Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37379-2000

April 1, 1997

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) - SUPPLEMENTAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION - TECHNICAL SPECIFICATION (TS) CHANGE REQUEST 96-01 ON CONVERSION TO FRAMATOME COGEMA FUEL (TAC NOS. M95144 AND M95145)

Reference: 1. TVA letter to NRC dated February 7, 1997, on the above subject

2. TVA letter to NRC dated March 17, 1997, on the above subject

3. TVA letter to NRC dated March 20, 1997, on the above subject

The purpose of this letter is to provide supplemental information to NRC for the TVA responses to NRC's request for additional information in the referenced letters.

Reference 1 provided responses to 36 NRC questions. NRC determined that no additional information to these responses was necessary with the exception of the responses to Questions 1, 2, 7, 8, 11, 12, 15, 20, 23, 25, 27, 28, 30, 31, 32, and 33. Reference 2 provided revised responses to each of the above questions with the exception of Question 1. Reference 3 provided the revised response to Question 1. Enclosure 1 provides revised responses to Questions 1, 7, 15, and 20 that were requested by NRC during telephone calls on March 21, 24, and 25, 1997.

Enclosure 2 provides the typed TS pages for TS Change 96-01 and has incorporated the changes discussed in Revisions 1 and 2 of the TS change.

Enclosure 3 contains the commitment made in this letter.

Please direct questions concerning this issue to Keith Weller at (423) 843-7527.

Sincerely,

R. H. Shell

R. H. Shell
Manager
SQN Site Licensing

Enclosures
cc: See page 3

cc (Enclosures):

Mr. R. W. Hernan, Project Manager
Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2739

NRC Resident Inspector
Sequoyah Nuclear Plant
2600 Igou Ferry Road
Soddy-Daisy, Tennessee 37379-3624

Regional Administrator
U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323-2711

ENCLOSURE 1

REVISED RESPONSES TO NRC QUESTIONS 1, 7, 15, AND 20

The following supplies supplemental information for the response to Question 1.

Response

The small break LOCA noding diagram submitted with the response to Question 1 was incorrect in its depiction of the break and the ECCS injection for the broken cold leg. The attached figure replaces that diagram. The artificial break node is shown as node 276 and the ECCS lines for the broken cold leg are shown to be entering node 276.

The ECCS injection for the broken cold leg is modeled in accordance with Section 4.3.2.2 of FTI(BWNT) small break LOCA evaluation model, Volume II of BAW-10168P-A (Reference 1). For the pump discharge break spectrum (PDB spectrum), node 276, the artificial break node, is located below the RCS piping with a downward orientation. This orientation assures that liquid resident in 276 will not flow into the pump discharge piping. Thus, when emergency core coolant (ECC) is injected into 276, it will flow out of the break and not be available to replenish the reactor vessel inventory.

The emergency core cooling system (ECCS) for Sequoyah comprises four injection systems: the high pressure, pumped centrifugal charging injection system (CCI); the intermediate pressure, pumped safety injection system (SI), the passive accumulator tanks; and the low pressure, pumped residual heat removal system (RHR). The FTI evaluation model[1], Section 4.3.2.2, requires that the broken loop ECCS flow be modeled as injecting into the artificial break node, node 276, after loop seal clearing. Since the artificial break volume is relatively small, 14.6 ft^3 for Sequoyah, the evaluation model allows that some portion of the broken loop ECCS flow can be placed directly in the RCS piping prior to loop seal clearing to prevent potential water hammer in the artificial break node.

For the Sequoyah analysis, all of the ECC system flow injected into the broken loop piping is modeled as flowing into node 276 after loop seal clearing has occurred. Prior to loop seal clearing, the CCI flow is modeled as entering the pump discharge piping, node 270; the SI, accumulator,

and RHR are modeled as entering node 276 both before and after loop seal clearing.

