

ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNIT 2

DOCKET NO. 50-328

(TVA-SQN-TS-88-33)

LIST OF AFFECTED PAGES

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POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR- $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2  $F_Q(Z)$  shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{[2.237]^\#}{P} [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq \frac{[2.237]^\#}{0.5} [K(Z)] \text{ for } P \leq 0.5$$

where  $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

and  $K(Z)$  is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1

ACTION:

With  $F_Q(Z)$  exceeding its limit:

- a. Reduce THERMAL POWER at least 1% for each 1%  $F_Q(Z)$  exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower Delta T Trip Setpoints (value of  $\Delta T_4$ ) have been reduced at least 1% (in  $\Delta T$  span) for each 1%  $F_Q(Z)$  exceeds the limit.
- b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; THERMAL POWER may then be increased provided  $F_Q(Z)$  is demonstrated through incore mapping to be within its limit.

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SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

*Insert  
Footnote "A" →*

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.2  $F_Q(z)$  shall be evaluated to determine if  $F_Q(Z)$  is within its limit by:

- a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.
- b. Increasing the measured  $F_Q(z)$  component of the power distribution map by 3 percent to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.
- c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{2.237^{**} \times K(z)}{P \times W(z)} \text{ for } P > 0.5$$

$$F_Q^M(z) \leq \frac{2.237^{**} \times K(z)}{W(z) \times 0.5} \text{ for } P \leq 0.5$$

where  $F_Q^M(z)$  is the measured  $F_Q(z)$  increased by the allowances for manufacturing tolerances and measurement uncertainty,  $F_Q$  limit is the  $F_Q$  limit,  $K(z)$  is given in Figure 3.2-2,  $P$  is the relative THERMAL POWER, and  $W(z)$  is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.14.

- d. Measuring  $F_Q^M(z)$  according to the following schedule:
  1. Upon achieving equilibrium conditions after exceeding by 10 percent or more of RATED THERMAL POWER, the THERMAL POWER at which  $F_Q(z)$  was last determined \* or
  2. At least once per 31 effective full power days, whichever occurs first.

\*During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

*Insert Footnote "A" →*

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

e. With measurements indicating

$$\text{maximum over } z \left( \frac{F_Q^M(z)}{K(z)} \right)$$

has increased since the previous determination of  $F_Q^M(z)$  either of the following actions shall be taken:

1.  $F_Q^M(z)$  shall be increased by 2 percent over that specified in 4.2.2.2.c, or
2.  $F_Q^M(z)$  shall be measured at least once per 7 effective full power days until 2 successive maps indicate that

$$\text{maximum over } z \left( \frac{F_Q^M(z)}{K(z)} \right) \text{ is not increasing.}$$

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f. With the relationships specified in 4.2.2.2.c above not being satisfied:

1. Calculate the percent  $F_Q(z)$  exceeds its limit by the following expression:

$$\left\{ \begin{array}{l} \text{maximum over } z \\ \left[ \frac{F_Q^M(z) \times W(z)}{\frac{2.237}{P} \times K(z)} \right] \end{array} \right\}^{-1} \times 100 \quad \text{for } P \geq 0.5$$

$$\left\{ \begin{array}{l} \text{maximum over } z \\ \left[ \frac{F_Q^M(z) \times W(z)}{\frac{2.237}{0.5} \times K(z)} \right] \end{array} \right\}^{-1} \times 100 \quad \text{for } P < 0.5$$

2. Either of the following actions shall be taken:

- a. Place the core in an equilibrium condition where the limit in 4.2.2.2.c is satisfied. Power level may then be increased provided the AFD limits of Figure 3.2-1 are reduced 1% AFD for each percent  $F_Q(z)$  exceeded its limit, or
- b. Comply with the requirements of Specification 3.2.2 for  $F_Q(z)$  exceeding its limit by the percent calculated above.

*Insert Footnote "A"*

Footnote "A"  
to be  
Inserted on Pages

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# The limit shall be 2.15 instead of 2.237 until an analysis in conformance with 10 CFR 50.46, using plant operating conditions and showing that a limit of 2.237 satisfies the requirements of 10 CFR 50.46(b), has been completed and submitted to NRC.

