TVA

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

March 11, 2002

TVA-SQN-TS-01-08 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. 01-08, RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) (TAC NOS. MB3435 AND MB3436)

References: 1. TVA letter to NRC dated November 15, 2001, "Sequoyah Nuclear Plant (SQN) - Technical Specification (TS) Change No. 01-08, 'Increase Maximum Allowed Reactor Power Level to 3455 Mega-Watt Thermal (MWt)'"

2. NRC letter to TVA dated February 7, 2002, "Sequoyah Nuclear Plant (SQN), Units 1 and 2 - Request for Additional Information on Technical Specification Change No. 01-08, 'Increase Maximum Allowed Reactor Power Level to 3455 Mega-Watt Thermal (MWt)' (TAC Nos. MB3435 and MB3436)"

TVA submitted TS Change 01-08 to NRC by the Referenced 1 letter to propose an increase in the maximum allowed reactor power level to 3455 MWt. This letter provides the responses to NRC questions contained in the Referenced 2 letter regarding proposed TS Change 01-08. The questions in the

D030

Referenced 2 letter were further clarified in a telephone conversation between NRC, Westinghouse Electric Corporation, and TVA personnel on February 26, 2002.

The enclosure to this letter provides responses to the NRC RAI in the Referenced 2 letter. There are no new commitments contained in this letter and the proposed TS change in the Referenced 1 letter is not altered by the enclosed responses. TVA requests NRC approval as soon as possible such that the leading edge flow meter (LEFM) installation on Unit 1 may be utilized to increase power output. In addition, to account for potential unforeseen problems during testing and final adjustments of the Unit 2 LEFM system following restart from the Unit 2 Cycle 11 refueling outage, an implementation duration of 120 days is requested.

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) 843-7170 or J. D. Smith at (423) 843-6672.

Sincerely,

Pedro Salas
Licensing and Industry Affairs Manager

Subscribed and sworn to before me
on this 11th day of March

Peggy M. Billingsley
Notary Public

My Commission Expires October 9, 2002

Enclosures

ENCLOSURE

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 01-08

RAI Question 1:

Provide results of an Anticipated Transient Without Scram (ATWS) analysis demonstrating that the plant at power uprate conditions is within the bounds considered by the staff during your documentation of compliance with the ATWS rule (Code of Federal Regulations at 10 CFR 50.62). For your power uprating discuss and justify that the assumptions for the ATWS analysis are adequate as they relate to input parameters such as the initial power level, moderator reactivity feedback, safety relief valves capacity and auxiliary feedwater supply. The submittal should include a discussion and applicable values of the unfavorable exposure time, if any, and ATWS core damage frequency.

Response:

For Westinghouse pressurized water reactors (PWRs), the licensing requirements related to ATWS are those specified in the Final ATWS Rule, 10CFR50.62(b). The requirement set forth in 10CFR50.62 (b) is that all Westinghouse designed PWRs must install ATWS Mitigation System Actuation Circuitry (AMSAC). In compliance with 10CFR50.62(b), AMSAC has been installed and implemented at SQN Units 1 & 2.

As documented in SECY-83-293 (Reference 1), the analytical bases for the final ATWS rule are the generic ATWS analyses for Westinghouse PWRs generated by Westinghouse in 1979. These generic ATWS analyses were formally transmitted to the NRC via letter NS-TMA-2182 (Reference 2) and were performed based on the guidelines provided in NUREG-0460 (Reference 3).

In the generic ATWS analyses documented in NS-TMA-2182 (Reference 2), ATWS analyses were performed for the various American Nuclear Society Condition II events (i.e., anticipated transients) considering various Westinghouse PWR configurations applicable at that time. These analyses included 2-, 3-, and 4-Loop PWRs with various steam generator models. For the SQN units, the generic ATWS analyses applicable at that time are those for a 4-Loop PWR with Model 51 steam generators and a core power of 3411 Mega-Watt Thermal (MWt) (nuclear steam supply system [NSSS] power of 3423 MWt). These conditions are summarized in Table 3-1-a of NS-TMA-2182.

The SQN units are currently licensed with a core power of 3411 MWt and both SQN units still operate with the original Westinghouse Model 51 steam generators. Hence, the generic ATWS analyses documented in NS-TMA-2182 continue to appropriately reflect the current plant configuration and licensed power level for SQN Units 1 and 2.

