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Safety Evaluation of Safety Valve Setpoint Tolerance Relaxation

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Prepared for
Tennessee Valley Authority
Sequoyah Nuclear Plant, Units 1 & 2

by

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1. Purpose

Currently the Sequoyah Nuclear Plant Units 1 & 2 of the Tennessee Valley Authority (TVA) have a setpoint tolerance of ±1% on the code safety valves. A proposed change is to relax the tolerances on the pressurizer safety valves (PSVs) and main steam safety valves (MSSVs) lift setpoints.

The purpose of this document is to support a change in the tolerance limits on the setpoints of the safety valves of the Sequoyah Nuclear Plant Units 1 & 2. Each event in Chapter 15 of the FSAR is evaluated to assess the effect of the proposed change in the valve lift tolerances. The objective is to show that all of the acceptance criteria will be met with the proposed change. For those cases that are affected by the change in the tolerance limits, analysis is performed to quantify the impact. Analyses are presented in the Mark-BW Fuel Assembly Topical Report for the Sequoyah Nuclear Plant (Reference 1) that allow for the additional safety valve setpoint tolerances. For the events which are either bounded by these events or which are not adversely affected by the proposed changes, an evaluation is performed to ensure that the events will meet the relevant acceptance criteria.

2. Background

Framatome Cogema Fuels/Framatome Technologies Inc. (FCF/FTI) performed a series of safety analyses to support the fuel reload at Sequoyah. As part of the reload effort, the limiting transients were analyzed assuming as much as +5% tolerance limit on the PSVs and +3% on the MSSVs.

TVA desires to relax the tolerance limit at Sequoyah. The reload analyses (Reference 1) showed that a relaxation to +3% from the currently approved limit of +1% is possible. This effort augments these studies via a full examination of events analyzed for Chapter 15 of the FSAR (Reference 2).

Increased PSV negative tolerances do not affect safety margin. However, prudence dictates that the lifting of the PSV will not interfere with the operation of the non-safety power-operated relief valves. MSSV negative tolerance relaxation is predicated on an examination of current plant dose release analyses.

3.0 Scope of Evaluation

All of the Sequoyah FSAR Chapter 15 events are examined in this document relative to the proposed change in the safety valve setpoint tolerance requested. Events that were analyzed in support of the Sequoyah reload (Reference 1), already consider the relaxation in tolerance and provide a basis for discussing the effects of the setpoint change.

These evaluations are arranged in the order of the FSAR Chapter 15 discussions. The Condition I, II, III and IV events are grouped together in the evaluation under similar headings.

The positive tolerance on the PSVs and the MSSVs are validated by comparison of each FSAR event with the limiting analyses performed in the Reference 1 reload report. A change in the negative tolerance on the MSSV setpoints has the potential to increase the dose release to the atmosphere. The effect of a negative tolerance change on the MSSVs is addressed in a separate section in this document.

4.0 Safety Evaluation

4.1 Introduction

The safety evaluation presented here is divided into various segments. The methodology segment indicates the general approach taken in the evaluation. It also contains the list of the limiting analyses performed for the Sequoyah reload for the Sequoyah Nuclear Plant Units 1 & 2. The conclusions based on the results of the reanalyzed transient are stated. Sections 4.2.4, 4.2.5, 4.2.6, and 4.2.7 show an evaluation of the increase in the probability of occurrence and the consequences of the events due to the changes proposed. The effect of negative tolerance on the dose releases through the MSSVs is addressed in section 4.2.8.

All the events discussed in the Sequoyah FSAR (Reference 2) are evaluated to determine the effect of changing the tolerance on the setpoints of the safety valves. The objective of the evaluation is to show that the changes proposed are either inconsequential to the particular event or, if consequential, the result of the impact on the consequences of the event is bounded by the limiting events and hence the change will not adversely affect the consequences. The limiting events that have been analyzed with the proposed changes to the setpoint tolerances have shown that the criteria of acceptance for these events are satisfied with the proposed changes.

4.2 Review of the FSAR Events

4.2.1 Methodology

Events that were analyzed in support of the Sequoyah reload used relaxed safety valve setpoint tolerances. The remaining events from Chapter 15 of the FSAR that were not analyzed will be evaluated for the effect of the changed tolerance. Over-pressure protection and dose release are the primary parameters that will be affected by the setpoint relaxation. Overpressurization is sensitive to positive tolerance and the dose release to the negative. The effects of tolerance relaxation on dose release will be addressed in a separate section(4.2.8) at the end of this evaluation. The remaining effects will be evaluated for each transient, relative to the limiting events that have already been analyzed in support of fuel reload.

Parameters noted to be sensitive to the change will be identified on an event-by-event basis. A descriptive discussion regarding the nature of transient progression and the extent to which the transient is affected by relaxed safety valve setpoint tolerance

is also offered. Based on this discussion, conclusions are drawn regarding the relative effects on the evaluated transient.

A list of the reload transients that were reanalyzed with the relaxed tolerance is presented in Table 1. The analytical results of those transients are reported in the Topical Report for Mark-BW Fuel Assembly Application for Sequoyah Nuclear Units (Reference 1).

These events have been reanalyzed with the proposed (+3%) or higher tolerance and have been shown to meet all the safety acceptance criteria.

Table 1

Description of FSAR event reanalyzed	Analysis Tolerance on	
	PSVs	MSSVs
1. Uncontrolled Rod Cluster Assembly	+5%	+3%
2. Complete Loss of Reactor Coolant Flow	+5%	+3%
3. Locked Rotor Event	+5%	+3%
4. Loss of External Electric Load	+5%	+3%
5. Main Steam Line Rupture	+5%	+3%
6. Small Break LOCAs	+5%	+3%
7. Large Break LOCAs	+5%	+3%

4.2.2 Evaluation of Events for Probability of Occurrence

The safety valves have specific functions that they have to perform during an event. The general function of the safety valves is to limit the overpressurization of the systems during a transient. The events of the FSAR, Chapter 15, are initiated by failures not related to the setpoint tolerances of the primary and secondary safety valves. Hence a change in the tolerance of the setpoint of the valves does not increase the probability of occurrence of any of the events.

4.2.3 Evaluation of FSAR Accidents

An evaluation of each FSAR Chapter 15 accident is presented here. Those accidents not analyzed for FCF reload fuel are evaluated to show that all acceptance criteria will continue to be met with the proposed relaxation in safety valve setpoint tolerances. Those events that were analyzed with the propcsed safety valve setpoint tolerances (Table 1) are included for completeness.

4.2.3.1 Condition I Events

Condition I events are bounded by events that are classified as Condition II, III & IV since the severity of Condition I events is minor in comparison.

4.2.3.2 Condition II Events

4.2.3.2.1 Uncontrolled Rod Cluster Assembly Withdrawal from a Subcritical Condition

This event assumes reactivity addition by either boron dilution or inadvertent operator action that results in the pull of two control rod banks of maximum combined worth pulled at the maximum speed possible from subcritical condition. The startup of the reactor from subcritical condition with a clean core is performed by boron dilution and the maximum reactivity addition due to boron dilution is below the rate of reactivity addition assumed for this analysis. In this analysis, the reactor is assumed to be at hot zero power to maximize the negative effects of heat transfer from the fuel to the coolant. This event is DNB-limited. This event is protected by the low power neutron flux trip which is lower than the full power neutron trip setpoint.

During the rod pull, the reactor system pressure increases due to mismatch of energy removal with the power production, until the reactor trips. The RCS pressure peaks and starts declining due to heat removal catching up with the stored energy in the primary system.

The acceptance criteria for this event are:

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.

Because this transient initiates from hot zero power conditions and is promptly terminated by a reactor trip on high neutron flux at 35% power, peak primary and secondary pressures for this transient are bounded by the loss-of-electrical-load transient. The LOEL transient initiates from full power (102%) conditions and terminates after a delayed reactor trip on high pressurizer pressure. Since the pressure acceptance criteria are met for a LOEL with the elevated valve tolerances, the acceptance criteria for this event will be met with elevated valve tolerances.

The parameters that affect the DNB response for this event are the reactivity insertion assumed to simulate the rod pull, initial axial power distribution, moderator temperature coefficient and Doppler coefficient. The safety valve lift tolerances do not affect the outcome of this event.