EMERGENCY CORE COOLING SYSTEMS (ECCS)

SURVEILLANCE REQUIREMENTS (Continued)

- b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the solution in the water-filled accumulator.
- c. At least once per 18 months by:
1. Verifying that each accumulator isolation valve closes automatically when the water level in the water-filled accumulator is ~~92.0 + 2.4 / - 5.8~~ ~~82.1 ± 5.6~~ inches above the tank vendor working line. This corresponds to ~~37.1 ± 5.6~~ inches when corrected for the mass of cover gas.
  2. Verifying that the total dissolved nitrogen and air in the water-filled accumulator is less than 80 SCF per 1800 cubic feet of water (equivalent to  $5 \times 10^{-5}$  pounds nitrogen per pounds water).
- d. At least once per 5 years by removing the membrane installed between the water-filled and nitrogen bearing accumulators and verifying that the removed membrane bursts at a differential pressure of  $40 \pm 10$  psi.

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ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNIT 2

DOCKET NO. 50-328

(TVA-SQN-TS-88-33)

DESCRIPTION AND JUSTIFICATION FOR  
REVISING UHI LEVEL SWITCH SETPOINT AND TOLERANCES  
AND REDUCTION IN HEAT FLUX HOT CHANNEL FACTOR LIMIT

## ENCLOSURE 2

### DESCRIPTION OF CHANGE

Tennessee Valley Authority proposes to modify the Sequoyah Nuclear Plant (SQN) unit 2 technical specifications to revise the upper head injection (UHI) level switch setpoint and tolerances of surveillance requirement (SR) 4.5.1.2.c.1 and the heat flux hot channel factor ( $F_0[z]$ ) of limiting condition for operation (LCO) 3.2.2 and SR 4.2.2.2. This proposed revision to the SQN unit 2 UHI technical specifications is consistent with the SQN unit 1 technical specification proposed change 88-20 (submitted August 15, 1988; and supplemented by letter dated September 21, 1988; which NRC approved by letter dated October 14, 1988) and 88-28 (submitted September 21, 1988).

### REASON FOR CHANGE

Condition adverse to quality report (CAQR) SQPS71644 documents that the level switches and setpoints that were used previously could allow more than the analytical limit of 1,130.5 cubic feet of UHI water to be injected during a postulated accident. Two changes in the design and configuration of the UHI system were pursued to correct this potential problem. First, the minimum delivered UHI water volume was reduced from 900 cubic feet to 850 cubic feet. This change is supported by Westinghouse Electric Corporation (W) evaluations described in a September 14, 1988 letter to TVA (included as attachment 1). Second, a new model of level switch is being installed in the UHI system. These new switches are essentially the same as those presently used, except for their span. Because of the span differences, the switches also have different accuracy characteristics. Demonstrated Accuracy Calculation 1-LS-87-21 determined a new setpoint and tolerances based on the new instrument characteristics. These new values are being incorporated into SR 4.5.1.2.c.1 to ensure that the delivered UHI water volumes are bounded by the volumes assumed in the large-break, loss of coolant accident (LOCA) analyses. This in turn ensures that the offsite doses from a postulated LOCA are bounded by the analyses of the Final Safety Analysis Report (FSAR), section 15.5.

The change in the delivered UHI water volume band described above is supported by W evaluations, which indicated that the potential decrease in delivered water volume to the core would result in increased peak clad temperatures (PCTs); but in all cases, PCT remained below the 2,200-degrees-Fahrenheit (F) limit of 10 CFR 50.46. NRC has indicated that operation of unit 2 could be supported by the sensitivity studies (provided a temporary exemption to certain administrative requirements of 10 CFR 50.46(a)(1) was obtained) and that operational restrictions be imposed to provide at least 100 degrees F of margin between the calculated PCT and the 10 CFR 50.46 limit.



Evaluations by W have determined that at least 100 degrees F PCT margin can be obtained by administratively limiting steam generator tube plugging (SGTP) to 5 percent and by reducing  $F_0(z)$  from 2.237 to 2.15. The proposed  $F_0(z)$  limit change is being submitted to reflect this operational restriction.

TVA's request for a temporary exemption to certain administrative requirements of 10 CFR 50.46(a)(1) was provided by separate correspondence.

#### JUSTIFICATION FOR CHANGE

##### Delivered UHI Water Volume

The UHI system is designed to passively supply additional inventory to the reactor core during the blowdown phase of a postulated LOCA. The UHI system is described in FSAR section 6.3.2. As described in FSAR section 15.4.1.1.4, a broad spectrum of LOCA analyses has been performed to evaluate UHI performance. The various UHI performance analyses are categorized by the assumed discharge coefficient ( $C_D$ ) of the break and the presence or lack of UHI water mixing in the upper head region of the vessel (perfect and imperfect mixing, respectively).