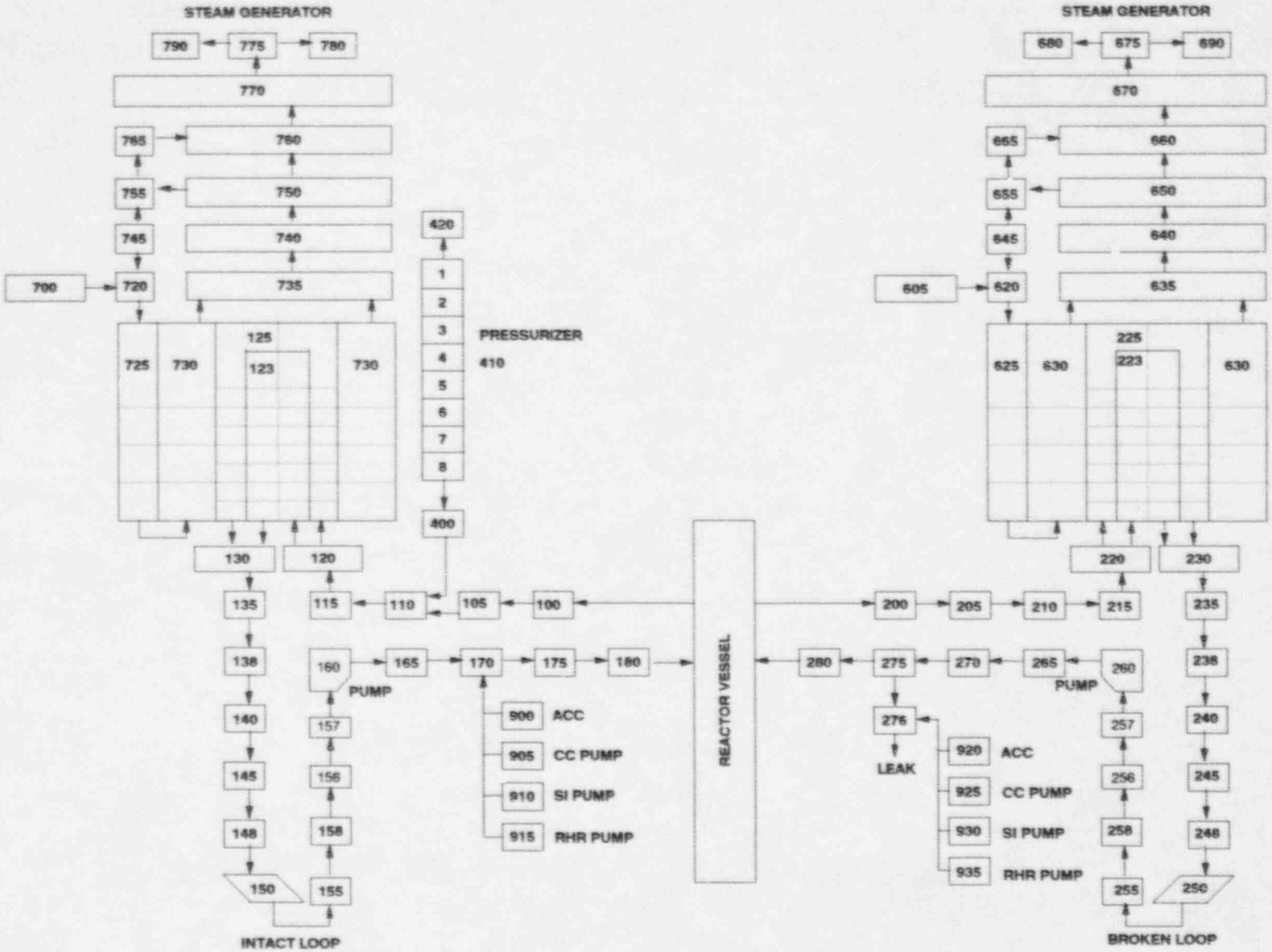
Node 276 is modeled as a nonequilibrium control volume. The break flow path, labeled LEAK on the diagram, is modeled as homogeneous. The containment volume area is set equal to the volume area of the break node to eliminate momentum flux gradients during periods of non-critical flow. The flow area for the path connecting the pump discharge piping to the break node, node 275 to node 276, has an area of one third of the area of the RCS piping. All of these selections are in accordance with the approved FTI small break LOCA evaluation model.

To assure that the problems mentioned in the response to Question 8 were no longer occurring and that the break flow during saturated discharge was being controlled by the Moody Critical Flow Model, several hand calculations of the break flow were made for the spectrum cases and compared to the RELAP5/MOD2-B&W output. At least one check was made for each of the seven cases submitted. The transient time at which the checks were done was varied to examine different portions of the transients. In all cases, the break flow rate calculated by RELAP5/MOD2-B&W matched the independent calculation of the Moody critical mass flux. This confirms that the code, under the homogeneous option for the leak path, conforms to the requirements of the evaluation model and to those of 10CFR50.46 and 10CFR50 Appendix K.

Reference:

1. BAW-10168P-A, Revision 3, "RSG LOCA BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," Volume II - Small Breaks, Framatome Technologies Incorporated, Lynchburg, Virginia, December 1996.

FIGURE 5.9-2. RELAP5/2 SBLOCA LOCA MODEL
Sequoyah Noding for Primary Loops with Model 51 SGs



Question 7 additional information on mixed core penalty

The LOCA mixed core evaluation performed by FTC for Sequoyah is based on the following:

1. There exists a valid set of LOCA calculations for each of the fuel assembly types that may be loaded into the Sequoyah core and each of these sets of calculations presumes that the particular fuel assembly type comprises the entire core. The peak cladding temperatures from these calculations may be, in fact probably are, indexed by the originator. Indexing has been widely employed by Westinghouse to correct for potential or real problems identified with calculations. Indexing does not invalidate a calculational set it merely adjusts the peak cladding temperature result by the amount of the index.

2. The only significant difference between the fuel designs involved in the mixed core configuration that would lead to a possible mixed core LOCA effect is the pressure drop or flow resistance of the assembly.

Sections 5.10 and 5.11 of BAW-10220 address these conditions. The text of Section 5.10 along with the comparisons offered in Table 5.10-1 clearly establish the pressure drop across the assemblies as the only design difference capable of mixed core interactions. However, the existence of appropriate, non-mixed core, LOCA calculations for the Westinghouse Standard and Vantage 5H assemblies was not clearly identified.

Sequoyah first loaded Westinghouse Vantage 5H fuel assemblies in the Spring of 1990. At that reload, two sets of LOCA calculations existed; one for the Standard design and one for Vantage 5H. A comparison between the calculations for the Standard and Vantage 5H designs, for the same assembly power and distribution, showed a deviation of approximately 100 F with the Vantage 5H having the lower peak cladding temperature (PCT). Further, a small mixed core effect of less than a 20 F increase in PCT was assessed for the Vantage 5H assemblies (Reference 1). A similar but negative effect would be appropriate for the Standard assemblies. TVA, with Westinghouse's concurrence, decided to apply the Standard assembly LOCA calculations and the resulting peaking limitations for the licensing of both the Vantage 5H and the Standard designs (Reference 2). This precluded the need for direct referral to a mixed core effect because the applied peak cladding temperature (PCT) remained at least 80 F conservative (100 F - 20 F).