The generic ATWS analyses documented in NS-TMA-2182 also support the analytical basis for the NRC approved generic AMSAC designs generated for the Westinghouse Owners Group (WOG) as documented in WCAP-10858-P-A, Revision 1 (Reference 4). For the purpose of these AMSAC designs, the generic ATWS analyses for the 4-Loop PWR configuration with Model 51 steam generators were used to conservatively represent all of the various Westinghouse PWR configurations contained in NS-TMA-2182. For the SQN units, TVA has employed WCAP-10858-P-A AMSAC Logic 1, AMSAC Actuation on Low Steam Generator Water Level.

For the subject power uprating, an increase from an NSSS power of 3423 MWt to an NSSS power of 3467 MWt is proposed. This reflects a power increase of 1.3% above that considered in the generic ATWS analysis for the 4-Loop PWRs with Model 51 steam generators. As documented in NS-TMA-2182, an increase in core thermal power adversely affects the results of the ATWS analyses. As reported for the generic 4-Loop PWR with Model 51 steam generators, an increase in power of 2%, increases peak reactor coolant system (RCS) pressure by 44 pounds per square inch in the limiting loss of load ATWS event. As demonstrated in NS-TMA-2182, the peak RCS pressure with the 2% increase in power remains below 3200 pounds per square inch gauge. This ATWS sensitivity analysis was performed assuming a 2% variation in power, consistent with the typical calorimetric measurement uncertainty on power at the time of these analyses. The proposed increase in power of 1.3% is within the applicable range of the 2% increase in power assumed in the sensitivity analysis.

As prescribed by NUREG-0460, the 1979 generic ATWS analyses for Westinghouse PWRs documented in NS-TMA-2182 assumed a full power moderator temperature coefficient (MTC) of -8 pcm/°F. A sensitivity analysis including the use of an MTC of -7 pcm/°F was also provided as prescribed by NUREG-0460. In 1979, the MTC values of -8 pcm/°F and -7 pcm/°F represented MTCs that Westinghouse PWRs would be more negative than for 95% and 99% of the cycle, respectively. The base case of 95% represents a 95% confidence limit on favorable MTC for the fuel cycle. For the SQN units, the TS requirement on MTC is limited to < 0 pcm/°F at all power levels. Hence, the current MTC TSs for the SQN units remains the same as that which was applicable for most Westinghouse PWRs in 1979. Hence, the reactivity feedback for the SQN units remains sufficiently negative to be comparable to the generic Westinghouse ATWS analyses presented in NS-TMA-2182. It should be noted that reactivity feedback performance of the SQN units was specifically addressed for ATWS in the Licensing Amendment Request for operation with tritium-producing burnable absorber rods (TPBARs). As shown in Reference 5, the MTC at hot full power conditions for the SQN units is sufficiently negative relative to the MTC assumption used in NS-TMA-2182 and is even more negative considering operation with TPBARs. The NRC acceptance of this for the SQN units relative to ATWS is documented in the NRC Safety Evaluation Report, Reference 6.

Relative to the other aspects of RAI Question 1, the safety valve relief capacity and auxiliary feedwater capacity is unaffected by the proposed 1.3% power uprate as documented in Sections 2.3.1.1 and 2.3.2.5, respectively, of WCAP-15726 (Reference 7). The design capacity of each of the three SQN pressurizer safety relief valves is 420,000 pounds mass per hour (lbm/hr). This is consistent with the pressurizer safety valve relief capacity assumed in the 1979 generic ATWS analyses for the Westinghouse 4-Loop plant configuration as documented in NS-TMA-2182.

Both of the original SQN pressurizer power operated relief valves (PORVs) supplied by Westinghouse were replaced in 1983 with TVA procured valves manufactured by Target Rock. The Target Rock PORVs are each certified to a maximum flow rate of 210,000 lbm/hr at 2339 pounds per square inch absolute and 650°F. Hence, the pressure relief capacities of the pressurizer safety valves and PORVs at SQN are considered to be consistent with those modeled in the 1979 generic ATWS analyses for the Westinghouse 4-Loop plant configuration as documented in NS-TMA-2182.