The limiting case break in the UHI Evaluation Model emergency core cooling system (ECCS) analysis presented in the original SQN FSAR was the discharge coefficient  $C_D=0.6$  double-ended, cold-leg guillotine (DECLG) break with imperfect mixing of UHI water assumed in the vessel upper head. Compliance with regulatory limits was achieved for this case by reducing the allowable core peaking factor ( $F_q$ ) from 2.32 to 2.237. Minimizing the volume of UHI water delivered maximizes PCT for imperfect mixing UHI LOCA cases. The lower bound value for UHI water volume delivery established in the original FSAR  $C_D=0.6$  DECLG imperfect mixing case is 900 cubic feet. This value also was employed in the imperfect mixing cases of the 10-percent SGTP analysis performed in the 1982-83 timeframe.

A complete spectrum of perfect mixing cases was analyzed for the original SQN FSAR. The limiting case with perfect mixing of UHI water assumed in the vessel upper head was the  $C_D=0.6$  DECLG; the calculated PCT for this case is 2,111 degrees F at an  $F_q$  of 2.32 with a UHI-delivered water volume of 1,053 cubic feet.

Using sensitivities appropriate to UHI plant perfect mixing cases, tradeoffs have previously been made among various input assumptions to justify increasing the maximum allowable UHI-delivered water volume to 1,130.5 cubic feet. Increasing the value of UHI water delivered maximizes PCT for perfect mixing UHI LOCA analyses. With the present technical specification  $F_q$  of 2.237 in force, 1,130.5 cubic feet is a valid maximum delivered water volume for the SQN UHI system because it results in a PCT of 2,163 degrees F.

It should be noted that separate safety evaluations performed for SQN have considered the impacts on PCT of guide tube flexure failures, increased feedwater isolation valve stroke time, reduced safety injection flow from a failed residual heat removal pump miniflow, and thimble tube filling during core reflood. For the perfect mixing cases, these scenarios do not impact PCT; and 2,163 degrees F remains the limiting PCT for perfect mixing cases.

The  $C_D=0.8$  and  $C_D=0.6$  DECLG imperfect mixing cases from the 1982-83 10-percent SGTP analysis have been reviewed to assess the PCT impact of reducing the delivered UHI water volume to 850 cubic feet. The calculated PCTs for the  $C_D=0.8$  and  $C_D=0.6$  DECLG cases that comprise the current licensing basis for SQN are 2,111 degrees F and 2,113 degrees F, respectively. Reducing the UHI water delivery in an imperfect mixing case will reduce the cooling of the fuel as the upper head drains during blowdown. During the core reflood phase, this hotter fuel will then expel more injection water as entrained liquid, producing a degraded flooding rate. Existing SQN imperfect mixing cases performed for the FSAR identify the penalty in core fuel heatup associated with decreasing UHI water delivery to 850 cubic feet, which reduces core inlet velocity by 7 percent for the licensing basis imperfect mixing cases.

The impact of degraded flooding rates upon hot rod calculated PCT has been determined by WREFLOOD/LOCTA sensitivity runs for each licensing basis imperfect mixing case. The 10-percent SGTP licensing basis imperfect mixing cases are acceptable at an 850-cubic-foot-delivered UHI water volume because the degraded reflood penalty only increases calculated PCT as follows:

$C_D=0.8$ DECLG	PCT = 2,151 degrees F
$C_D=0.6$ DECLG	PCT = 2,166 degrees F

The PCT penalties imposed upon the imperfect mixing cases are 20 degrees F for postulated guide tube flexure failures and 12 degrees F for thimble tube filling during core reflood. Because the net PCT for the limiting imperfect mixing  $C_D=0.6$  DECLG case becomes 2,166 degrees F + 20 degrees F + 12 degrees F = 2,198 degrees F, compliance with the regulatory limit is maintained.

Both the perfect and imperfect mixing cases of the SQN large-break LOCA analysis remain in compliance with 10 CFR 50.46 if the UHI water-delivered volume is within the bounds of 850-1,130.5 cubic feet.

#### Calculation of Level Switch Setpoints

As described in FSAR section 6.3.2, four automatic hydraulic isolation valves are used to isolate the UHI accumulators from the reactor coolant system (RCS) after UHI has injected. These valves receive automatic closure signals from level switches on the UHI water accumulator. The level switch setpoints are selected to ensure that the delivered UHI water volume is within the limits described above.