This licensing approach remained in effect until 1995. In 1995, TVA replaced the Standard assembly LOCA calculations

with the Vantage 5H LOCA calculations with no mixed core provision. Because the core was, by this time, essentially all Vantage 5H assemblies with only a few Standard assemblies present, the maintenance of the mixed core ΔPCT and the use of the Standard assembly based LOCA calculations no longer made sense. LOCA coverage for the few Standard assemblies that would remain in the core was based on the low energy potential for those assemblies. All Standard assemblies at the Sequoyah plant had, at that time, experienced at least two cycles of operation and no longer retained the ability to operate at or approach limiting power conditions. Thus, when loaded, there is no possibility that these assemblies will be limiting with respect to LOCA. To provide further assurance, the fuel management procedures contain a proscription based on the assembly average power in that no PCT indexing is required for Standard assemblies if the bundle average radial peak is less than 1.28 for the cycle loading pattern. If a Standard assembly should exceed that limit, a +100 F indexing would be applied.

For the 1997 reload, Sequoyah is adding Mark-BW fuel assemblies supplied by Framatome Cogema Fuels (FCF) and reducing the thermal design flow to 348,000 gpm. For this reload, three LOCA calculational sets are required: one for Westinghouse Standard fuel, one for Westinghouse Vantage 5H fuel, and one for FCF's Mark-BW fuel. Calculations for the Standard and the Vantage 5H exist but were performed at the older thermal design flow of 362,000 gpm. To apply these calculations at the new thermal design flow, TVA requested that Westinghouse determine an appropriate index (ΔPCT) for the shift in thermal design flow. The evaluation conducted by Westinghouse showed that there would be no change in calculated PCT for the four percent reduction in system flow. Thus, the existing calculational sets for the Standard and the Vantage 5H designs apply directly to this reload. The calculational set for the Mark-BW fuel was originally done for this reload and used the reduced thermal design flow of 348,000 gpm. All of these calculations presume a full core of the reference fuel design. The applicable sets are: for the Standard fuel, the calculations for the Vantage 5H fuel plus appropriate indexes (In accordance with the limitation outlined in the above paragraph, the cladding temperatures will be indexed by 100 F if the bundle average radial peak is 1.28 or greater.); for the Vantage 5H fuel, the Vantage 5H calculations plus the appropriate indexing; and for the Mark-BW, the new calculational set performed by FCF. The current calculated large break LOCA PCTs of record for Sequoyah are: 1911 F for the Standard assembly and 1911 F for the Vantage 5H assembly. These values were reported to the NRC in the June 26, 1996 Annual 10CFR50.46 Report. The PCT for the Mark-BW assembly is 2115 F.

The mixed core effects are obtained from an FCF evaluation that considers all three co-resident designs. This evaluation has shown that the mixed core configuration has only a small effect on the results of LOCA calculations such that the full core calculations, with some indexing, are sufficient for the licensing of transition cores.

As a result of this reload, the mixed configuration of the Sequoyah core over the next several cycles will comprise:

Cycle	Mark-BW assemblies	Vantage 5H assemblies	Standard assemblies
1st	≅ 1/3 of core	≅ 2/3 of core	small number of assemblies
2nd	≅ 2/3 of core	≅ 1/3 of core	small number of assemblies
3rd & on	≅ full core	small number of assemblies	small number of assemblies

The evaluation of these configurations on the reference LOCA calculations was described in Section 5.10 of BAW-10220 but it may not have been clear in its applicability to all resident assembly designs. The evaluation of the pressure drop effect proceeds from consideration of the hot channel simulations during the blowdown and refill phases of the LBLOCA calculations. Both the Westinghouse and Framatome evaluation models separate the calculation of the hydraulics of the hot and average channels during blowdown and compute both. During refill and reflood both evaluation models compute the fluid flow in the hot channel from the average core flooding rate. This essentially means that the mass flux in the hot channel for the reflooding phase is determined by an average core calculation. Only the thermal effects within the hot pins are determined by hot channel-specific calculations during reflooding.

The expected mixed core LOCA effect can be determined by comparing the pressure drop within the average channel for a full core of the reference design to that of a mixed core. If the average core for the mixed configuration would develop a higher pressure drop than that appropriate for a full core of the reference design, two things will happen: some flow will be diverted into the hot channel during blowdown, lowering the cladding temperature slightly; and the refill rate (dependent on the more resistive average channel) will be slightly slower, raising the cladding temperature slightly. In this analysis, it makes no difference how the average channel became more resistive, one alternate fuel design or multiple alternate fuel designs, it is only important that as an aggregate it is more resistive. If the opposite is true, the average channel is less resistive than the full implementation of the reference fuel design, flow will be diverted from the hot channel during blowdown and the cladding temperature increased slightly, but the reflooding rate will increase slightly which will decrease the temperature somewhat. In either case, a swing in cladding temperature during blowdown will be compensated for by a change in the opposite direction during reflooding.

The configuration for Sequoyah for each of the fuel assemblies will be as shown in the following table.