For SQN, the design capacities of the auxiliary feedwater (AFW) pumps are as follows:

- Motor-Driven AFW Pump - 440 gallons per minute (gpm) at 2900 feet of head
- Turbine-Driven AFW Pump - 880 gpm at 2600 feet of head

The SQN AFW system has two motor-driven AFW pumps (each pump aligned to two steam generators) and one turbine-driven AFW pump (aligned to all four steam generators). Hence, the design of the SQN AFW system is consistent with the total AFW system capacity of 1760 gpm assumed in the 1979 generic ATWS analyses for the Westinghouse 4-Loop plant configuration as documented in NS-TMA-2182.

Based on the above, it is concluded that operation of the SQN units at an uprated NSSS power of 3467 MWt remains within the bounds of the generic Westinghouse ATWS analysis documented in

NS-TMA-2182 and, therefore, would remain in compliance with the final ATWS rule, 10CFR50.62(b).

References:

1) SECY-83-293, "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," W. J. Dircks, July 19, 1983.

2) NS-TMA-2182, Letter from T. M. Anderson (Westinghouse) to Dr. S. H. Hanauer (NRC) dated December 30, 1979, "ATWS Submittal."

3) NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," December, 1978.

4) Westinghouse Topical Report, WCAP-10858-P-A, Rev. 1, "AMSAC Generic Design Package," M. Adler, June 1985.

5) TVA letter S64 000920 801, "Sequoyah Nuclear Plant (SQN) - Tritium Production Program - Anticipated Transients Without Scram (ATWS)," Docket Nos. 50-327 and 50-328, P. Salas (TVA) to USNRC, September 29, 2000.

6) NRC SER, "Sequoyah Units 1 and 2, and Watts Bar, Unit 1, RE: Tritium Production Program - NUREG-1672 Interface Issue 17 - Anticipated Transient Without Scram Analyses (TAC Nos. MA9583 and MB0515), L. M. Padovan (USNRC) to Mr. J. A. Scalice (TVA), March 16, 2001.

7) WCAP-15726, "Sequoyah Units 1 and 2 1.3-Percent Power Uprate Program Licensing Report," November 2001.

RAI Question 2:

Page 3-10 of Westinghouse Topical Report WCAP-15726 discusses use of the BWCMV-A critical heat flux (CHF) correlation and statistical core design (SCD) methodology for the departure from nucleate boiling ratio (DNBR) reanalysis. The licensee is requested to list the titles of topical reports that document the CHF correlation and SCD methodology, and reference the associated NRC acceptance letters to confirm the acceptance of the correlation and SCD methodology used in the DNBR reanalysis for power uprate applications. Provide a discussion to address the compliance with each of limitations and restrictions specified in NRC safety evaluations for the applicable topical reports.

Response:

The topical reports that document the BWCMV-A CHF correlation and the SCD methodology supporting the existing SQN licensing basis (3411 MWt) and also used for the proposed power uprate licensing basis (3455 MWt) are References 1, 2, and 3. These topical reports were applied in demonstrating acceptable performance of the Mark-BW fuel assembly when the design was being introduced to the SQN cores, as described in 7.1.1 LYNXT Modeling of Reference 4. The BWCMV CHF correlation was based on an extensive set of inconel mixing grid CHF test data. However, in August 1993, Framatome Cogema Fuels submitted to the NRC additional test data (in Reference 3) that showed that the Mark-BW zircaloy mixing grid performed at a level that was superior to the original BWCMV database. This enhanced CHF performance of the Mark-BW mixing grid has been incorporated into the nucleate boiling (DNB) analyses through the use of design-specific equivalent grid spacing. When using the BWCMV correlation in this manner, referenced specifically to Reference 3, it is generally referred to as BWCMV-A.