This event has the potential of opening the secondary MSSVs and hence offsite dose release can occur. However, changes in the MSSV lift tolerance will not result in fuel pins in DNB for this event. Therefore, offsite dose is bounded by the loss of AC power event, evaluated in Section 4.2.8.

Conclusion:

The Uncontrolled Rod Cluster Withdrawal from Subcritical Condition can pass the criteria for acceptance for primary and secondary over-pressurization limits with +3% tolerance in the MSSV setpoints and +5% tolerance in the PSV setpoint.

4.2.3.2.2 Uncontrolled Rod Cluster Assembly Withdrawal at Power

Uncontrolled Rod Cluster Assembly Withdrawal at Power results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and the resultant coolant temperature rise could eventually result in DNB. Therefore, in order to avert damage to the fuel cladding, the reactor protection system is designed to terminate any such transient before the DNBR falls below the limit values.

The automatic features of the reactor protection system that prevents core damage following the postulated accident include the following:

1. Reactor trip is actuated if any two-out-of-four power range neutron flux instrumentation channels exceed an overpower setpoint.

2. Reactor trip is actuated if any two-out-of-four ΔT channels exceed an overtemperature OTΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature and pressure to protect against DNB.

3. Reactor trip is actuated if any two-out-of-four ΔT channels exceed an overpower ΔT (OPΔT) setpoint. This setpoint is

automatically varied with axial power imbalance to ensure that the allowable heat generation rate (kW/ft) is not exceeded.

4. A reactor trip is actuated if any two-out-of-four pressurizer pressure channels exceed a fixed setpoint. This set pressure is less than the set pressure for the pressurizer safety valves.

5. A reactor trip is actuated if any two-out-of-three pressurizer level channels exceed a fixed setpoint when the reactor power is above approximately 10%.

In addition to the above listed reactor trips, there are the following RCCA withdrawal interlocks:

1. High neutron flux (one-out-of-four power range)
2. Overpower ΔT (two-out-of-four)
3. Overtemperature ΔT (two-out-of-four)

The Uncontrolled Rod Withdrawal at Power event is analyzed in Reference 1 with the following tolerances on the setpoints of PSVs and MSSVs. PSVs have +5% tolerance on their setpoints and the MSSVs have +3% tolerance. With these tolerances, the results show that the primary pressure does not exceed the design limits.

The DNB response during this event is a function of the reactivity insertion rate, Doppler coefficients, moderator coefficient, reactor trip setpoints, and PORV lift setpoint. Consequently, changes in safety valve lift tolerances have no effect on DNBR. Because no fuel pins experience DNB for this event, offsite doses are bounded by the loss of AC event, evaluated in Section 4.2.8.

Conclusion:

The Uncontrolled Rod Withdrawal at Power event passes the criteria of acceptance for dose releases and over-pressurization of the primary and the secondary systems with tolerances of +5% on the setpoint of the PSVs and of +3% on the setpoints of MSSVs.

4.2.3.2.3 Rod Cluster Control Assembly Misalignment

The rod cluster Control Assembly Misalignment event is a power reduction event characterized by dropping a full-length rod, or a full-length assembly bank or statically misaligning a full length assembly. Hence the upper limit of the safety valve setpoint tolerances will have no impact on the pressurization of the primary or the secondary systems. The lower limit of the

tolerance on the PSVs does not have any relevance to the primary system protection since this will keep the primary pressure lower than in the upper limit if PSVs come into play in pressure control in this event. On the secondary side, since the MSSVs do not open in this event, the lower limit or the upper limit of the setpoint tolerance does not affect the offsite dose releases.

The parameter of importance in this event is the DNBR in the hot channel due to increased peaking factors. The safety valves setpoint tolerances do not alter the important parameter of this event in any way. This is true for all the various categories of events included in this general event.

Conclusion:

This event is analyzed for evaluating the DNB in the hot channel due to increased peaking factors. The setpoints of the PSVs and the MSSVs do not play any part in the DNB. Therefore, this event is unaffected by changes in safety valve lift tolerances.

4.2.3.2.4 Uncontrolled Boron Dilution

The boron dilution event can be one of various types of this event depending upon the initial condition of the plant. The dilution event that occurs when the plant is at full power happens to produce the worst consequences. The reactivity insertion equivalent for this event is about 2.5 pcm/sec. This event is cycle-specific and the equivalent reactivity insertion for the analysis for a particular cycle will be evaluated for each cycle. This equivalent reactivity insertion rate is generally far smaller than the 75 pcm/sec reactivity insertion used for the rod- withdrawal-at-power event analyzed in Reference 1. The consequences of the boron dilution events will therefore be bounded by the rod-withdrawal-at-power event.

The acceptance criteria for this event are

1. The design DNBR limits shall not be exceeded

2. 110% of the design pressures of the primary and the secondary systems shall not be exceeded.

The rod-withdrawal-at-power event, has been analyzed with +5% PSV setpoint tolerance and +3% setpoint tolerance on the MSSV setpoint. For overpressurization response of the primary and the secondary systems, the rod-withdrawal-at-power passes the acceptance criteria.

The DNB response to this event is unaffected by a change in PSV setpoint tolerance because the pressurizer PORV is used to limit the pressure to a value lower than the PSV set pressure.

Conclusion:

The boron dilution event will allow a setpoint tolerance of +5% on the setpoint of the PSVs and of +3% on the setpoints of MSSVs.

4.2.3.2.5 Partial Loss of Forced Reactor Coolant Flow

A partial loss of forced reactor coolant flow accident can be caused by a mechanical or electrical failure in a reactor coolant pump, or by a fault in the power supply to the pump or pumps supplied by a reactor coolant pump bus.

If the reactor is at power at the time of the accident, the immediate effect of the partial loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not promptly tripped.

Normal power for the pumps is supplied through buses from common station service transformers connected to 161-kV external power lines. Each pump is supplied from a different bus. When a generator or turbine trip occurs, the buses continue to be supplied from external power lines, and the pumps continue to circulate coolant through the core. The necessary protection against a partial loss of coolant flow accident is provided by the low primary coolant flow reactor trip signal which is actuated in any reactor coolant loop by two-out-of-three low flow signals.

The low flow signal trips the reactor under all power levels of operation (permissives 7 and 8 in addition to full power trip) and protects the DNB from exceeding the design limits.

The acceptance criteria for this event are

1. The DNB in the fuel hot channels shall not exceed the design limits

2. The peak pressures of the primary and the secondary systems shall not exceed the 110% of the design pressures

A similar event that has been reanalyzed in Reference 1 is the Complete Loss of Reactor Coolant Flow event. This event is far more severe than the partial loss of flow event. The analysis of the complete loss of flow event shows that all pressure

acceptance criteria are met with the increased safety valve setpoint tolerances. The DNB response is not affected by a change in PSV setpoint tolerance because the pressurizer PORV is used to limit the primary pressure to a value well below the PSV lift pressure.

Conclusion:

This event is bounded by the complete loss of forced reactor coolant flow event. The complete loss of forced reactor coolant flow event has been analyzed and the results show that the setpoint tolerance of +5% on the setpoint of the PSVs and of +3% on the setpoints of MSSVs will not challenge the design pressure limits or the DNB response for this event (Reference 1).

4.2.3.2.6 Startup of An Inactive Reactor Coolant Loop

Sequoyah Technical Specifications require that all four reactor pumps be operating while the reactor is critical. Consequently, this is not a credible event, and it is not evaluated.

4.2.3.2.7 Loss of External Electric Load and/or Turbine Trip

The loss of electric load event is initiated by a loss of external load or a turbine trip. Power to plant components is available and RC pumps continue to operate. The turbine valve closure interrupts the heat sink for the plant, resulting in an overheating transient in the primary. The primary heat removal is impaired until the secondary safety valves open. The secondary steam dump valves are assumed not available. On the primary side, the pressure will increase and the PSVs will open to maintain the pressure within the design limits. This transient extensively challenges the primary and the secondary safety valves compared with other overheating transients and is significantly affected by a change in the tolerance limits of the safety valves.

The LOEL event is the most severe overheating Condition II event. The acceptance criteria for this event are

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.