Demonstrated Accuracy Calculation 1-LS-87-21, included in the August 15, 1988 letter, generates the level switch setpoint and tolerances that ensure that the delivered UHI water volume is between 850 and 1,130.5 cubic feet. As seen on page 6 of the calculation, a tank level of 95.2 inches (above the working line) equates to a delivered volume of 850 cubic feet; and a tank level of 85.1 inches equates to a delivered volume of 1,130.5 cubic feet. The calculation then continues to establish setpoint and tolerance between 95.3 and 85.1 inches. Pages 7 through 23A are a compilation of the various inaccuracies associated with the level switches, including drift characteristics. The limiting inaccuracies of +3.29 inches and -6.83 inches are calculated on page 22. Because of the nature of the drift characteristics, a curve-fit program was utilized to determine the optimum setpoint for the level switches. As described on page 22 of the calculation, the optimum setpoint is calculated to be 92 inches. This yields limiting level switch setpoints of 95.29 inches to 85.17 inches, which are within the analytical limits described above.

The tolerances used in the revised SR of +2.6/-5.8 inches represent the normal accuracy of the level switches excluding process variables that are unmeasurable at the time of calibration (see pages 3 and 14).

As calculated on page 22 and shown on page 23, the accuracy characteristics of the level switches necessitate calibration at least every 480 days. This level switch calibration is independent of the level switch/isolation valve functional response test required by SR 4.5.1.2.c.1. As such, the level switch calibrations will be scheduled and tracked independently. This will also allow for the extension of the calibration intervals based on evaluation of the new level switch performance. The calibration evaluations are in accordance with our previous commitment made in response to NRC Bulletin 86-02. TVA will continue to monitor level switch performance through the normal reporting process.

#### F<sub>Q</sub>(z) Reduction

As defined in SQN FSAR section 4.3.2.2.1,  $F_Q(z)$  is the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux. Limiting this ratio minimizes the magnitude of localized "hot spots" along the fuel cladding surface. This in turn helps ensure that PCTs will remain below the 10 CFR 50.46 limit of 2,200 degrees F during postulated LOCA conditions.

The proposed reduction in  $F_Q(z)$  is a conservative change and will provide additional margin in PCT. As described in the attached W evaluation (page 4), a reduction in  $F_Q(z)$  from 2.237 to 2.15 reduces PCT by 87 degrees F for the limiting imperfect mixing case and by 96 degrees F for the limiting perfect mixing case. As summarized on page 5 of the evaluation, this PCT reduction, combined with the reduction obtained by administratively limiting SGTP to 5 percent, results in PCTs of 2,089 degrees F for the limiting imperfect mixing case and 2,067 degrees F for the limiting perfect mixing case. As can be seen, these PCT values provide over 100 degrees of margin to the regulatory PCT limits.

ATTACHMENT 1

Technical Specification Change 88-33

W Letter Dated September 14, 1988  
(B25 880927 004)

# QA Record



B25 880927 004

September 14, 1988

Westinghouse  
Electric Corporation

Power Systems

Nuclear Technology  
Systems Division

Box 355  
Pittsburgh Pennsylvania 15230-0355

Mr. P. G. Trudel  
Sequoyah Project Engineer  
Tennessee Valley Authority  
Sequoyah Nuclear Power Plant, DSC-A  
P. O. 2000  
Soddy Daisy, TN 37379

TVA-88-761  
NS-OPLS-OPL-II-88-572  
Ref. 1) TVA RD #428373  
2) W.G.O. CO-42680  
3) TVA-88-746

Contract #  
85P62-965930

TENNESSEE VALLEY AUTHORITY  
SEQUOYAH UNITS 1 & 2  
DECREASED UHI VOLUME DELIVERY LOCA SAFETY EVALUATION  
(SECL-88-417, Revision 1)

Dear Mr. Trudel:

In accordance with our telecon of September 7, 1988, the LOCA safety evaluation provided in Reference 3 has been revised to reflect the impact of reducing F(Q) and SGTP, and a supplemental information document is being provided in response to the NRC request for additional information addressing the LOCA models referenced, clarification of the appropriate limiting breaks, and clarification of the effect of the postulated instrumentation thimble and guide tube flexure failures.