The NRC acceptance letter, Reference 5, for the submitted SCD methodology identified limitations for the approval. These limitations have been incorporated in the routine use of the methodology for the SQN units. The limitations are written to capture the need to (1) review the component uncertainties and their distributions for their applicability, (2) demonstrate the "bounding" assembly-wise power distributions assumed in the core-wide statistical design limit (SDL) bounds expected cycle-specific power distributions, and (3) validate and revise (as necessary) the response surface model when applying it to extended operating conditions. For the proposed power uprate condition, all three limitations have been examined. The component uncertainties distributions remain intact for the uprate condition. In addition, the component uncertainties are initially being maintained. Although maintaining the 2.0% core power uncertainty within the SDL basis is a conservative measure when the expected core power uncertainty could be as low as 0.7% for the power uprate condition, TVA has elected to delay the permissible extraction of this conservatism. During the power uprate, the cycle-specific power distributions will continue to be examined according to limitation 2. With regard to limitation 3, Framatome has concluded that the response surface model (RSM) remains applicable for a power uprate of this magnitude. Note in Reference 1 that the RSM range extends to 130% core power, which is well beyond the conditions establishing DNB-based safety limits and alarms.

The NRC acceptance letters for the BWCMV and BWCMV-A CHF correlation topical reports, References 2 and 3, respectively, are available in References 6 and 7, respectively. The limitations defined in the Safety Evaluation Reports (SERs) for these CHF correlation topical reports are primarily associated with application of the correlation to specific fuel assembly designs, within specific local coolant conditions, and using a specifically approved thermal-hydraulic code. These restrictions are being satisfied for the uprate power condition. The application of these correlations for SQN at a power uprate condition for the same fuel designs using the same approved codes demonstrate the predicted DNBRs are acceptable as a result of the nominal core power increase.

References:

1) BAW-10170PA, "Statistical Core Design for Mixing Vane Cores," Babcock & Wilcox, Lynchburg, Virginia, December 1988.

2) BAW-10159P-A, "BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies," B&W Fuel Company, Lynchburg, Virginia, July 1990.

3) BAW-10189P-A, "CHF Testing and Analysis of the Mark-BW Fuel Assembly Design," Framatome Cogema Fuels, Lynchburg, Virginia, January 1996.

4) BAW-10220P, "Mark-BW Fuel Assembly Application for Sequoyah Nuclear Units 1 & 2," Framatome Cogema Fuels, Lynchburg, Virginia, March 1996. (see Reference 25 in Enclosure 2 of letter, Ronald W. Hernan [NRC] to Oliver D. Kingsley [TVA], Issuance of Technical Specification Amendments for the Sequoyah Nuclear Plant, Units 1 and 2 [TAC Nos. M95144 and M95145] [96-01], April 21, 1997.)

5) Letter, Ashok C. Thadani (NRC) to J. H. Taylor (B&W), Acceptance for Referencing of Topical Report BAW-10170P, "Statistical Core Design for Mixing Vane Cores" (TAC No. 66318), September 14, 1988.

6) Letter, Ashok C. Thadani (NRC) to J. H. Taylor (B&W), Acceptance for Referencing of Augmented Topical Report BAW-10159P, "BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies," May 22, 1989.

7) Letter, Gary M. Holahan (NRC) to J. H. Taylor (B&W Fuel Company), Acceptance for Referencing of Babcock & Wilcox Fuel Company Topical report BAW-10189(P), "CHF Testing and Analysis of the Mark-BW Fuel Assembly Design," January 3, 1995.

RAI Question 3:

As stated in the NRC Standard Review Plan (SRP), one of the acceptance criteria for the transient analysis is related to the calculated DNBR. The staff finds that the information regarding the DNBR reanalysis for power uprate provided in Sections 3.3.7, 3.3.8 and 3.3.9 of WCAP-15726 involves qualitative discussion in nature. The licensee is requested to list the events with calculated DNBRs affected by power uprating and provide calculated results (such as figures showing calculated DNBRs, or margins to DBNR safety limits) for these events to show that the *calculated DNBRs for power uprate conditions are within the acceptable safety limits. If the results of reanalysis are more limiting than the analysis of record, the reanalysis results should be included in the updated FSAR.*

Response:

The DNBR predictions for numerous transient analyses were determined for the power uprate condition. These limiting DNB transients included the four reactor coolant pump flow coastdown, turbine trip, steam line break, rod withdrawal at power, locked rotor, and ejected rod transients. Table Q3.1 lists the transient minimum DNBRs, the DNB design criteria, and margin to the criteria. For the SQN cores at the power uprate condition, the DNB design criteria is based on the thermal design limit (TDL) established for the plant/cycle. For the case of SQN at 3455 MWt, a TDL of 1.431 was used for BWCMV-A. As described in Appendix E of Reference 1, the TDL is a means of establishing retained DNB margin above the SDL. For the power uprate condition, approximately 6% margin corresponding to 100% (TDL-SDL)/TDL = 100% (1.431 - 1.345)/1.345 has been retained.