This event was analyzed in Reference 1 with +5% and +3% lift tolerances on the PSVs and MSSVs, respectively. The results of

the analysis indicates that the primary and secondary pressure responses are less than the design pressure limits. Minimum DNBR for this event is bounded by that for the complete loss of flow event. This relationship is not affected by changes in safety valve setpoints.

The negative tolerance on the PSVs is immaterial so far as the pressure protection of the primary is concerned. Since the relaxation of PSV setpoint lift tolerance will not affect the margin to DNB, the limiting Condition II event for offsite dose, a loss of AC power, remains limiting. The dose release aspect of a negative tolerance on the MSSVs is addressed in Section 4.2.8.

Conclusion:

Positive tolerance limits, based on system pressure criteria are set by analysis of LOEL shown in Reference 1. The results indicate that safety valve lift tolerances of +3% are acceptable.

4.2.3.2.8 Loss of Normal Feedwater

A number of causes can result in a loss of normal feedwater to the steam generators. These include pump failures, valve malfunctions, and a loss of offsite AC power. In all cases, the loss of normal feedwater represents a decrease in heat removal capability by the steam generators below the rate of heat generation in the core. The core is protected by the low-low steam generator level trip which trips the reactor long before core thermal limits are approached. Ample steam generator inventory is available to provide, in conjunction with auxiliary feedwater, long term decay heat removal. For the limiting FSAR case, loss of feedwater with loss of offsite power upon reactor trip, the primary system response is the same as that of the loss of offsite power event.

Because of the continued capacity for heat transfer to the steam generators, the loss of main feedwater event does not approach the more limiting DNBR conditions presented by other transients. Although the OTΔT and OPΔT trips are available to protect the core, neither of these setpoints is encountered in the FSAR calculations. Post-trip system statepoints are well removed from the thermal limit conditions. The loss of feedwater and loss of AC power events, then, are analyzed primarily to demonstrate sufficient capability for removal of core decay heat by the steam generators without overpressurization of the primary coolant system or discharge of liquid coolant from the pressurizer.

The important components/systems that assure the successful termination of this event are

1. Reactor trip on low-low level in any SG

2. Auxiliary feedwater system availability on lack of feedwater to any of the steam generators

These components/systems are not affected by the proposed changes to the tolerance on the setpoints of the safety valves.

The acceptance criteria for this event are

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. There can be no liquid relief through the safety valves in the primary.

4. An ultimate heat sink for decay heat removal must be assured.

Because the steam generators continue to transfer energy after a loss of normal feedwater flow until the low-low steam generator reactor trip setpoint is reached (leaving decay heat as the only heat source), the peak pressures in the primary and secondary systems are bounded by the loss-of-electrical-load transient. The LOEL assumes that steam generator heat transfer capability is lost upon initiation of the transient and the primary system continues to produce energy at full power until the high pressurizer pressure reactor trip setpoint is reached. Similarly, the minimum DNBR for the loss of normal feedwater is bounded by the complete loss of forced reactor coolant pump flow transient. In the loss of reactor coolant pump flow analysis, DNBR is challenged by the full power core energy produced between initiation of the transient and the resultant reactor coolant pump underfrequency reactor trip. Post reactor trip decay heat is the only challenge to core DNBR in the loss of feedwater transient.

In this event the pressurizer fill can occur. The PSV tolerance does not affect the pressurizer fill because in the analysis, the PORV is used to maximize the pressurizer fill and the tolerance on the PSV setpoint does not affect the results. An increased MSSV setpoint negative lift tolerance affects the pressurizer

fill favorably by reducing the fill. A positive lift tolerance on the MSSV setpoint, however, aggravates the pressurizer fill. An increase of +3% on the MSSV setpoint will increase the post-trip T_{avg} by approximately 3.5 °F, but there is sufficient margin to pressurizer fill available to accommodate this change.

The heat sink is assured by auxiliary feedwater flow to the steam generators. An increase in the MSSV lift setpoint tolerance of 3% will not significantly affect auxiliary feedwater flow capacity. Adequate core cooling is, therefore, assured.

Since the relaxation of PSV setpoint lift tolerance will not affect the margin to DNB, the limiting Condition II event for offsite dose, a loss of AC power, remains limiting. The dose release aspect of a negative tolerance on the MSSVs is addressed in Section 4.2.8.

Conclusion:

The loss of feedwater event is bounded by Complete Loss of Forced Coolant Flow event with respect to margin to acceptance criteria. The latter event is analyzed in Reference 1. The analysis results show that the acceptance criteria are met with setpoint tolerances of +5% on the PSVs and +3% on the MSSVs. The setpoint tolerance has no major impact on the auxiliary feedwater availability. TVA has independently verified that the auxiliary feedwater flow assumed in the FSAR analysis is available with a +3% MSSV lift tolerance.

4.2.3.2.9 Loss of Offsite Power to Station Auxiliaries

Loss of non-emergency AC power can cause loss of power to such plant auxiliaries as the reactor coolant pumps and condensate pumps--with attendant loss of main feedwater--among others. Possible causes include loss of the offsite grid, with consequent turbine generator trip, and failure in the onsite AC distribution system. A detailed description of the loss of AC power sequence for the SQN unit is provided in Section 15.2.9 of the reference FSAR. In the bounding loss of AC power case developed in the safety analysis, the loss of AC power initiates a loss of main feedwater event. The ensuing sequence, with reactor trip on low-low steam generator level, coolant pump coastdown following reactor trip, and long term heat removal via auxiliary feedwater is the same transient as that analyzed for the loss of main feedwater event with offsite power unavailable.

The acceptance criteria for this event are

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.

Following a loss of AC power with turbine and reactor trip, the plant vital instruments are supplied power by emergency power sources. Assuming that power operated relief valves and steam dump valves are not available, the steam generator pressure will be controlled below the design pressure by main steam safety valves, to dissipate the decay heat of the core and the residual heat of the coolant.

The primary and the secondary pressurization will be controlled by the PSVs and the MSSVs opening at the setpoint plus tolerance. This event is not as severe as the LOEL event that was analyzed in Reference 1, with maximum positive tolerances of +5% and +3% for the pressurizer and steam line safety valves, respectively. Because all pressure acceptance criteria are met for the LOEL events, these criteria will be met for this event with the increased setpoint tolerances.

With respect to DNB, this event is bounded by the Complete Loss of Coolant Flow event. The loss of flow transient was analyzed in Reference 1 and it was demonstrated that margin to DNB is adequate. The DNB results remain unaffected by primary safety valve tolerance but do assume a maximum positive tolerance of +3% on the steam line safety valve setpoint. Achievement of DNB acceptance criterion for this event is therefore assured for tolerance of ± 5% and +3% on the primary and the secondary safety valve setpoints, respectively.

Since the relaxation of PSV setpoint lift tolerance will not affect the margin to DNB, the limiting Condition II event for offsite dose, a loss of AC power, remains limiting. The dose release aspect of a negative tolerance on the MSSVs is addressed in Section 4.2.8.

Conclusion:

This event is bounded by Complete Loss of Forced Coolant Flow event and the LOEL event, both of which satisfy all the acceptance criteria. Setpoint tolerances of ± 5% and ± 3% on the primary and the secondary safety valves respectively will not

adversely affect the reactor safety system's ability to meet the acceptance criteria in this event.

4.2.3.2.10 Excessive Heat Removal Due to Feedwater System Malfunction

Reductions in feedwater temperature or additions of excessive feedwater can overcool the primary system, leading to an increase in core power. The high neutron flux, OTΔT, and OPΔT reactor trips prevent any power increase which could lead to a DNBR less than the limit.

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. Continuous addition of excessive feedwater is prevented by the steam generator high-high level trip, which closes all feedwater regulator isolation valves, trips main feedwater pumps and trips the turbine.

The reference safety analysis for the Sequoyah units (FSAR Section 15.2.10) provides descriptions of the causes, effects, and consequences of the increased feedwater event. The primary system temperature response to increased feedwater is a function of the feedwater flow and temperature. Conservative reactivity coefficients are combined with the primary system cooling to obtain conservative reactivity insertion rates. The subsequent reactivity insertion rates are lower than those obtained for rod withdrawal events.

