06, REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) (TAC NOS. MB2972 AND MB2973)

References: 1. TVA letter to NRC dated September 21, 2001, "Sequoyah Nuclear Plant (SQN) - Units 1 and 2 - Revision of Instrumentation Measurement Range, Boron Concentration Limits, Reactor Core Limitations, and Spent Fuel Pool Storage Requirements for Tritium Production Cores (TPCs) - Technical Specification (TS) Change No. 00-06"

2. TVA letter to NRC dated August 30, 2002, "Sequoyah Nuclear Plant (SQN) - Units 1 and 2 - Technical Specification (TS) Change No. 00-06, Response to Request for Additional Information (RAI) (TAC Nos. MB2972 and MB2973)"

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3. TVA letter to NRC dated September 5, 2002, "Sequoyah Nuclear Plant (SQN) - Units 1 and 2- Technical Specification (TS) Change No. 00-06, Response to Request for Additional Information (RAI) (TAC Nos. MB2972 and MB2973)"

TVA submitted TS Change 00-06 to NRC by the Reference 1 letter to propose changes to the SQN TSs that will accommodate the production of tritium. TVA submitted the Reference 2 and 3 letters in response to requests for additional information associated with the SQN tritium TS change request. As a result of subsequent discussions with NRC, TVA agreed to submit additional information associated with Questions 1.4 and 2 of Reference 2. The response to Question 1.4 addresses the justification for the use of 51 cubic feet per minute for control room inleakage for the determination of control room dose consequences. The response for Question 2 addresses the applicability of General Design Criteria 19 to potential accident scenarios that have not previously been a part of the SQN licensing bases and the basis for current evaluations bounding potential accident scenarios.

NRC also provided comments on the revised response to Question 16 in Reference 3. These comments addressed a schedule for the resolution of the generic downcomer boiling issue for emergency core cooling system models at SQN. TVA's schedule is based on Framatome-Advance Nuclear Power's resolution of the generic industry issue on downcomer boiling and is addressed in the revised response to Question 16.

TVA's revised responses to the above-mentioned questions are provided in Enclosure 1. The one new commitment contained in this letter is summarized in Enclosure 2. The proposed TS change in the Reference 1 letter is not altered by the enclosed responses.

This letter is being sent in accordance with NRC RIS 2001-05. If you have any questions about this response, please telephone me at (423) 843-6672 or K. C. Weller at (423) 843-7527.

Sincerely,

James D. Smith
Acting Licensing and Industry Affairs Manager

Subscribed and sworn to before me
on this 12th day of September

Notary Public

My Commission Expires August 12, 2006

JDS:KCW:PMB
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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2
DOCKET NOS. 327 AND 328

REVISED RESPONSES TO
REQUEST FOR ADDITIONAL INFORMATION (RAI)
QUESTIONS 1.4 AND 2 (AUGUST 30, 2002)
AND
QUESTION 16 (SEPTEMBER 5, 2002)
TECHNICAL SPECIFICATION (TS) CHANGE 00-06

RAI Question 1.4 (August 30, 2002):

A justification for the assumed 51 cfm control room unfiltered inleakage given recent industry experience with integrated tracer gas testing of the control room envelope.

Revised Response:

The irradiation of tritium producing burnable absorber rods does not impact analysis assumptions regarding control room inleakage. The issue of main control room habitability and General Design Criteria (GDC) 19 compliance was addressed in Sections 1.5.5 and 2.15.6.3 of BAW-10237 of the Sequoyah License Amendment Request (LAR). The data in the LAR table demonstrates that operation with a Tritium Production Core (TPC) will result in compliance with GDC 19.

In accordance with GDC 19, the control room ventilation system and shielding have been designed to limit deep dose equivalent during an accident period to 5 rem. Thyroid dose is limited to 30 rem and beta skin dose should not exceed 30 rem. To appropriately account for the radiological consequences of the increased tritium in the TPC, TVA has included a calculated Total Effective Dose Equivalent (TEDE) in the accident analysis for information.

It should be noted that tritium is unaffected (unfiltered) by the particulate and iodine filtration provided in the ventilation system. Its concentration within the control room is, therefore, only minimally impacted by changes in control room unfiltered inleakage.