With respect to the last sentence of the question, the current TVA design change process requires review and update of the Final Safety Analysis Report (FSAR) as necessary for all plant modifications. Recommended FSAR changes have been submitted by Framatome Advanced Nuclear Power for TVA review and consideration in support of the SQN power level uprate.

Table Q3.1
DNB Predictions for Re-Analyzed Transients
at the Power Uprate Conditions

Transient Event	Predicted Minimum DNBR	DNB Design Criteria	Margin to Criteria	Comments
Four RC Pump Coastdown	1.88	1.431	45 DNB points[1]	
Turbine Trip	2.08	1.431	65 DNB points	
Steam Line Break	1.69	1.20[2]	50 DNB points	
Rod Withdrawal	2.12 2.04 1.84 1.56 1.50	1.431	69 DNB points 61 DNB points 41 DNB points 13 DNB points 7 DNB points	+7 pcm/°F, -12.0 pcm/%FP, 75 pcm/sec +7 pcm/°F, -6.5 pcm/%FP, 75 pcm/sec -45 pcm/°F, -12.5 pcm/%FP, 0.82 pcm/sec -45 pcm/°F, -12.5 pcm/%FP, 15 pcm/sec 0 pcm/°F, -6.5 pcm/%FP, 15 pcm/sec
Locked Rotor	< 1.431	1.431	0 DNB points	The locked rotor event has <5% pins in DNB.
Ejected Rod	< 1.431	1.431	0 DNB points	The ejected rod event has <10% pins in DNB.

[1] where 1 DNB point = 0.01

[2] The steam line break transient has local conditions outside the range of the BWCMV-A CHF correlation, therefore, the NRC approved BWU-Z CHF correlation is used, see Reference 2. The BWU-Z CHF correlation approved ranges bound the local conditions for the

event. A non-SCD, or deterministic, evaluation was performed that conservatively applied uncertainties directly to their associated statepoint parameters.

References:

1) BAW-10170PA, "Statistical Core Design for Mixing Vane Cores," Babcock & Wilcox, Lynchburg, Virginia, December 1988.

2) BAW-10199P-A, "The BWU Critical Heat Flux Correlations," Framatome Cogema Fuels, Lynchburg, Virginia, August 1996.

RAI Question 4:

Stress Corrosion Cracking of Reactor Internals

Increased power is expected to increase the corrosion rates and speed up degradation of reactor internals. Identify the plant programs that are in place to periodically inspect reactor internals and discuss whether these programs are adequate to manage the projected increase of reactor internals degradation due to stress corrosion cracking (SCC) and primary water stress corrosion cracking (PWSCC).

Response:

The reactor upper internals are removed and subject to a nominal visual inspection with every 18 month refueling outage. The reactor lower internals are removed and subject to a nominal visual inspection in conjunction with each 10-year reactor vessel in-service inspection. The relevant changes associated with the leading edge flow meter (LEFM) power uprate involve the susceptibility of Alloy 600 penetrations to PWSCC and a change in the susceptibility of the stainless steel reactor vessel internals to SCC. The proposed power uprate involves a very small (0.4°F) increase in the reactor vessel outlet temperature and no changes in the RCS chemistry control.

The increase in service temperature of 0.4°F is expected to result in a very small, insignificant increase in the PWSCC susceptibility of Alloy 600 reactor vessel head penetrations. By using the susceptibility equation in the reference, this increase in susceptibility is predicted at 1.8%. The occurrence of SCC for the stainless steel reactor vessel internals, either as Intergranular Stress Corrosion Cracking (IGSCC) or Transgranular Stress Corrosion Cracking (TGSCC), requires the presence of oxygen. The current RCS chemistry control precludes the presence of oxygen and no changes in RCS chemistry will be made associated the LEFM power uprate, therefore, no change in the susceptibility to material degradation of the stainless steel reactor vessel internals is expected. Although IGSCC and TGSCC can be accelerated with service temperature, a 0.4° F change in service temperature is not expected to result in any appreciable acceleration effect. On this basis, the proposed power uprate is not expected to have any appreciable impact on the reactor vessel and internals component materials and so, no inspection changes are warranted.