The feedwater temperature reduction can cause an increase in power in the core and an approach to DNB in the core. Higher feedwater flow than normal can cause an overcooling of the secondary and the primary systems. Either way, this is not an overheating event since the heat removed by the steam generators is higher than normal, and the setpoint relaxation in the PSVs and the MSSVs do not play a part in the consequences of this transient. The maximum reactivity rate insertion due to overcooling is bounded by the rod withdrawal from subcritical condition event.

The important parameters in this event are

1. DNB in the core due to increase in power caused by overcooling.

2. Fuel temperature and clad strain limits should not be exceeded.

Changes in safety valve setpoint tolerances have no effect on the response to this event because the setpoints are not involved in the consequences in this event.

Conclusion

This event is unaffected by a relaxation of the setpoint tolerance on the safety valves.

4.2.3.2.11 Excessive Load Increase Accident

The excessive load increase accident may occur due to either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control. An increase in steam flow is defined as excessive if it is sufficient to cause a mismatch between reactor core power and the heat removal rate at the steam generators. More specifically, increases in steam flow beyond those accommodated by the design of the reactor control system--such as a 10% step load increase or a 5% per minute ramp load increase--may cause an increase in power that can challenge the reactor protection system. In the reference FSAR, Section 15.2.11, the bounding excess steam flow/load event is represented by a step increase in steam flow of 10%.

The reference safety analysis considers four cases of the 10% step load increase. The event is analyzed for the reactor in both automatic and manual control modes, assuming both minimum and maximum moderator reactivity feedback cases for each control mode. While the reactor protection system is assumed operable for these analyses, no reactor trips are encountered. In each case, the plant is predicted to reach a new equilibrium operating point at a higher power value corresponding to the increased steam flow. DNBR results are presented, but in no case does the minimum ratio approach the design limit. Thus, the excessive load increase is not a limiting event from the DNB standpoint. Furthermore, comparison of the primary coolant system conditions for the load increase transient with those predicted for other overcooling events, indicates that the steam generator relief or safety valve failure and the steam line failure events represent more limiting transients in the overcooling category.

The core-related parameters that could affect the consequences of the excessive load event are those that govern reactivity feedback during the overcooling. Use of a lower bound absolute value for Doppler feedback assures conservative prediction of the core power increase. Because both BOL and EOL cases are considered, both upper and lower bound values are used for the moderator density coefficient.
This is an overcooling event caused by excessive steam removal from the secondary system and the setpoint tolerance will not affect this transient adversely.

The acceptance criteria for this event are

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.

As an overcooling event, the primary and secondary pressure decrease, and the safety valves do not lift, therefore, this event is unaffected by changes in safety valve lift tolerances.

Conclusion:

This event is unaffected by changes in safety valve lift tolerances.

4.2.3.2.12 Accidental Depressurization of the Reactor Coolant System

Spurious opening of a pressurizer relief or safety valve will result in reduction in reactor coolant system pressure. Should this occur during power operation, the decreasing reactor coolant pressure represents a corresponding reduction in the DNB ratio, potentially challenging the core thermal design limits. The reactor protection system provides two reactor trip signals, either of which will terminate this event before the core thermal limits are exceeded. These are the reactor trips on reactor coolant OTΔT and on low pressurizer pressure. The transient can be characterized as one of decreasing pressure at a moderate rate while other system conditions--core power, loop flows, and average temperature--remain essentially constant. The OTΔT setpoint is responsive to decreasing RCS pressure (whereas the low pressure reactor trip setpoint is static) and is the one encountered in the FSAR calculation Because there is minimal lag between neutron and thermal power, the minimum DNBR for this event immediately follows the overtemperature trip and is well

removed from the design limit. The parameter that most closely approaches the design limit is the DNB for an accidental depressurization of the RCS.

The acceptance criteria for this event are

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

The primary safety valve setpoint or the secondary safety valve setpoint do not alter the transient since they are not needed in the transient for safety functions. MSSVs may open in this transient as part of the decay heat and stored energy removal after the reactor is tripped. The effect of the higher positive tolerance on the setpoints of the MSSVs is bounded by other events such as overheating events(LOEL) where the MSSVs are required to open for longer periods of time than in a depressurization transient.

Since the relaxation of PSV setpoint lift tolerance will not affect the margin to DNB, the limiting Condition II event for offsite dose, a loss of AC power, remains limiting. The dose release aspect of a negative tolerance on the MSSVs is addressed in Section 4.2.8.

Conclusion:

This event can withstand a relaxation of the setpoint tolerance in the positive direction on the safety valves without any adverse effects.

4.2.3.2.13 Accidental Depressurization of the Main Steam System

The spurious opening or failure of a steam generator relief, safety, or steam dump valve represents the most severe overcooling event of frequency associated with accidental depressurization of the secondary system. The increased steam flow resulting from the relief, safety, or dump valve failure can be sufficient to cause a reduction in coolant temperature and pressure. This cooldown could produce a positive reactivity insertion and power increase that could challenge the fuel thermal limits. More detailed descriptions of this transient and the systems that normally provide protection against accidental depressurization of the secondary system are presented in Section 15.2.13 of the reference Sequoyah FSAR.

The bounding analysis presented in the reference FSAR assumes initiation of the event from an end-of-life, shutdown condition

and considers the most reactive RCCA to be stuck in its fully withdrawn position.

The core-related parameters that determine the potential return-to-power, the effective moderator reactivity, and power feedback coefficients, are taken at maximum and minimum values, respectively. The assumed steam release represents the maximum capacity of any single steam dump, relief, or safety valve.

Since this event is caused by the accidental opening of a single steam dump, relief or safety valve, the result is an overcooling transient on the primary side. The important safety parameter in this event is the DNB. Safety injection system actuation or overpower ΔT trip will trip the reactor in this event.

The acceptance criteria for this event are

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.

In the accidental depressurization of the main steam system event, the secondary is depressurizing and the primary is getting over-cooled. Safety valves in both the systems are neither challenged nor do they assist in the mitigation of the event. Changes to the setpoint tolerances will not, therefore, affect the progression of this transient.

Conclusion:

This event can withstand a relaxation of the setpoint tolerance on the safety valves without any adverse effects.

4.2.3.2.14 Spurious Operation of the Safety Injection System at Power

An error by the operator or a false actuation signal could produce spurious operation of the emergency core cooling system during full power operation. Westinghouse-designed plants typically provide for reactor and turbine trip signals immediately following a safety injection actuation. These would normally produce immediate termination of this event.

In the absence of these trip signals, actuation of the safety injection system will result in delivery of highly borated water to the reactor coolant system. While both high- and low-head

safety injection trains are actuated, the injection flow is from the high-head system only because the reactor coolant system is at normal operating pressure.

The effect of the injected boron is a negative reactivity excursion and decrease in reactor power. The reactor coolant system temperature and pressure decrease in direct response to the decrease in neutron and thermal power. For the bounding cases developed in the FSAR, the normal direct reactor and turbine trips are assumed bypassed, and the transient is terminated by reactor trip on low pressurizer pressure. Because the power and coolant temperature decrease throughout the transient, and the reactor coolant conditions are following the power reduction, DNBR increases throughout the event.

Because the SI system will not be able to pump excessive fluid into the primary when the reactor is at normal pressure, this event does not have any adverse effect in terms of overpressurization or dose release. The reactor will trip on the SI signal or later by the reactor protection system (RPS) and the reactor is shutdown either in a normal mode or by trip due to system low pressure, due to reduced power caused by the boron injection, depending upon the starting failure and the course of the event. The trips involved in this event are the low pressurizer pressure trip of the RPS and the SI injection signal trip.

The acceptance criteria for this event are

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.

This event does not open the safety valves in the primary since the primary pressure is continuously decreasing due to lowering of the power, as long as the operator action to terminate the SI takes place on time. If the SI injection is not terminated on time, the pressurizer will fill up and the primary system will repressurize. The pressurization rate is slow compared to overheating events. Consequently, the primary pressure will remain near the lift setpoint, and changes in the safety valve tolerances will not result in violation of the pressure acceptance criteria.

With regard to pressurizer overfill, the pressurizer liquid volume is maximized if the pressurizer PORVs are used to limit

pressure below the PSV setpoint. Therefore, pressurizer overfill is unaffected by the changes to safety valve setpoint tolerances.