If the Hot Channel Fuel is:	The Average Channel Is Made Up Of:	The Affect on Average Channel Resistance relative to Reference Calculation Is:
Westinghouse Standard Fuel	Mix of Mark-BW and Vantage 5H, some Standard	Average channel resistance increases above reference calculations
Westinghouse Vantage 5H Fuel	Mix of Mark-BW and Vantage 5H, some Standard	Average channel resistance decreases below reference calculations
Mark-BW Fuel	Mix of Mark-BW and Vantage 5H, some Standard	Average channel resistance increases above reference calculations

FCF has calculated the effect for a representative core in which the average core resistivity was increased above the reference calculations when Mark-BW fuel assemblies replaced Westinghouse OFA fuel assemblies at McGuire and Catawba (Reference 3). The pressure drop for the OFA assemblies was measured to be 1 psi higher than the Mark-BW assemblies. This is the same as the Sequoyah configuration where the Vantage 5H assemblies also have a pressure drop that is approximately a 1 psi higher than that of the Mark-BW. The difference in cladding and fuel pellet average temperatures, for the Mark-BW assembly in mixed core operation when compared to a full core loading of Mark-BW assemblies, was a decrease of 30 to 50 F at the end of blowdown. This was caused by the diversion of some liquid from the more resistive average channel to the hot assembly. As expected, the temperature rise during reflood increased due to the slower flooding rate. The increase was 30 F. Because the peak cladding temperature occurs during reflood, the differences in temperature rises are additive and the net change was small decrease in PCT, less than 20 F. This decrease in temperature is appropriate for the Mark-BW assembly during mixed core operation with the Vantage 5H design.

For the Vantage 5H fuel assembly, the average channel of the mixed core will have a lower resistance (1/3 of the core will be comprised of the 1 psi less resistant Mark-BW). Here, the opposite results would occur. Some flow would divert away from the hot assembly during blowdown and the blowdown temperature in the hot channel would increase by 30 to 50 F. The temperature rise during reflood would then decrease by 30 degrees or so. The cumulative result would be a potential increase in PCT of less than 20 F for the Vantage 5H fuel.

The Standard assembly would evaluate much like the Mark-BW. Because it has a lower resistance than the Mark-BW, it would experience an even larger drop in PCT. Additionally, these

assemblies will not be prevalent in the core, only a few may be used in each cycle, and all, having experienced 2 cycles of operation prior to this reload, have very low energy potential. They can not become the LOCA limiting assemblies in the Sequoyah core.

For each of the fuel designs to be used in the Sequoyah core the mixed core effects are presented in the following table.

Mixed Core Effects per Assumed Limiting Assembly

Hot Channel Fuel Design	Fuel Designs in Average Channel	Affect on Average Channel Resistance relative to Reference Calculation	Potential Change in Peak Cladding Temperature
Westinghouse Standard Fuel	Mix of Mark-BW and Vantage 5H, some Standard	Average channel resistance increases	PCT decreases by less than 20 F
Westinghouse Vantage 5H Fuel	Mix of Mark-BW and Vantage 5H, some Standard	Average channel resistance decreases	PCT increases by less than 20 F
Mark-BW Fuel	Mix of Mark-BW and Vantage 5H, some Standard	Average channel resistance increases	PCT decreases by less than 20 F

In accordance with the depletion of energy production capability for twice burned fuel, fuel assemblies in their third cycle of operation can not comprise the assemblies within which the LOCA PCT will occur. Therefore, no mixed core considerations will be applied to these fuel assemblies. The Mark-BW, which benefits in the mixed core configuration, will be licensed with its full core reference calculations. The Vantage 5H assemblies, because of the potential for a slight increase in PCT, will have a +20 F PCT index applied to the reference full core calculation results for all assemblies that have not gone through 2 cycles of operation. The indexing of the Vantage 5H assemblies will be removed for their third cycle of operation. With the ΔPCT applied to the Vantage 5H fuel the fully indexed PCTs for each fuel type are: 1911 F for the Standard assembly, 1931 F for the Vantage 5H assembly, and 2115 F for the Mark-BW.

In light of the discussion provided, two specific statements in BAW-10220 need correcting. In Section 5.10, page 5-100, the statement is made, "Additionally, the relative average power limitation assessed by Westinghouse for side-by-side operation of Vantage 5H and Standard fuel assemblies is unaffected by this evaluation and continues to be applicable to those fuel types." This statement does not reflect the fact that the Standard design assemblies are all in their third cycle of operation and can not comprise the assembly of LOCA PCT. It is not correct and no side-by-side limitation will be applied to these assemblies. Further, the Vantage 5H assemblies will have a +20 F indexing applied

but that is caused by their side by side operation with the Mark-BW assemblies. The sentence should be removed from the topical.

A similar statement, in Section 5.11, page 5-102, is: "Moreover, PCT penalties assessed by Westinghouse for the mixed core operation of Vantage 5H and Standard fuels, remain applicable to those fuel types." This statement is incorrect. The only ΔPCT necessary will be applied to the Vantage 5H LOCA calculations. The sentence should be disregarded in reading the topical.

References:

1. Addendum 2-A of WCAP-10444-P-A or Addendum 2 of WACP-10445-NP-A,S.L. Davidson, Vantage 5H Fuel Assembly, Westinghouse Electric Corporation, April, 1988. Note, the main body of this report is appropriate for the Vantage 5 design and may not be directly applicable to Sequoyah. Addendum 2, however, is specifically for the Vantage 5H design.

2. B.W. Gergos and L.V. Tomasic, THFL-89-762, "Plant Safety Evaluation for Sequoyah Nuclear Plant Units 1 and 2 Vantage 5H Fuel Upgrade, December 1989, Westinghouse Electric Corp.

3. BAW-10174A revision 1, Mark-BW Reload LOCA Analysis for Catawba and McGuire, Babcock & Wilcox/Framatome Technologies Inc., November 1990.

15. This request is in addition to the original question 15 regarding the Sequoyah safety evaluation for reduced thermal design flow.

Arguments made in defense of DNB acceptance criteria for the feedwater malfunction (4.0.2.10), excessive load increase (4.0.2.11), and feedwater line break (4.0.4.3) stated that margin to DNB was assured because there was no trip on overtemperature ΔT or overpressure ΔT for these events. Please clarify these arguments.