Reference:

Gutti Rao "Methodologies to Assess PWSCC Susceptibility of Primary Alloy 600 Locations in Pressurized Water Reactors" Proceedings of the Sixth International Symposium, NACE, August 1-5, 1993.

RAI Question 5:

Flow Assisted Cracking (FAC)

Since the effects of FAC on degradation of carbon steel components are plant specific, the licensee needs to provide a predictive analysis methodology which must include the values of the parameters affecting FAC, such as velocity, and temperature before and after the power uprate (PU) and the corresponding changes in components wear rates due to FAC.

Response:

Generic Letter 89-08 specifically deals with Erosion/Corrosion. The current industry terminology for Erosion/Corrosion is Flow Accelerated Corrosion. The responses provided to Questions 5-8 deal specifically with Flow Accelerated Corrosion.

The upper tier program requirements for Flow Accelerated Corrosion are specified in SPP-9.7, "Corrosion Control," Appendix B, "Technical Requirements for the Flow Accelerated Corrosion Program," and DS-M4.2.1, "Flow Accelerated Corrosion Program Methods."

The main Flow Accelerated Corrosion Program elements include the following:

1. Susceptibility Study: Delineates specific portions of systems deemed susceptible to the Flow Accelerated Corrosion phenomenon and reasons for exclusion of system portions from the Flow Accelerated Corrosion Program based on system parameters and fluid content. If piping is deemed susceptible to Flow Accelerated Corrosion, then it is documented whether the subject piping is maintained in the CHECWORKS models or included in the Non-Modeled Risk Based Evaluation.

2. CHECWORKS 1.0G: Electric Power Research Institute (EPRI) computer code utilized through out the industry to rank plant piping components deemed susceptible to Flow

Accelerated Corrosion. Components with well known operating and system parameters are included in this model.

3. <u>Non-Modeled Risk Based Evaluation:</u> Risk based ranking methodology for components with limited use or limited operating information. The components are ranked according to the relative susceptibility and the consequence of failure.

4. <u>Industry Experience Database</u>: Industry Flow Accelerated Corrosion experience items are reviewed for applicability to SQN. An industry experience item is prepared as required based on the review. An experience item includes the industry item number, brief description, areas to be addressed in the SQN Flow Accelerated Corrosion Program, and recommended outage for implementation. The plant experience item is closed as the recommendations are completed.

5. <u>Site Experience Database:</u> Site experience items are prepared when sample expansion is implemented during an outage to bound wall thinning. The intent is to ensure that wall thinning identified will be bound on the remaining unit. An experience item includes a brief description, areas to be addressed in the SQN Flow Accelerated Corrosion Program, and recommended outage for implementation. The plant experience item is closed as the recommendations are completed.

6. <u>Inspected Component Database:</u> Includes components that have previously been inspected. The database uses the remaining time to reach Tmin as guidance on which outage to reinspect the component.

7. <u>Flow Accelerated Corrosion Inspection List:</u> The outage inspection list is comprised of pick point locations obtained from the Flow Accelerated Corrosion Program Elements 1 through 6 listed above.

8. <u>Evaluation Methodology:</u> Incorporates a 10% safety factor on time as recommended in NSAC-202L. The safety factor provides assurance that the inspected components will continue to meet system requirements and maintain structural integrity.

Predictive Models:

CHECWORKS 1.0G is the current predictive model in use at SQN for Units 1 and 2 and includes parameters such as flow velocity, temperature, and pressure. CHECWORKS is a predictive tools which ranks components according to susceptibility and also provides wear rates based on parameters entered into the model.

The CHECWORKS model is being modified to incorporate the heat balance diagram information associated with the 1.3% power uprate. Lines are also being redefined to allow use of the advanced run definition capability associated with power uprate incorporation into the CHECWORKS 1.0G model.

Based on previous experience with the SQN CHECWORKS models, the 1.3% power uprate heat balance information, and power uprate experience at other utilities, CSI Technologies has provided preliminary information stating that the effects vary by analysis line based on the system parameter changes associated with that line during the power uprate. Because system parameter changes will vary by line, some lines will have a negligible change in wear rates while other lines will have the potential for a slight increase or slight decrease in the Flow Accelerated Corrosion wear rates.