The DNBR in the fuel never goes below the initial value in this transient and the proposed changes have no effect on the DNB of the fuel. Since normal shutdown of the reactor due to trip is the DNB-related consequence of this event, no adverse effect on the safety functions of the systems involved will be caused by the proposed changes, in the scenario presented in the FSAR.

Conclusion:

All acceptance criteria for this event are met with revised lift tolerances of ± 5% on the PSVs and ± 3% on the MSSVs.

4.2.3.2.15 Complete Loss of Forced Reactor Coolant Flow

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps.

If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor were not tripped promptly.

Normal power for the reactor coolant pumps is supplied through buses from a transformer connected to the generator. Each pump is on a separate bus. When a generator trip occurs the buses continue to be supplied by automatic transfer to external power lines, and the pumps continue to supply coolant flow to the core.

The following signals provide the necessary protection against a complete loss of flow accident:

1. Reactor coolant pump power supply undervoltage or underfrequency.

2. Low reactor coolant loop flow.

3. Pump circuit breaker opening.

The reactor trip on reactor coolant pump undervoltage is provided to protect against conditions that can cause a loss of voltage to all reactor coolant pumps, that is, station blackout. This function is blocked below approximately 10% power.

The reactor trip on reactor coolant pump underfrequency is provided to trip the reactor for an underfrequency condition,

resulting from frequency disturbances on the power grid. The reactor trip on low primary coolant loop flow is provided to protect against loss of flow conditions that affect only one reactor coolant loop. This function is generated by two-out-of-three low flow signals per reactor coolant loop.

The proposed changes to the safety valve setpoint tolerances have no adverse effect on the protection system trips in this event. Hence the changes will not precipitate any unplanned consequences or events.

The acceptance criteria for this event are

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.

This event was analyzed in Reference 1 with positive tolerance of +5% on the PSV setpoint and +3% on the MSSV setpoints. All pressure acceptance criteria were met. The DNB response for this event was calculated with a +3% tolerance on the MSSVs and assuming a constant primary pressure. Consequently, changes in PSV lift tolerances do not affect DNB response.

Assuming a negative PSV tolerance would result in lower primary pressures than analyzed. A negative tolerance on the MSSV setpoint, however, can produce an integrated mass of effluent steam higher than those produced with positive tolerances. Since the relaxation of PSV setpoint lift tolerance will not affect the margin to DNB, the limiting Condition II event for offsite dose, a loss of AC power, remains limiting. The dose release aspect of a negative tolerance on the MSSVs is addressed in Section 4.2.8.

Conclusion:

This event can withstand a relaxation of the positive setpoint tolerance on the safety valves without any adverse effects.

4.2.3.3 Condition III Events

Condition III events are generally more limiting than Condition II events. Condition III events are allowed to have some fuel failures so long as the site dose releases are within the 10CFR100 limits. The Condition III events are events of infrequent occurrence.

Condition III events of limiting nature have been reanalyzed for the Sequoyah Nuclear Plant with the proposed or higher tolerances on the safety valve setpoints and the Topical Report for Mark-BW Fuel Assembly Application for Sequoyah Units (Reference1) shows the results. The results indicate that the higher tolerances on the safety valve setpoints are acceptable. The following addresses each of the events under this category specifically from the point-of-view of the setpoint tolerance changes proposed.

4.2.3.3.1 Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes which Actuates Emergency Core Cooling System

The "Small Break LOCA" event is classified as a Condition III event for Sequoyah. The event is initiated from full power conditions. The limiting break configuration, a small rupture in the reactor coolant pump suction piping, is assumed. Further, a spectrum of break sizes is analyzed to assure that the limiting break size is captured. The protection mechanisms for this event include (1) the low pressurizer pressure reactor trip, (2) the safety injection actuation signal, and (3) the alignment and delivery of replacement coolant from emergency core cooling system.

The acceptance criteria for this event are dictated by General Design Criterion 35:

1. Fuel and clad damage that could interfere with continued effective core cooling is prevented.

2. Clad metal-water reaction is limited to negligible amounts.

The pressurizer safety valves play no part in the protection of the plant in the small break event as the reactor coolant system pressures are, in all cases, less than normal operating pressure. Changes to the setpoint tolerances associated with the primary valves, therefore, have no effect on this transient.

The steam line safeties are relied on for heat removal in this transient. The turbine is tripped and steam dump and power-operated atmospheric relief valves are assumed inoperable. An increase in the positive setpoint tolerance associated with the steam line safeties adversely affects primary heat removal and the progression of the small break transient. An increase in the negative setpoint tolerance would effectively enhance the margin to acceptance for this transient.

This event was analyzed in support of the FCF/Sequoyah fuel reload, Reference 1. A maximum positive secondary valve tolerance of +3% was assumed. The results of the analysis indicated significant margin to the relevant acceptance criteria for small break LOCA.

Conclusion:

The proposed changes in the safety valve tolerances have been considered in the analysis of the Small Break LOCA event for Sequoyah. The analysis results indicate that these relaxed tolerances are acceptable.

4.2.3.3.2 Minor Secondary System Pipe Breaks

Minor secondary system pipe breaks are of size less than 6 inch diameter pipe breaks. This event is a part of the spectrum of steam line break events and the feedwater system pipe ruptures. This event is not a bounding event. For offsite dose, the main steam line break is bounding because the postulated one gallon per minute primary-to-secondary leakage would leak to a nearly dry, depressurized steam generator with no iodine partitioning.

This event is generally a depressurizing event for the secondary side and an overcooling event for the primary side. As an overcooling event, the primary and secondary limits are not challenged. Therefore, safety valve tolerances have no effect. With respect to dose and DNB, this event is bounded by the Main Steam Line Break event. Changes in safety valve tolerances do not alter this relationship. Because changes in safety valve tolerances do not affect MSLB, this event also is unaffected.

Conclusion:

Changes to the safety valve setpoint tolerances have no effect on the consequences of this event.

4.2.3.3.3 Inadvertent Loading of a Fuel Assembly into an Improper Position

This event occurs when one or more of the fuel assemblies are loaded at the wrong position or if the manufacturing error has created fuel pins of wrong enrichment. Power distortions can occur under the conditions of this event, causing higher peaking factors and higher than normal heat fluxes. This can lead to DNB in the fuel assemblies.

Since this event is a power anomaly event, it does not involve safety valves in any way and hence safety valve setpoint tolerance changes are irrelevant in this event.

Conclusion:

Safety valve setpoint tolerance changes have no impact on the consequences of this event.

4.2.3.3.4 Waste Gas Decay Tank Rupture

This event is described as an uncontrolled and unexpected release of radioactive gases stored in the waste gas decay tank. This event does not take place in the NSSS.

This event does not involve the use of code safety valves either in the primary system or the secondary system and hence the tolerance change on these valves is irrelevant to the consequences of this transient.

Conclusion:

Safety valve setpoint tolerance changes have no impact on the consequences of this event.

4.2.3.3.5 Single Rod Cluster Assembly Withdrawal at Full Power

This event is possible only due to inadvertent operator action and not due to any single failure in the rod control system. The inadvertent rod cluster pull by the operator will be accompanied by alarms and visual indications in the control room. This event is bounded by the Rod Withdrawal at Power event that has been analyzed with the higher tolerances on both the primary and the secondary system safety valves, so far as the impact of the safety valve setpoint tolerances are concerned. This event is analyzed for local effects on the fuel and not for the overall effects on the system. RWAP events are very similar and bounding this event. The main parameters that are important in this event are the DNBR in the fuel assemblies at the spot where the rod cluster is withdrawn. Overtemperature ΔT trip affords the protection needed in this event by tripping the reactor, since the high neutron power trip may not occur before the overtemperature ΔT trip.

This event is a DNB-limited event and the local peaking factors may be affected in the transient by the nature of the event. But the safety valve setpoint tolerances have no impact on the local neutron fluxes and hence do not have any adverse effect on the fuel elements or the off-site dose release. Since this event is

bounded by the RWAP events, the primary and the secondary system pressure boundaries are protected by the code valves against overpressurization, with the positive tolerance on the safety valve setpoints, as shown in the RWAP events (section 4.2.3.2.2).

Since the relaxation of PSV setpoint lift tolerance will not affect the margin to DNB, the limiting Condition III event for offsite dose, steam line break, remains limiting. The dose release aspect of a negative tolerance on the MSSVs is addressed in Section 4.2.8.