Response:

These events generally result in a change in and the establishment of an increased power level. The extent to which power is changed is not sufficient to produce a reactor trip on either overtemperature ΔT or overpressure ΔT. The transient, if it produces a reactor trip, produces a trip that is not affected by a reduction in thermal design flow.

MAP Limits have been developed at operational statepoints corresponding to overtemperature ΔT or overpressure ΔT boundaries that indicate an adequate margin to DNB. Since the transients do not trip on the overtemperature ΔT or overpressure ΔT trip, they are within the MAP boundary. Margin to DNB is, therefore, assured.

Revisions to the relevant pages of the safety evaluation are attached which clarify the overtemperature ΔT - overpressure ΔT arguments.

4.0.2.10 Excessive Heat Removal Due to Feedwater System Malfunction

Excessive feedwater flow could be caused by a full opening of one or more feedwater regulator valves due to a feedwater control system malfunction or an operator error. At power this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generators. At no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator coefficient of reactivity. Overcooling for the event is mitigated by the steam generator high-high level trip, which closes all main feedwater isolation valves, trips the main feedwater pumps, and trips the turbine.

The excessive heat removal due to feedwater system malfunction event is a Condition II event and has the following acceptance criteria:

1. Peak primary and secondary system pressure shall not exceed 100% of design value.

2. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for the correlation used.

Prior to the high-high steam generator water level signal, both the RCS and steam generator secondary are cooled by the excessive feedwater and the respective system design pressures are not challenged. Turbine trip occurs first and an imbalance in heat production/removal is indicated in the UFSAR analysis for this event that is associated with a 2.5 second delay between the turbine trip on level and the reactor trip (Table 15.2-1). The delay between turbine trip and reactor trip for the LOEL transient is on the order of 6 to 8 seconds. The LOEL is, therefore, bounding in system pressure response.

The LOEL event was analyzed in Reference 1, Section 6.2.7, in support of FCF fuel loading at Sequoyah. The analysis utilized a reduced RCS thermal design. System pressures do not exceed the design limits subsequent to an LOEL.

The analysis of feedwater malfunction event is presented in Section 15.2.10 of the Sequoyah UFSAR. Main feedwater addition to the steam generators is increased by a conservatively prescribed amount to characterize feedwater regulator valve failure. A reduction in RCS thermal design flow does not affect the initial RCS average temperature or the reactivity feedback parameters. The RCS cooling, and the increase power that results, is proportional to the increase in feedwater flow and is independent of RCS flow. Because the thermal design flow is being reduced, however,

the calculated margin to DNB for this event could be reduced.

The excessive feedwater event results in an increase in core power and the establishment of a new stable power level prior to reactor trip. The event is mitigated via reactor trip and feedwater isolation on high - high steam generator level, independent of thermal design flow. MAP limits have been developed at operational statepoints associated with overtemperature and overpower ΔT trip boundaries and verify that adequate DNB margins exist at these statepoints. Since, for this transient, the resulting increase in power is not high enough to cause an overtemperature or overpower ΔT trip, there is no challenge to the fuel thermal limits. Consequently, with respect to minimum DNBR, the excessive feedwater event is bounded by the rod withdrawal at power event.
The RCCA withdrawal at power event was analyzed in Reference 1, Section 6.2.2, in support of FCF fuel loading at Sequoyah. The analysis utilized a reduced RCS thermal design flow. An adequate DNB margin was demonstrated for the RCCA withdrawal at power event.

Conclusion

The system pressure response of the excessive feedwater event is bounded by LOEL. The reanalysis of Reference 1 demonstrated the system pressure acceptance criterion is met for an LOEL. With respect to minimum DNBR, the excessive feedwater event is bounded by an RCCA withdrawal at power. The reanalysis in reference 1 indicates a sufficient margin to DNB for an RCCA withdrawal at power. It is assured that all of the acceptance criteria for the excessive feedwater event are successfully met for operation of Sequoyah with a thermal design flow of 348,000 gpm.

4.0.2.11 Excessive Load Increase Accident

An excessive load increase incident is defined as a rapid increase in steam generator steam flow that causes a mismatch between the reactor power and the steam generator heat removal. The resultant cooling of the reactor primary system fluid causes an increase in reactor power due to a negative end-of-cycle moderator temperature coefficient. It also causes a reduction in primary system pressure due to the contraction of the reactor coolant. The increase in power and decrease in primary system pressure produce a reduction in DNBR.

The excessive load increase event is a Condition II event and has the following acceptance criteria:

1. Peak primary and secondary system pressure shall not exceed 110% of design value.

2. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for the correlation used.

An excessive load increase causes an overcooling and depressurization of both the RCS and steam generator secondary systems. The respective system design pressures are not challenged. The LOEL, therefore, bounds this event in system pressure response.

The LOEL event was reanalyzed in Reference 1, Section 6.2.7, in support of FCF fuel loading at Sequoyah. The reanalysis utilized a reduced RCS thermal design. System pressures do not exceed the design limits subsequent to an LOEL.

UFSAR Section 15.2.11 (Reference 2) reports the analysis of the excessive load increase. The excessive load increase event results in an increase in core power and the establishment of a new stable power level. The analysis results demonstrate that a reactor trip on overtemperature ΔT, overpower ΔT, or high nuclear flux does not occur for this transient. MAP limits have been developed at operational statepoints associated with overtemperature and overpower ΔT trip boundaries and verify that adequate DNB margins exist at these statepoints. Since, for this transient, the resulting increase in power is not high enough to cause an overtemperature or overpower ΔT trip, there is no challenge to the fuel thermal limits. Correspondingly, an adequate margin to DNB is assured.