Updates to the CHECWORKS model will be in place prior to the power uprate going into affect. Inspection scope and frequency will be adjusted as necessary based on the results from the model.

The Non-Modeled Risk Based components are ranked according to the relative susceptibility and the consequence of failure. The changes associated with the 1.3% power uprate are minimal and should not affect the risk based rankings.

Incorporation of the 1.3% power uprate into the Flow Accelerated Corrosion Program will not circumvent the basic elements or implementation of the Flow Accelerated Corrosion Program.

RAI Question 6:

Indicate the degree of compliance with NRC Generic Letter 89-08, "Erosion/Corrosion in Piping." This letter requires that an effective program be implemented to maintain structural integrity of high-energy carbon steel systems. Describe how this program was modified to account for the PU. If the computer code used in predicting wall thinning by FAC in this program is a generic code (e.g., CHECWORKS), specify it. However, if the code is plant specific provide its description.

Response:

The original response to Generic Letter 89-08 is documented in the letter from TVA to the NRC on July 19, 1989. The Flow Accelerated Corrosion Program at SQN meets and exceeds the requirements specified in the original response.

The Flow Accelerated Corrosion Program has continued to improve based on incorporation of changes in the industry, training, including NSAC-202L and EPRI recommendations. Flow Accelerated

Corrosion personnel remain cognizant of current issues through review of industry experience items and participation in the CHUG Users Group.

See the response to RAI Question 5 for additional details.

RAI Question 7:

Identify the predicted change of wear rates calculated by the revised code for the components most susceptible to FAC.

Response:

The CHECWORKS model for SQN is being modified to incorporate the heat balance diagram information associated with the 1.3% power uprate with no significant change to the wear rates predicted.

See the response to RAI Question 5 for additional details.

RAI Question 8:

Will the PU have significant effect on FAC in balance of plant (BOP) components? What is the value of the change in FAC wear rates?

Response:

Based on previous experience with the SQN CHECWORKS models, the 1.3% power uprate heat balance information, and power uprate experience at other utilities, CSI Technologies has provided preliminary information stating that the effects vary by analysis line based on the system parameter changes associated with that line during the power uprate. Because system parameter changes will vary by line, some lines will have a negligible change in wear rates while other lines will have the potential for a slight increase or slight decrease in the Flow Accelerated Corrosion wear rates.

See the response to RAI Question 5 for additional details.

RAI Question 9:

The response to Question 2 under TXX-99105 (page E8-3 of Enclosure 8) addresses the acceptability of previously performed equipment qualification analyses for a 1.3 percent power increase. Please provide a statement that the previous analyses envelope conditions that will exist after the 1.3 percent power increase, if that is the case.

Response:

The following conclusions were reached in evaluating the effect of the proposed 1.3% power uprate on equipment qualification:

- The 1.3% uprate has an insignificant effect on process fluid temperatures in the auxiliary, control, turbine and containment buildings. Therefore, normal environmental conditions are not affected.

- The current post-accident thermal environmental analyses were performed at 102% of the licensed power. Evaluations concluded that through the use of the reduced 0.7% power calorimetric uncertainty to offset the 1.3% increase in reactor power, the existing mass and energy releases used in the environmental analyses for both inside and outside containment would remain valid. Because the mass and energy releases are not changed, the resulting environments are also unchanged. Therefore, the 1.3% power uprate has no impact on the SQN non-radiological equipment qualification program.

- The current radiological design basis was performed in accordance with Regulatory Guide 1.49 which requires the normal power level to be 102% of the licensed power. For both post-accident and normal-operating, the SQN source terms were based on 104.5% of the licensed power or greater. Therefore, it can safely be concluded that a power uprate of 1.3% would not cause dose rates or integrated doses to exceed design basis values and equipment qualification is not affected.

In summary, the current analyses for post-accident thermal and radiological environmental consequences envelop conditions that will exist after the 1.3% power uprate. In addition, the 1.3% power uprate has a negligible effect on normal environmental conditions in the auxiliary, control, turbine, and containment buildings.