Conclusion:

Safety valve setpoint tolerance changes have no impact on the consequences of this event.

4.2.3.3.6 Steam Line Break Coincident with Rod Withdrawal at Power

The steam line break coincident with rod withdrawal at power is an event that is classified as Condition III event for the Sequoyah plants. This event occurs at full power conditions. The initiating event is a steam line break which causes the rod withdrawal to occur. The reactivity addition due to the rod withdrawal is assumed to be slow (45 inches/minute with a differential worth of 20 pcm/inch). The high neutron flux trip and the overtemperature ΔT trips are disabled in this event. Only the overpower ΔT trip provides the necessary protection to the reactor. Due to the depressurization on the secondary side, SI signal will be generated and this also provides the necessary protection to trip the reactor in a timely fashion, if the break size is large enough.

The acceptance criteria for this event are

1. the thermal margin limits (DNBR) as specified in the Standard Review Plan shall be met.

2. Fuel centerline temperatures as specified in the Standard Review Plan shall not be exceeded.

3. Reactor pressure boundaries shall not exceed the ASME Code design limits on pressures.

This event has been analyzed for the FSAR with the proposed or higher tolerances on the setpoints of the safety valves. Minimum DNBR occurs shortly after rod insertion following the reactor trip. Therefore, positive or negative changes in setpoint

tolerances do not affect DNB. The reload analyses used +5% and +3% tolerances and the pressure limits were not exceeded. The results show that with the proposed positive tolerance changes the event passes all the acceptance criteria, with adequate margin. This event is documented in Reference 1.

This event is DNB-limited and hence change in the tolerances of the setpoints of safety valves does not affect the peak pressures adversely. The off-site dose due to MSSVs opening during the post-trip period is bounded by a main steam line break. The offsite dose releases due to negative tolerance on the MSSV setpoints are addressed in Section 4.2.8.

Conclusion:

All applicable acceptance criteria are met with ± 5% PSV lift tolerance and ±3% MSSV lift tolerance.

4.2.3.4 Condition IV Events

The Condition IV events are generally more limiting than Condition III events. Condition IV events are not expected to take place but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These represent the most limiting design cases.

Since in these events, radioactive gases will be released to the atmosphere due to the nature of the events, the dose release through the MSSVs become unimportant. The radioactive releases due to the major failures assumed in these events will bound all other classes of events in radioactive dose releases. MSSV setpoint tolerances do not alter the total radioactive releases in most of these events.

All Condition IV events have the potential to have fuel damage and since the Condition IV events are not expected to occur on any regular basis, certain amount of radioactive releases are allowed following the event. Hence all Condition IV events are examined for fuel damage and if determined as such, the dose release due to the assumed fuel damage is calculated.

4.2.3.4.1 Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

This class of events of any size break in the primary system piping will depressurize the reactor very rapidly. The important parameters in this event are fuel cladding temperatures during the uncovery phase and during the reflood phase and the fuel pin

failures, Zirc-water reactions and resultant hydrogen generation. The containment pressurization and release of radioactive gases to atmosphere also will be a major parameter to study in this set of events. None of these parameters are affected by the change in the tolerance limits of the safety valves since the safety valves do not play any role in this set of events. Reference 1 contains the reanalysis of this set of events and the results show that with the positive tolerances on the safety valve setpoints, the consequences are acceptable per the acceptance criteria set for this class of events.

Since in an LBLOCA event, the primary and the secondary pressures do not increase but fall during the event, safety valve setpoints do not affect the results. However, the safety valves setpoint tolerance change does affect the peak values of some of the parameters of importance in this set of events, but Reference 1 shows that all the acceptance criteria are met.

Conclusion:

Safety valve setpoint tolerance changes do not have any negative impact on the acceptability of this event.

4.2.3.4.2 Major Secondary System Pipe Rupture

The main steam line pipe rupture with a DEG of the main steam pipe is a bounding event in this category for containment pressurization and dose release to the atmosphere. This event results in overheating of the core and hence the DNB is of primary concern in this event. Getting the secondary system under control to ensure long-term viable decay heat removal through the secondary system is another major concern in this event, besides the dose releases through the break and the containment overpressurization. On the primary side, feedback effects under certain conditions can cause DNB concern in the hot channel. Generally a spectrum of break sizes are analyzed to determine the various aspects of this event, with break both inside and outside the containment.

In all the steam line break events, since the secondary system is getting depressurized, the MSSVs do not come into play in overpressure protection of the secondary until the intact steam generators are isolated. Since in this event, the steam releases through the bounding break size is far greater than the combined steam releases of the MSSVs during the post-trip period, the tolerance on the setpoints of the MSSVs become irrelevant in the way it may affect the resultant dose releases.

On the primary side, this event results in overcooling of the primary system and hence the PSV setpoint tolerance is irrelevant.

The importance of this event in the safety valve setpoint tolerance study is that the offsite dose for this event generally bounds the dose for Condition III & IV events where the activity releases are through the MSSVs. This event has been shown to be acceptable by the criteria established for dose releases in Reference 1.

Conclusion:

Safety valve setpoint tolerance changes have no impact on the consequences of this event.

4.2.3.4.3 Major Rupture of a Main Feedwater Pipe

The main feedwater pipe rupture results in a rapid blowdown of the affected steam generator through the feedwater nozzle. This results initially in a major overcooling of the primary system. Consequently, the core power increases due to the positive reactivity feedback of the moderator coefficient. Following the dryout of the affected steam generator, the primary system starts heating up. In the analysis, the low-low steam generator level will start the SI injection signal and this trip is delayed in the analysis until 0% narrow span level is reached and with the signal delays associated with the low level reactor trip. The reactor is tripped at this point and this trips the turbine. The intact steam generators will be isolated and the auxiliary feedwater system will be turned on. The primary system pressure will increase until the PSVs will control the peak primary pressure. This event can be analyzed for peak primary pressures or for pressurizer overfill. In the event that gives the peak primary pressures, the PSVs limit the peak pressures and the event is bounded by the LOEL.

This event is a limiting fault event and the acceptance criteria for this event are

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.

Since SI is initiated in this event, eventually the PSVs will vent liquid through the valves, if PORVs are not modeled. In the overfill analysis, PORVs are used to accelerate the overfill and hence the PSV setpoint does not come into play. The RCS pressure will not increase above the overpressurization limits as shown in the FSAR results, since in the overfill event, the PORV will limit the peak primary pressures below the setpoint of the PSVs. Also, the relief valve capacity is far greater than the SI injection capacity and the RCS expansion rate and hence the PSVs are not challenged in the overfill scenario.

The unaffected steam generator would pressurize following the closure of the turbine stop valve on reactor trip. However, the steam generator pressure will be bounded by the LOEL event results because, in the LOEL event, the turbine stop valves close instantaneously before the reactor trips. A +3% MSSV lift tolerance will be acceptable for this event because the peak pressures for the reanalyzed LOEL event are acceptable.

The peak primary pressure for this event is bounded by the LOEL event because the reactor trip on the LOEL is delayed after turbine trip as compared with no trip delay for feedwater line rupture. Because peak pressure for LOEL with +3% tolerance is acceptable, the peak pressure for feed line break with +3% tolerance is acceptable also.

The reactor core will remain covered in this event as shown in the FSAR analysis. The tolerance change has no impact on the relief valve capacity or the SI delivery rates. However, the integrated release through the pressurizer PORV is a function of auxiliary feedwater energy removal. TVA will independently verify that the AFW flow capacity with an MSSV set pressure including a +3% tolerance meets the assumed AFW flow rate for feed line break.

Conclusion:

Safety valve setpoint tolerance changes have no adverse effect on the consequences of this event.

4.2.3.4.4 Steam Generator Tube Rupture

This event is analyzed with a complete double-ended rupture of a single tube in the steam generator, at full power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. With a coincident loss of off-site power, the secondary system can be contaminated due to the primary-to-secondary leak. As the tube rupture event proceeds, the MSSVs will discharge

radioactive contaminated steam to the atmosphere. The operator is expected to diagnose the tube rupture and identify the steam generator. He is expected to isolate the faulty steam generator on a restricted time scale to minimize and terminate the contaminated steam from escaping to the atmosphere.