A reduction in thermal design flow can only affect the rate at which the primary system responds to an increase load. The initial core power, RCS average temperature, and reactivity feedback parameters are unchanged. The increase

in steam flow (to 110 percent) for this transient is a fixed, prescribed, input. The ultimate primary cooldown and core power levels would be identical, therefore, at the reduced RCS flow. As a result, the reactor would not trip and there would be no challenge to the fuel thermal limits for the excessive load increase event.

Conclusion

The system pressure response of the excessive load increase event is bounded by an LOEL. The reanalysis of Reference 1 demonstrates that the system pressure acceptance criterion is met for an LOEL. The reference analysis demonstrates that a reactor trip does not occur for a 10 percent load increase. A reduction in RCS flow would not affect this result. A margin to both overtemperature ΔT and DNB is, therefore assured. The acceptance criteria for the excessive load increase event are successfully met for operation of Sequoy h with a thermal design flow of 348,000 gpm.

4.0.4.3 Major Rupture of a Main Feedwater Pipe

A major rupture of a main feedwater line represents a rapid decrease in heat removal capability of the secondary system because it reduces the supply of feedwater to the steam generators. Main feedwater is, in fact, assumed to be lost to all of the steam generators at the time of rupture. Reverse blowdown of the affected steam generator results in a relatively rapid reactor trip signal on low-low level in that generator.

Following reactor trip, the main feedwater line break is characterized by excess heat removal from and cooldown system as the affected steam generator continues to blow down through the break. When the low steam line pressure setpoint is reached the main steam isolation valves are closed and safety injection to the RCS is initiated. The loss of steam generator inventory and rising pressure steam pressure cause primary temperatures to rise. Successful termination of the transient is achieved when the auxiliary feedwater supplied to the steam generators is sufficient to remove core decay heat.

The main feedwater line break is a Condition IV event, subject to the following acceptance criteria:

1. Peak primary and secondary system pressures shall not exceed 110% of the design pressures.

2. Fuel clad integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for the correlation used.

3. Liquid in the RCS shall be sufficient to cover the reactor core at all times.

The main feedwater line break event results in a depressurization of the steam generator secondary. Isolation of the steam lines occurs on low steam pressure and is delayed following reactor trip. In comparison, the LOEL is caused by an abrupt loss in steam flow before the reactor trip. The secondary heat load is, therefore, comparatively higher for the LOEL event. The secondary pressurization is, therefore, bounded by the LOEL transient response. This relationship is unaffected by a reduction in thermal design flow.

The primary pressure response during a LOEL bounds that for the main feedwater line break because the reactor trip is delayed with respect to the LOEL. In contrast, reactor trip and turbine trip occur at the same time on low-low steam generator level in the main feedwater line break, resulting

in a lower peak pressure as compared with the LOEL event. This relationship is unaffected by RCS flow rate.

LOEL was analyzed in support of FCF fuel reload at Sequoyah (Section 6.2.7 of Reference 1). The analysis assumed a reduced thermal design flow, and acceptable margin to the primary pressure acceptance criteria was demonstrated.

Ultimately, the feedwater line break event is mitigated via reactor trip and feedwater isolation on a low-low steam generator level. This result is independent of thermal design flow. MAP limits were developed at operational statepoints associated with overtemperature and overpower ΔT trip boundaries and verify that adequate DNB margins exist at these statepoints. Since the transient does not trip on overtemperature or overpower ΔT, there is no challenge to the fuel thermal limits. No DNB occurs and the feedwater line break is, therefore, bounded by the rod withdrawal at power event which relies on the overtemperature ΔT for protection.

The uncontrolled rod withdrawal at power event was analyzed in Reference 1 in support of Framatome Cogema Fuels fuel loading at Sequoyah (Section 6.2.2). The analysis utilized a reduced RCS thermal design flow and demonstrated that adequate DNB margin is retained.

The feedwater line break is also analyzed in the UFSAR to demonstrate that long-term cooling is maintained. Long-term decay heat removal capability is related to secondary side liquid inventories and auxiliary feedwater flow. These parameters are unaffected by RCS flow rate.

Conclusion

The feedwater line break event is bounded, in pressure, by the LOEL event. LOEL was analyzed with reduced thermal design flow and system pressures were demonstrated to remain within 110% of design values. There is no challenge to DNB for this event as the reactor trips prior to the approach to the safety limits. Furthermore, with respect to DNB, the feedwater line break is bounded by the rod withdrawal at power event. The rod withdrawal at power event was analyzed with a reduced thermal design flow and margin to DNB was demonstrated. Long-term decay heat removal is related to steam generator secondary liquid inventory and auxiliary feedwater flow capacity. These are not affected by primary system flow, and the existing UFSAR analysis remains bounding. Therefore, it is assured that all of the acceptance criteria for this event are successfully met for operation of Sequoyah with a thermal design flow of 348,000 gpm.

20. For the main steam line break analysis (p. 6-54), the volume of water between the cold leg piping and the first check valve is considered to be at 0 ppm boron concentration. The current Final Safety Analysis Report (FSAR) analysis assumes the volume of water in the piping from the RWST to the cold leg piping (a much bigger volume) is all at 0 ppm. Please describe how you can assure that the water in the piping between the RWST and the first check valve is at least 1950 ppm.

Response

For the FCF main steam line break analysis it is assumed that the coolant in the four safety injection lines downstream of the *second* check valve removed from the cold leg, has a boron concentration of 0 ppm, not the *first* set which is incorrectly alluded to on page 6-54. Modeling an unborated purge volume accounts for possible dilution via the diffusion of RCS coolant back into the lines plus potential leakage of the first set of check valves. Once the unborated water is purged, boron is assumed to be injected into the RCS at a concentration of 1950 ppm via a single intermediate head pump. This is a reasonable method of modeling post-accident boron injection in that it both is conservative and reflective of the actual plant configuration.