The revised LOCA safety evaluation, SECL-88-417, Revision 1, entitled, Safety Evaluation for a 50 Cubic Feet Decrease in the UHI Accumulator Deliverable Water Volume (LOCA, SGTR, Post-LOCA Long Term Core Cooling and Hot Leg Switchover Accident), is attached. This revision incorporates the impact of reducing F(Q) from 2.32 to 2.15 and the Steam Generator Tube Plugging (SGTP) level from 10% to 5%.

The supplemental information document is also attached and is entitled Supplemental Information to SECL-88-417, Revision 1.

If you have any comments or questions, please contact the undersigned.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

*T. A. Lordi*  
T. A. Lordi, Manager  
ESSD Projects  
Mid-South Area

L. V. Tomasic/tu  
Attachment

cc: D. W. Wilson                      W. R. Mangiante                      S. J. Smith  
R. W. Meadows                      J. A. Vogel                              M. J. Burzynski  
R. C. Weir                              R. G. Davis                              R. E. Daniels  
M. J. Ray

RIMS. SL 28 C-K - w/attachment

SEP 19 1988

WESTINGHOUSE  
NUCLEAR SAFETY EVALUATION CHECK LIST

- 1) NUCLEAR PLANT(S) SEQUOYAH UNITS 1 AND 2 (TVA/TEN)
- 2) CHECK LIST APPLICABLE TO: SAFETY EVALUATION FOR A 50 CU.FT. DECREASE IN  
(subject of Change) THE UHI ACCUMULATOR DELIVERABLE WATER VOLUME
- 3) The written safety evaluation of the revised procedure, design change or modification required by 10CFR50.59 has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on Page 2.  
Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

CHECK LIST - PART A

- (3.1) Yes  No  A change to the plant as described in the FSAR?
- (3.2) Yes  No  A change to procedures as described in the FSAR?
- (3.3) Yes  No  A test or experiment not described in the FSAR?
- (3.4) Yes  No  A change to the plant technical specifications  
(Appendix A to the Operating License)?
- 4) CHECK LIST - PART B (Justification for Part B answers must be included on Page 2.)
  - (4.1) Yes  No  Will the probability of an accident previously evaluated in the FSAR be increased?
  - (4.2) Yes  No  Will the consequences of an accident previously evaluated in the FSAR be increased?
  - (4.3) Yes  No  May the possibility of an accident which is different than any already evaluated in the FSAR be created?
  - (4.4) Yes  No  Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
  - (4.5) Yes  No  Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
  - (4.6) Yes  No  May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
  - (4.7) Yes  No  Will the margin of safety as defined in the bases to any technical specification be reduced?

If the answers to any of the above questions are unknown, indicate under 5) REMARKS and explain below.

If the answer to any of the above questions in 4) cannot be answered in the negative, based on written safety evaluation, the change cannot be approved without an application for license amendment submitted to NRC pursuant to 10CFR50.90.

5) REMARKS:

The following summarizes the justification upon the written safety evaluation, (1) for answers given in Part B of the Safety Evaluation Check List:

See the attachment  
 \_\_\_\_\_  
 \_\_\_\_\_  
 \_\_\_\_\_

(1) Reference to document(s) containing written safety evaluation:  
NS-SAT-SAI-88-362

FOR FSAR UPDATE

Section: \_\_\_\_\_ Page(s): \_\_\_\_\_ Table(s): 15.4.1-9

Reason for/Description of Change:

Change Table 15.4.1-9 for UHI Accumulator water volume delivered to reflect 850 cu.ft. minimum volume evaluated in this safety evaluation and the associated footnote.

6) APPROVAL LADDER

- (6.1) Prepared by (Nuclear Safety): RM Kenner (SAI) Date: 9/14/88
- Reviewed by (Nuclear Safety): M. A. Emery (SAI) Date: 9/14/88
- (6.2) Coordinated with Engineer(s): NO REVIEW (SAII) Date: \_\_\_\_\_
- Coordinated with Engineer(s): NECESSARY (TSA) Date: \_\_\_\_\_
- Coordinated with Engineer(s): PREVIOUS AP-(COA) Date: \_\_\_\_\_
- Coordinated with Engineer(s): PROVAL STILL (SAI) Date: \_\_\_\_\_
- (6.3) Coordinating Group Manager(s): APPLIES SINCE (SAII) Date: \_\_\_\_\_
- Coordinating Group Manager(s): ONLY LARGE (TSA) Date: \_\_\_\_\_
- Coordinating Group Manager(s): SMALL LOCAL COATS Date: CHANGED
- (6.4) Nuclear Safety Group Manager: Walter J. Tumble (SAI) Date: 9/14/88