The important parameters in this event are

1. The amount of reactor coolant transferred to the secondary side.

2. Consequent release of radioactivity through the MSSVs and steam dump valves.

3. Signal to SI actuation to keep the primary make-up equal to the break flow, so that primary inventory can be maintained within safe limits.

The primary will depressurize during this event and the PSVs are not challenged. Changes in lift tolerances for the PSVs will not affect this event. A positive MSSV lift tolerance would effectively reduce reactor coolant leakage and reduce dose release. A negative MSSV lift tolerance would, on the other hand, tend to increase dose release. The effects of MSSV lift tolerance on this event, from the standpoint of dose estimates, are addressed in Section 4.2.8.

Conclusion:

Safety valve setpoint tolerance changes have no adverse impact on the consequences of this event.

4.2.3.4.5 Single Reactor Coolant Pump Locked Rotor

The locked rotor event is analyzed assuming the sudden seizure of an RC pump rotor, when the reactor is operating at full power. The rapid flow reduction in the affected primary loop and the flow reversal in that loop tends to reduce the flow through the core. Consequently, the DNBR in the hot channel is reduced.

This is a limiting fault event. The acceptance criteria for this event are

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

2. Peak clad temperature shall not exceed 1800 °F

The parameters that affect the consequences of this event are the locked rotor resistance, pressurizer safety valve capacity, low flow trip delays and the primary system flow resistance. The initial margin available in the core at steady-state operation before the event is initiated also will affect the DNBR obtained during the event. This is a function of the feedback parameters and the power profile in the core.

The tolerance change on the safety valves setpoints does not affect most of these parameters except the PSV opening time. The higher tolerance will make the PSVs open later and hence the peak primary pressure will be slightly higher. The locked rotor event was analyzed with the higher positive tolerances on the setpoints of both the PSVs and the MSSVs. It was shown in the Sequoyah Reload Topical Report (Reference 1) that the peak pressures obtained in both the primary and the secondary systems are well below the design pressures.

A negative MSSV lift tolerance would tend to increase dose release. The environmental consequences of this event, however, would be bounded by steam line break. The effects of MSSV lift tolerance on dose estimate are addressed in Section 4.2.8.

Conclusion:

This event was re-analyzed with an assumed safety valve lift tolerance of +5% and +3%, applicable to the pressurizer and main steam line valves, respectively. This analysis indicates that the acceptance criteria for the locked rotor event are met.

4.2.3.4.6 Fuel Handling Accident

This event takes place in the spent fuel pit floor. A spent fuel assembly is dropped on the pit floor, resulting in the rupture of the cladding of all fuel rods.

This event has no connection to the setpoint tolerance changes since this event does not take place in the NSSS.

Conclusion:

Safety valve setpoint tolerance changes have no effect on the consequences of this event.

4.2.3.4.7 Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

The RCCA ejection is a postulated event caused by rupture of the control rod drive housing. The rupture results in a rapid

ejection of an RCCA. The resultant positive reactivity insertion and subsequent core power excursion is limited by the Doppler feedback due to increased fuel temperature and is terminated by reactor trip in high neutron power.

The ejection of an RCCS is classified as a limiting fault and is subject to the following acceptance criteria:

1. Average fuel pellet enthalpy at the hot spot below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel.

2. Fuel melting will be limited to less than 10% of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion 1 above.

3. Peak reactor coolant pressure less than that which would cause stresses to exceed the faulted condition stress limits.

The fuel response during this event is determined by the ejected control rod worth, Doppler reactivity coefficient, reactor trip delay and, to a lesser extent, moderator reactivity coefficient. None of these parameters are affected by changes in safety valve tolerances.

An increase in the pressurizer safety valve setpoint tolerance could cause an increase in the peak primary system pressure for this event. However, the peak pressure would be well below the faulted condition acceptance criterion.

The secondary system response is bounded by the LOEL event, which was reanalyzed in Reference 1. With the safety valve lift tolerance limits and shown to be acceptable.

Conclusion:

All acceptance criteria for this event are met with revised safety valve setpoint tolerances.

4.2.4 Probability and Consequences of Malfunctions of Equipment Important to Safety

The pressurizer and secondary safety valves are passive devices used to mitigate overpressure events. Even with the proposed setpoint tolerance changes, the peak pressures for events in the FSAR remain within the acceptance criteria. Consequently, the proposed changes in safety valve setpoint tolerances do not

increase the probability of malfunctions of equipment important to safety.

The dose sequences of FSAR accidents are discussed in a separate section, later in this document. That evaluation concluded that all dose acceptance criteria are met with the proposed safety valve setpoint changes. Consequently, the proposed change does not increase the consequences of malfunctions of equipment important to safety.

4.2.5 Possibility of an Accident of Different Type

The pressurizer and secondary safety valves are passive devices used to mitigate overpressure events. The proposed change is a change in the lift setpoint tolerances of the existing valves. No new connections or equipment were added to the plant. The plant operating conditions are unaffected by safety valve setpoints. Therefore, the proposed changes to the pressurizer and the main steam safety valves do not create the possibility of an accident of a different type than any evaluated in the FSAR.

4.2.6 Possibility of a Malfunction of Equipment important to Safety previously evaluated in the FSAR

The pressurizer and main steam safety valves are passive devices used to mitigate overpressure events. The proposed change is a change in the lift setpoint tolerances of the existing valves. No new connections or equipment will be added to the plant. The plant operating conditions are unaffected by safety valve setpoints. Furthermore, the peak system pressures remain below the acceptance criteria for all FSAR accidents with the proposed changes in safety valve lift tolerances. Therefore, the proposed changes in valve lift setpoint tolerances do not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR.

4.2.7 Evaluation of the Margin of Safety

The evaluations of the FSAR accident analyses in Section 4.2.3 and of the offsite dose consequences in Section 4.2.8, determined that all NRC-approved acceptance criteria are met with the proposed changes in the safety valve setpoint tolerances. Consequently, the proposed relaxation in the pressurizer and main steam safety valve setpoint tolerances does not reduce the margin of safety as defined in the basis for any Technical Specification.

4.2.8 Environmental Consequences

The evaluation of environmental consequences contained in this section was taken from Reference 4.

4.2.8.1 Environmental Consequences of a Postulated Loss of AC Power to Plant Auxiliaries

Secondary steam relief resulting from a -3% tolerance on the main steam safety valve setpoints can potentially impact the relief rate assumed in this radiological consequences analysis. Sensitivity studies on steam releases with the relaxed valve setpoint tolerance were performed and were compared to the following steam releases in the present Sequoyah analysis.

0-2 hours	487,000 lbs
2-8 hours	1,090,000 lbs

The sensitivity studies indicated that the actual steam releases with the -3% setpoint tolerance are bounded by the values used in the present analysis. Accordingly, the results of the analysis are not affected by the setpoint tolerance increase and the results presented in the FSAR remain valid.

4.2.8.2 Environmental Consequences of a Postulated Steam Line Break

Secondary steam relief resulting from a -3% tolerance on the main steam safety valve setpoints can potentially impact the relief rate assumed in this radiological consequences analysis. Sensitivity studies on steam releases from the three intact steam generators with the relaxed valve setpoint tolerances (the faulted steam generator depressurizes such that the blowdown mass is not affected by the main steam safety valve setpoint) were performed and were compared to the following steam releases assumed in the present Sequoyah analysis.

0-2 hours	479,000 lbs
2-8 hours	1,030,000 lbs

The sensitivity studies indicated that the actual steam releases with the -3% setpoint tolerance are bounded by the values used in the present analysis. Accordingly, the results of the analysis are not affected by the setpoint tolerance increase and the results presented in the FSAR remain valid.

4.2.8.3 Environmental Consequences of a Postulated Steam Generator Tube Rupture

The pressurizer safety valves are not modeled in the steam generator tube rupture (SGTR) analysis. An increase in the setpoint tolerance for these valves will have no impact on the SGTR analysis.

Increasing the setpoint tolerance on the main steam safety valves will increase the primary-to-secondary break flow and the steam released from the ruptured steam generator to the atmosphere. In general, the increased setpoint tolerance results in lower steam generator pressure after a reactor/turbine trip and a higher break flow rate due to the increased primary-to-secondary pressure differential.