During the SQN original pre-operational testing, an unfiltered inleakage test was conducted on the two pressurized air ducts that pass through the control room envelope from other areas. These tests were conducted in August of 1980 in response to test deficiencies associated with unfiltered inleakage to the control room. The battery room exhaust duct was measured at 7.7 cubic feet per minute (cfm) and the normal control building pressurization fan duct at 21.3 cfm. TVA also evaluated the contribution from pneumatic instrumentation and valve operators and from door ingress and egress to the envelope. This contribution was determined to be 19 cfm for pneumatic components and 3 cfm from door activities. These leakage amounts total to 51 cfm and was used to determine acceptability. TVA revised calculations to include the 51 cfm total for the control room dose evaluations. The revised calculations indicated that the 51 cfm was acceptable, therefore TVA accepted the test results. Since 1980 SQN has made significant ducting modifications and has changed its amount of emergency pressurizing air from 200 cfm maximum to 1000 cfm maximum. Ductwork modifications included sealing ducts, removing ducting, and isolating ducts. TVA has also removed the functions of the normal control building pressurization fans such that the associated duct no longer has the potential to have a positive pressure with respect to the control room. This change alone removes 21.3 cfm of the original 51 cfm inleakage measurements. These plant modifications have had a significant effect in reducing the amount of unfiltered inleakage.

SQN performs surveillance testing of its control room and, as part of that testing, verifies that the control room habitability zone (which includes the control room) is at a positive pressure (minimum of 1/8-inch water gauge) to the outside and is at a positive pressure to all spaces adjacent to the zone. Presently there is only one positive pressure duct that passes through the zone from other areas (the battery room exhaust duct) and the other heat, ventilation, and air conditioning systems serving the zone are contained within the zone or are at a negative pressure with respect to the zone. The leakage from this duct was measured at 7.7 cfm in 1980. The duct passing through the zone is approximately 12 feet in length, bolted flange type construction, and all of the flange joints, bolts, and access panels are sealed with room temperature vulcanizing (RTV) 106. A recent walkdown of this ductwork verified the above described configuration for this duct and the sealed areas were in excellent condition. Therefore, the 7.7 cfm measured inleakage from this duct is still an acceptable value to use for inleakage considerations.

The fact that SQN verifies that all zone spaces are at a positive pressure to adjacent spaces, the normal control building pressurization fan have been abandoned, ductwork modifications were performed to reduce inleakage, and emergency pressurizing air flow has been increased, all support the assurance that inleakage has not increased since 1980. The expected issuance of a generic letter (GL) later this year, on this same subject, will prompt additional consideration of the inleakage issue industry-wide. SQN will provide additional information to address this matter along with all other affected licensees at that time in accordance with the GL requirements.

RAI Question 2 (August 30, 2002):

Section 1.5.5 discusses the issue of control room habitability. The discussion is limited to the emergency core cooling system (ECCS) leakage component of the LOCA, consistent with the interface item. However, reference is made in Section 1.5.5 of BAW-10237 to Table 2.15.6-2 as the basis of the TVA conclusion that General Design Criteria -19 (GDC-19) will continue to be met with tritium-producing burnable absorber rod (TPBAR) use. The staff notes that the language of GDC-19 is not restricted to LOCAs, but applies to all accidents. In its letter dated May 21, 2002 with regard to Watts Bar Nuclear Plant, TVA stated that accidents other than the LOCA were limiting with regard to control room doses. TVA is requested to explain whether and how its conclusion that GDC-19 will be met considers all of the design basis radiological accidents addressed in the SQN Updated Final Safety Analysis Report.

Revised Response:

Section 1.5 of BAW-10237 of the SQN LAR was developed to address each of the NRC Interface Items as identified in NUREG-1672, "Safety Evaluation Report Related to the Department of Energy's Topical Report on the TPC." Section 1.5.5 addresses Interface Item 5, on the control room habitability systems. NUREG-1672, Section 2.6.1 discussed the impact of ECCS leakage on main control room habitability and required a plant specific assessment of ECCS impacts. The discussion in Section 1.5.5 is limited to discussion of this subject and the associated accident (loss-of-coolant accident [LOCA]) and the impact on GDC-19 criteria for main control room habitability.