Even though boron injection is modeled for the analysis of steam line break, the return to power associated with the limiting break is not mitigated by boron. The limiting break, a complete severance of the steam line upstream of the pipe flow reducer with offsite power available, is reported in Section 6.4.1 of the reload topical report. The results clearly show that the core power turns over prior to the injection of boron. Core power is controlled instead by the dry-out of the broken steam generator and the subsequent cessation of both RCS cooling and positive reactivity feedback.

Subsequent to the original FCF analysis, sensitivity studies have been performed related to the assumption of boron delivery to the RCS for the main steam line break event. The sensitivity studies examine the extent that leakage/diffusion of RCS coolant can occur in the safety injection lines before the results of the limiting case are adversely affected. The studies show that coolant with a boron concentration equivalent to that of the End of Life RCS (0 ppm) can exist in the safety injection system piping back as far as three check valves removed from the cold leg up to the discharge of the safety injection pump (see Figure 20-1). A boron concentration of 1950 ppm is assumed from the RWST to the discharge of the SI pumps with 0 ppm from the pump discharge to the RCS. Based on these conditions,

the limiting break (a complete severance of the steam line upstream of the pipe flow reducer with offsite power available), as reported in Section 6.4.1 of the reload topical remains valid and limiting. The sensitivity studies performed prove that the original FCF analysis remains bounding and the return to power associated with the other non-limiting steam line break cases will also remain less than the return to power associated with the limiting case assuming 0ppm boron back to the discharge of the SI pump.

Boron injection is conservatively modeled in the FCF analysis of the main steam line break for Sequoyah, even though the assumption does not match the assumption of the original FSAR analysis in that:

- a conservative volume of unborated liquid is assumed to be purged prior to the injection of borated water into the RCS

The boron injection assumption also reflects the physical arrangement of the plant in that:

- the SI system is isolated from the unborated charging lines when the charging system is aligned for normal makeup
- the SI system always draws suction from the RWST during normal operation, which is borated to a minimum of 2500 ppm
- check valve leakage, which could potentially dilute boron in the SI lines, is detectable via pressurization as a result of leakage back into liquid-solid lines

Again, the limiting steam line break presented in the Sequoyah reload topical report does not require the addition of boron for the mitigation of the core power transient for this event.

The SI pump casing is vented every 31 days per Technical Specification 4.5.2.b.1. This in turn causes the upstream piping back to the RWST to be vented. Venting occurs until a steady stream of water is noted at the pump casing vent. The pressure source is the head difference between the RWST and the SI pump. The RWST is verified to have at least 2500 ppm boron every 7 days per Technical Specification 3.5.5. The piping downstream of the SI pump is also vented for the hot and cold leg injection paths. The procedure requires that the flowpath through the SI pump, the pump discharge check valve and each test header isolation valve (63-21, 23, 121 and 167) be open for ten minutes, for a total of forty minutes of venting through the piping header. Again, the pressure source is the head difference between the RWST, the SI pump and the discharge piping (see Figure 20-1).

The Technical Specification minimum RWST volume is 375,000 gallons. The volume of water between the SI pumps supply check valve and the RWST tank entrance is approximately 4250 gallons. The approximate volume of water between the SI pump supply check valve and the discharge of one SI pump is 65 gallons. These are the volumes assumed to be at 1950 ppm of boron.

The approximate volume of water between one SI pump discharge and a typical secondary check valve inside containment is 160 gallons. The approximate volume of water between a typical secondary and primary check valve is 60 gallons. The approximate volume between a primary check valve and the hot/cold leg piping is 20 gallons. These are the volumes assumed to be at 0 ppm.

The venting procedures previously described for the downstream piping of the SI pump require venting each of the four paths for 10 minutes each. Based on an average flow rate of 10 gallons per minute for each vent path, a minimum of 400 gallons is displaced in the section of piping between the RWST and the secondary check valves inside containment. An additional volume is also displaced in this suction piping due to the venting of each SI pump casing. This displaced water is replaced with RWST inventory with a minimum of 2500 ppm boron every 31 days.

The system is a water solid closed system during operation. Any leakage past any of the check valves will result in the rapid pressurization of the piping volume upstream of the leaking check valve. This pressurization will only require a relatively small amount of liquid to equalize the pressure, thereby limiting the potential for the dilution of the contained volume. This condition is therefore self-limiting. Upstream of the SI pump suction but downstream of the common SI pump suction check valve, a code relief valve is installed (relief setpoint of 220 psi). Downstream of the SI pump discharge, three code relief valves (relief setpoint of 1750 psi) are installed, one on each injection flowpath. Continued leakage through multiple check valves would be required to create a leak path through any of the above relief valves. The following is an excerpt from the Sequoyah FSAR section 5.2.7.8, ECCS Intersystem leakage:

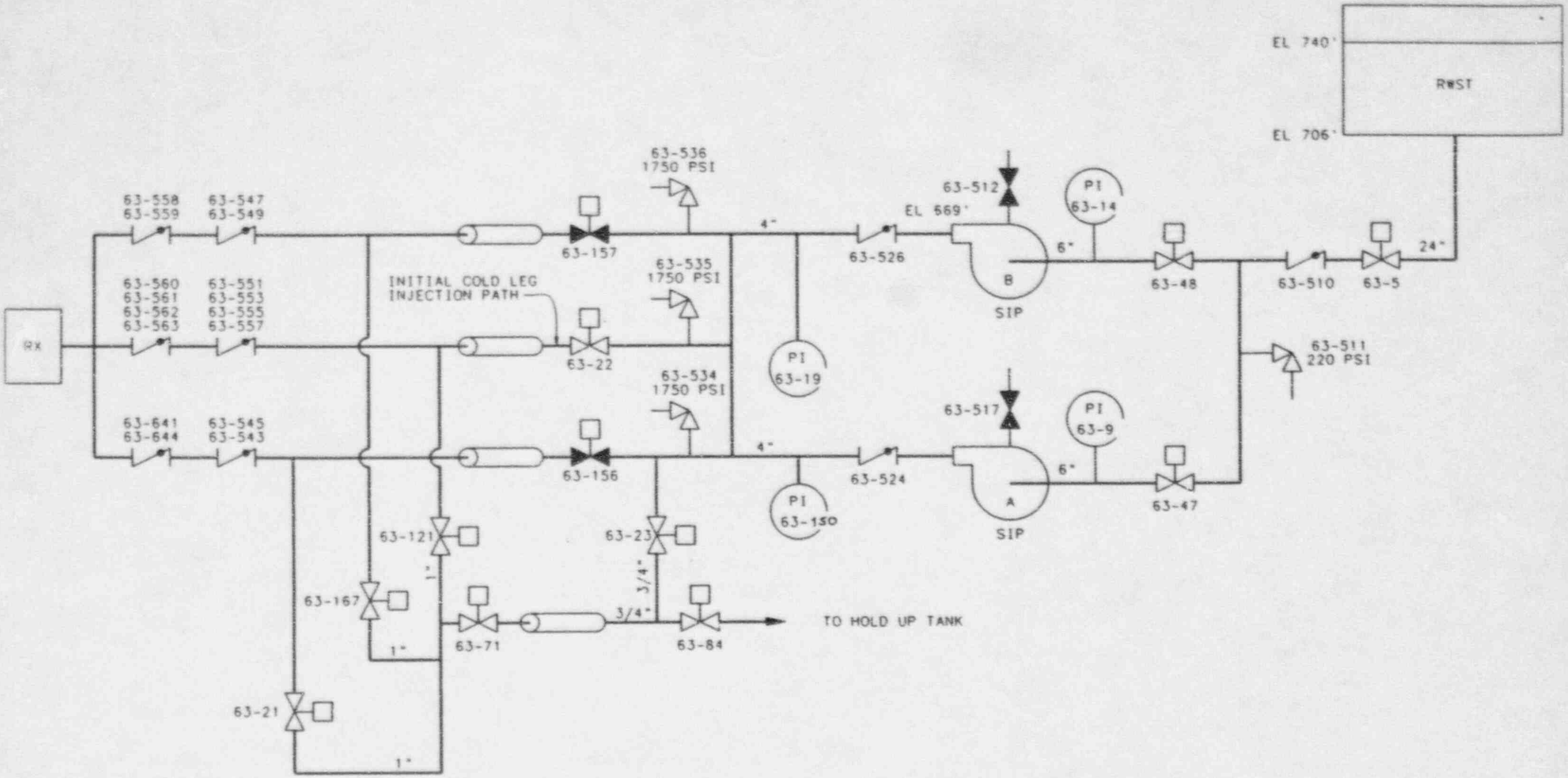
> *Leakage from the RC System into low pressure portions of several ECCS lines is prevented by the use of two check valves in series. The check valves are tested for leakage in accordance with the applicable surveillance instructions. The probability of a major leak through any pair of check valves will therefore be limited to approximately 5.5 x 10-9 per reactor year. This probability is low enough to eliminate any concern for a*

major intersystem leak into low pressure ECCS Systems. However, means are available to continuously monitor and alarm intersystem leakage across the interfaces between the RC System and the following: Cold Leg Accumulators (CLA), Chemical and Volume Control System (CVCS), Safety Injection System (SIS), and RHR System. Leakage into these systems can be detected both by monitoring for signs of incoming leakage and by monitoring the RC System for signs of outgoing leakage.

Intersystem leakage across the two check valves in each of the four SIS cold leg injection lines or across the two check valves and one normally closed gate valve in each of the four SIS hot leg injection paths would increase the pressure in those segments of the lines. A separate pressure sensor is provided in each of the two SIS pump discharge lines with indication continuously available in the MCR. The two pump discharge lines are connected with a normally open crossover line so a pressure increase in this segment would be detectable by either sensor. Three pressure relief valves are also provided for these SIS lines. When the pressure in the line reached 1750 lb/in2g, the relief valves would discharge a total of 60 gal/min to the pressurizer relief tank. Discharge into this tank would increase the tank level, pressure, and temperature. A level sensor is provided on the tank having both continuous indication and alarm available in the MCR.

In addition, verification by sample that a sufficient boron concentration exists in the SI pump suction piping will be performed if required by Abnormal Operating Procedure AOP-R.05, "RCS Leak and Leak Source Identification". This will ensure that a sufficient boron concentration exists in the suction piping if leakage past three check valves occurs.

Based on the system design and the Technical Specification required venting surveillance (with a 31 day frequency) of the SI pump and piping, it is technically acceptable to allow the assumption of a 1950 ppm boron concentration in the SI pump and suction piping for the main steam line break event. Assumptions regarding the modeling of boron injection for the main steam line break event (i.e., assumption of 0ppm from the reactor vessel to the discharge of the SI pump) will be clearly stated in revisions to the FSAR related to the Mark-BW fuel load at Sequoyah.



SIMPLIFIED FLOW PATH RWST/SIP/VESSEL

Figure 20-1

ENCLOSURE 2

PAGES FOR

TECHNICAL SPECIFICATION CHANGE 96-01