TVA concurs that GDC-19 applies to the spectrum of potential accident scenarios in the SQN Updated Final Safety Analysis Report (UFSAR). These scenarios, with exceptions of LOCA and main steam line break (MSLB), have not been previously analyzed against the GDC-19 criteria for main control room habitability. However, based on the results of the SQN plant specific LOCA evaluation and the similarly of SQN to the spectrum of analyses performed for Watts Bar Nuclear Plant (WBN), we believe that SQN plant specific evaluations for the balance of the UFSAR Chapter 15.5 accidents will confirm that the SQN control room design meets the requirements of GDC-19 for both current operating conditions as well as operation with the tritium production core.

The similarities between SQN and WBN are included in the following comparisons. Table 1 presents the SQN and WBN key parameters used in control room dose analyses. Table 2 presents the calculated LOCA control room doses without ECCS leakage. The SQN and WBN parameters are almost identical, except for the X/Q values. Examination of the doses shows that the two plants are very similar in accident consequences. Therefore, it is reasoned that SQN control room doses for other accidents will be similar to the WBN values. WBN fully conforms to GDC-19 requirements, therefore, SQN should also.

Table 1
Comparison of Key WBN/SQN Control Room Dose Parameters

	SQN	WBN	**SQN/WBN ratio**
Control Room Free Volume [cuft]	257198	257198	**1.00**
CR Makeup flow [cfm]	711	1000	**0.71**
CR Unfiltered inflow [cfm]	51	51	**1.00**
CR recirculation flow [cfm]	3600	3000	**1.20**
CREVS filters	HEPA/ 2" charcoal	HEPA/ 2" charcoal	**1.00**
wall/ceiling/floor concrete thickness [ft]	3/2.25/0.67	3/2.25/0.67	**1.00**
layout of external plant (release points relative to CR intakes)	same as WBN	same as SQN	**1.00**
ARCON96 CR (LOCA/FHA) X/Q [sec/cum] 0-2hr	5.63E-4	1.12E-3	**0.50**
2-8 hr	3.78E-4	9.78E-4	**0.39**
8-24 hr	1.12E-4	1.21E-4	**0.93**
1-4 day	9.38E-5	9.36E-5	**1.00**
4-30 day	6.96E-5	7.77E-5	**0.90**
ARCON96 CR (MSLB/SGTR) X/Q [sec/cum] 0-2hr	1.93E-3	8.15E-4	**2.37**
2-8 hr	7.02E-4	4.75E-4	**1.48**
8-24 hr	2.85E-4	2.27E-4	**1.26**
1-4 day	2.13E-4	1.81E-4	**1.18**
4-30 day	1.61E-4	1.45E-4	**1.11**
core MWt	3480	3480	**1.00**
CR isolation signal	Safety Injection or high radiation at CR intake	Safety Injection or high radiation at CR intake	**1.00**
control room intake monitor	RD-32-01	RD-32-01	**1.00**
ABGTS exhaust rate (release for FHA) [cfm]	7000	7000	**1.00**
ABGTS filters	HEPA/4" charcoal	HEPA/4" charcoal	**1.00**
end of life core inventory [Ci] I-131	9.027E+07	9.01E+07	**1.002**
I-132	1.314E+08	1.31E+08	**1.003**
I-133	1.908E+08	1.88E+08	**1.015**
I-134	2.109E+08	2.08E+08	**1.014**
I-135	1.779E+08	1.76E+08	**1.011**

Table 2
ARCON96 X/Q Control Room LOCA Doses
with TPC and without ECCS Leakage (rem)

	SQN	WBN	**SQN/WBN ratio**
Gamma	1.1375	0.796	**1.43**
Beta	5.655	6.769	**0.84**
Inhalation (ICRP-30)	3.637	2.076	**1.75**
TEDE	2.1859	1.9083	**1.15**

TVA will conduct the additional plant specific analyses and provide the results including a tabulation of analysis inputs and assumptions used in the control room habitability analyses by December 2002.

In addition to addressing the NRC regulatory requirements and dose guidelines, TVA will include calculated TEDE in the control room accident analysis to appropriately account for the radiological consequences of the increased tritium in the TPC. The TEDE values will be calculated for informational purposes only and do not replace the whole body and thyroid dose guidelines currently in the SQN licensing basis.

Table 3 contains the control room GDC-19 MSLB (current steam generators with alternate repair criteria) analysis results. These calculated radiological consequences are well within the NRC regulatory guidelines of GDC-19.

TABLE 3
SQN Control Room Dose
Radiological Consequences of a Main Steam Line Break Accident

Control Room	**SQN Operation with Pre-existing Iodine Spike**	**SQN Operation with Accident-Initiated Iodine Spike**	**Acceptance Limit**
Thyroid dose (ICRP-30)	**1.6 rem**	**2.1 rem**	**30 rem**
Whole body dose (γ) **TEDE**	**0.192 rem** **0.373 rem**	**0.52 rem** **0.72 rem**	**5 rem**
Beta-skin	**0.067 rem**	**0.084 rem**	**30 rem**

Table 4 contains the control room GDC-19 LOCA analysis results previously provided in the Watts Bar and SQN LARs. The similarity of these data, and the fact that all calculated radiological consequences were well within NRC regulatory requirements, provide a high level of assurance that TVA meets the requirements of GDC-19 for all postulated events.

TABLE 4
SQN and WBN Table 2.15.6-2 Control Room Dose Comparison
Radiological Consequences of a Design Basis LOCA (rem)

Control Room	SQN Operation with 2,256 TPBARs	WBN Operation with 2,304 TPBARs	Acceptance Limit
Thyroid dose (ICRP-30)			
– Containment leakage	3.637	2.076	
– Recirculation leakage	6.668E-02	3.082E-02	
Total	3.704	2.106	30
Whole body dose (γ)			
– Containment leakage	1.138	7.960E-01	
– Recirculation leakage	4.728E-03	1.263E-03	
Total	1.142	7.973E-01	5
TEDE	2.213	1.911	
Beta-skin			
– Containment leakage	5.655	6.769	
– Recirculation leakage	9.77E-01	1.404E-02	
Total	6.642	6.783	30

RAI Question 16 (September 5, 2002):

In Section 2.15.5.1 of the SQN Topical Report the licensee states that the boundary conditions (fuel rod temperatures and fluid conditions) for the TPBAR temperature calculations are taken from the Appendix K LOCA analyses of record. Modeling of the downcomer region and downcomer boiling have recently been shown to substantially impact peak clad temperature (PCT) and oxidation following a loss-of-coolant accident (LOCA), especially for ice condenser containments. Please discuss how the downcomer region and downcomer boiling are modeled in the SQN LOCA Appendix K evaluation model, and dicuss any potential adverse impacts this modeling may have on PCT, oxidation, and TPBAR temperatures and oxidation.

Revised Response:

Framatome-Advance Nuclear Power, Inc. (FRA-ANP) used an NRC-approved Appendix K evaluation model, which is comprised of the RELAP5/MOD2-B&W, REFLOD3B, and BEACH computer models, to establish the licensing basis for fuel operational limits at SQN. The "worst" case LOCA transient, in terms of peak cladding temperature, was used to establish boundary conditions for predicting the post-LOCA thermal response of Tritium-Producing Burnable Absorber Rods (TPBAR). The TPBARs were shown to respond acceptably in the LOCA environment.

The REFLOD3B reflooding model is not as fine as the RELAP5/MOD2-B&W blowdown model; for example, the downcomer and lower reactor vessel head are simulated by a single node. The REFLOD3B model, however, includes the heat content of all of the vessel structures and the wall heat transfer coefficients in the prediction of vessel fluid conditions. The reflooding calculation showed no indication of reflood fluid saturation or the presence of downcomer boiling. All of the SQN Appendix K LOCA calculations were carried out until the core was completely quenched, which occurred at about 800 seconds.

Although the limited noding in the REFLOD3B model is a simplification that does not model all local fluid conditions, the Appendix K model contains conservatisms that more than compensate for any local phenomena that are not explicitly simulated, such as the behavior of the downcomer fluid. Appendix K requires that highly conservative assumptions and unrealistic modeling be used in the analysis of LOCA events. These requirements were included to compensate for LOCA phenomena that were either incompletely understood or unrecognized at the time. The following Appendix K requirements provide a substantial degree of conservatism:

Section I.A.

> "Fission Product Decay. The heat generation rates from radioactive decay of fission products shall be assumed to be equal to 1.2 times the values for infinite operating time in the American Nuclear Society (ANS) Standard (Proposed American Nuclear Society Standards - 'Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors.' Approved by Subcommittee ANS - 5, ANS Standards Committee, October 1971)."

> This decay heat requirement, originally intended to adjust LOCA calculations for variations in core design and operational parameters, has been demonstrated to be extremely conservative by experiments and decay heat models.

Section I.C.

> ". . . For postulated cold leg breaks, all emergency cooling water injected into the inlet lines or the reactor vessel during the bypass period shall in the calculations be subtracted from the reactor vessel calculated inventory."

> This requirement causes the analyst to discard fluid entering the reactor vessel and neglects the actual fluid

penetration into the downcomer, an occurrence verified by experiment.

Section I.C.

"After critical heat flux is first predicted at an axial fuel rod location during blowdown, the calculation shall not use nucleate boiling heat transfer correlations at that location subsequently during the blowdown even if the calculated local fluid and surface conditions would apparently justify the reestablishment of nucleate boiling . . . . Transition boiling heat transfer shall not be reapplied for the remainder of the LOCA blowdown, even if the clad superheat returns below 300 F. . . ."

Film boiling lock-in during blowdown, particularly applied near the core entrance, artificially retards core quench by keeping the fuel and cladding surface temperature elevated.

Each of these conservatisms artificially reduces the rate of core reflood and elevates the calculated PCT, thus overshadowing the occurrence of any local phenomena that are not specifically modeled, such as downcomer boiling.

FRA-ANP has requested, and expects to receive within the next few months, NRC approval of its realistic LOCA methodology. This realistic model explicitly models downcomer fluid conditions and the effects of structural wall heat transfer in sufficient detail to predict the occurrence of downcomer boiling. Although the method has not been specifically applied to SQN, it has been used to analyze large break LOCAs for a Westinghouse-designed plant that has similar but more bounding containment pressure response. (Since low containment pressures during reflood enhance the possibility of downcomer boiling, this condition is an important factor governing downcomer behavior.) PCT predictions in this application are on the order of 1700 degrees Fahrenheit (°F).

The significantly lower PCTs predicted by realistic methods demonstrate that local phenomena, such as downcomer boiling, are dominated by Appendix K conservatisms, as intended by the rule. Specifically, the PCT predicted by FRA-ANP's deterministic model is more than 400°F higher than that obtained using the realistic model.

The inclusion of TPBARs in a reactor is similar to the effect of using burnable poison rods. Despite this reality, the thermal response of the TPBARs is calculated using boundary conditions and assumptions based on the highly-conservative results from an Appendix K model, which are intended for fuel rod thermal analysis. Because of the added conservatism of applying the results of an Appendix K model to a burnable poison, it is concluded that downcomer boiling is not pertinent to the calculation of the thermal response of TPBARs.

It is concluded that the conservatism included in the prediction of PCT using the FRA-ANP Appendix K method more than accounts for any effects that might result from downcomer boiling. The application of boundary conditions from the highly conservative Appendix K model to the calculation of the thermal response of the TPBARs confirms their safe operation in a post-LOCA environment. FRA-ANP is aware of the potential importance of the downcomer boiling in certain plant designs and has independently initiated discussions with the NRC on downcomer boiling in the context of another application not related to SQN. These discussions will define a process and schedule for resolving how this matter should be incorporated into future LOCA analyses. Upon reaching this resolution, an evaluation of whether downcomer boiling should be included in the calculation of the post-LOCA thermal response of TPBARs will be made. Any changes to the present SQN evaluation required to address downcomer boiling will be addressed by FRA-ANP and TVA in accordance with the requirements of 10CFR50.46(a)(3)(i). Any required reanalysis will be provided to NRC in accordance with 10CFR50.46(a)(3)(ii) within six months of NRC approval of any model changes required to resolve the issue.

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2
DOCKET NOS. 327 AND 328

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI)
TECHNICAL SPECIFICATION (TS) CHANGE 00-06

COMMITMENT LISTING

TVA will provide to NRC any required reanalysis of the Appendix K emergency core cooling system evaluation model in accordance with 10CFR50.46(a)(3)(ii) within six months of NRC approval of any model changes required to resolve the generic downcomer boiling issue.