A sensitivity study was performed on the present Sequoyah SGTR analysis to determine the increase in primary-to-secondary system flow and the corresponding atmospheric steam release resulting from a -3% valve setpoint tolerance. The sensitivity conservatively indicated that a 3% tolerance on the present valve setpoints will increase the integrated primary-to-secondary break flow by 5% and increase the steam released from the ruptured steam generator to the atmosphere by 37%.

Offsite thyroid and gamma body doses are proportional to steam release and/or break flow. The Sequoyah thyroid and gamma body doses will conservatively increase by approximately 37 and 5%, respectively, as a result of the increased steam release from the ruptured steam generator to the atmosphere. The present 0-2 hour doses at the site boundary are 5.5 rem thyroid and 0.6 rem gamma body. These doses are well below the 300 rem thyroid and 25 rem gamma body guidelines of 10CFR100. Although the offsite doses have increased as a result of the setpoint tolerance increase, the increases are small and the total doses remain well within 10CFR100 guidelines. As such, the conclusion of the FSAR analysis remain valid.

4.2.8.4 Environmental Consequences of a Postulated Rod Ejection Accident

Secondary steam relief resulting from a -3% tolerance on the main steam safety valve setpoints can potentially impact the relief rate assumed in this radiological consequences analysis. The present analysis assumes that 81,000 lbs of steam is discharged through the main steam safety valves during the first 350 seconds of the transient. Qualitative inspection of the present analysis indicates that the results are dominated by releases from the primary system (reactor building leakage) rather than secondary

system atmospheric releases. Increases in the secondary system steam releases up to the entire secondary system inventory (approximately 440,000 lbs) will not significantly alter the results of the analysis. In reviewing the need to perform a formal sensitivity analysis on a parameter which has little or no impact on the overall analysis results, it was noted that the methodology used to analyze the environmental consequences of a loss-of-coolant accident in Sequoyah FSAR Section 15.5.3 is bounding for the rod ejection transient (with a -3% main steam safety valve setpoint tolerance). Given that the loss-of-coolant accident analysis bounds the rod ejection transient with acceptable results, further analysis of the rod ejection transient is not required. This conclusion is consistent with the methodology recently used to evaluate the rod ejection environmental consequences analysis for the Watts Bar Nuclear Plant.

5.0 GENERAL CONCLUSIONS

All accidents in the Sequoyah FSAR were evaluated to determine the effects of a relaxation of the pressurizer and main steam safety valve setpoint tolerances. The evaluation determined that all NRC-approved acceptance criteria continue to be met with the proposed changes in safety valve setpoint tolerances. Certain limiting accidents were analyzed for FCF reload fuel (Reference 1). Those analyses incorporated the revised setpoint tolerances and showed that all acceptance criteria were met. Those events that were not reanalyzed for the FCF reload fuel were systematically examined to determine how the proposed relaxation in safety valve setpoint tolerances affect the approach to the acceptance criteria.

The probability of occurrence of any of the Chapter 15 transients will not be increased due to the changes proposed. No new transients will be created as a consequence of the changes proposed. The probability of malfunction of equipment important to safety will not be increased in any way due to the proposed changes. Because all NRC-approved acceptance criteria continue to be met with the proposed changes in safety valve setpoint tolerances, the margin of safety is not reduced.

~~This evaluation supports Technical Revisions included in Section 7.0.~~ JDS 9/26/96

6.0 REFERENCES

1. FTI Topical Report BAW-10220, Revision 00, `MARK-BW Fuel Assembly Application for Sequoyah Nuclear Units 1 and 2,' March 1996.

2. Sequoyah Final Safety Analysis Report, updated through Amendment 11.

3. Technical Specifications - Sequoyah Unit 1 & 2, Appendix A to License, Unit 1 - Amendment 221 R225, Unit 2 - Amendment 212 R212.

4. TVA Letter S-051, MJ Lorek to LW Newman, `Main Steam/Pressurizer Safety Valve Setpoint Tolerance Relaxation Analysis - Radiological Consequences,' June 26, 1996. FTI Document 38-1247160-00.

4. TVA Letter S-052, MJ Lorek to LW Newman, `Document Submittal - Main Steam/Pressurizer Safety Valve Setpoint Tolerance Relaxation Analysis,' July 22, 1996. FTI Document 38-1247177-00.

ENCLOSURE 6

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE

SEQUOYAH NUCLEAR PLANT (SQN) UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-96-07)

REVISED UPDATED FINAL SAFETY ANALYSIS REPORT PAGES

LIST OF TABLES (Continued)

LIST OF FIGURES (Continued)

boundary and the low population zone as a function of primary to secondary leak rate are given in Figures 15.5.4-1 through 15.5.4-3. The doses from this accident are well within the limits as defined in 10 CFR 100 (25 rem whole body and 300 rem thyroid) for the range of credible steam generator tube leakages.

A similar conservative analysis has been performed which evaluated the resulting offsite doses assuming no steam generator blowdown. The whole body gamma and beta doses increased by a factor of 2, and the thyroid doses increased approximately 22 percent when the steam generator blowdown assumption was changed from 12.5 gpm per steam generator to zero steam generator blowdown; however, in comparison with the 10 CFR 100 limits, the offsite doses due to a steam line break, even assuming no steam generator blowdown, are still miniscule.

15.5.5 Environmental Consequences of a Postulated Steam Generator Tube Rupture

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the Reactor Coolant System to the secondary system in the steam generators. A conservative analysis of the postulated steam generator tube rupture assumes the loss of offsite power and hence involves the release of steam from the secondary system. A conservative analysis of the potential offsite doses resulting from this accident is presented with steam generator leakage as a variable parameter. This analysis incorporates assumptions of one percent defective fuel and steam generator leakage prior to the postulated accident for a time sufficient to establish equilibrium specific activity levels in the secondary system. ~~A realistic analysis is also performed.~~ Parameters used in ~~both~~ the ~~realistic and~~ conservative analyses are listed in Table 15.5.5-1. 2

The conservative assumptions used to determine the equilibrium concentrations of isotopes in the secondary system are as follows:

1. The primary to secondary leakage in steam generators occurs when the reactor starts up, and the leakage remains constant during plant operation.

2. The primary to secondary leakage is evenly distributed in steam generators.

3. Primary coolant activity is associated with one percent defective fuel and is given in Table 11.1.1-1. The secondary side concentrations associated with a given steam generator level rate can be determined from Table 11.1.2-1.

4. The iodine partition factor is $\frac{\text{amount of iodine/unit mass steam}}{\text{amount of iodine/unit mass liquid}}$ = 0.1 (Reference 1) in steam generators. The iodine partition factor is $\frac{\text{amount of iodine/unit vol. gas}}{\text{amount of iodine/unit vol. liquid}} = 10^{-4}$ (Reference 1) in the condenser.

5. No noble gas is dissolved or contained in the steam generator water; i.e., all noble gas leaked to the secondary system is continuously released with steam from the steam generators through the condenser off-gas system.

6. The blowdown rate from steam generators is continuous at 12.5 gpm per steam generator.

The following conservative assumptions and parameters are used to calculate the activity releases and offsite doses for the postulated steam generator tube rupture:

1. Prior to the accident, an equilibrium activity of fission products exists in the primary and the secondary systems due to primary to secondary leakage in steam generators.

2. Offsite power is lost, main steam condensers are not available for steam dump. 2

3. Eight hours after the accident, the residual heat removal system starts operation to cool down the plant.

4. The primary to secondary leakage is evenly distributed in the steam generators.

5. Defective fuel is one percent.

6. After eight hours following the accident, no steam and activity are released to the environment.

7. No air ejector release and no steam generator blowdown during the accident.

8. No noble gas is dissolved in the steam generator water.

9. The iodine partition factor is $\frac{\text{amount of iodine/unit mass steam}}{\text{amount of iodine/unit mass liquid}}$ = 0.1 (Reference 1) in the good steam generators.

10. During the postulated accident, iodine carryover from the primary side in the three good steam generators is diluted in the incoming feedwater.

11. Steam release to atmosphere and the associated activity release from the nondefective steam generators is terminated at eight hours after the accident when the residual heat removal system takes over in cooling down the plant.

12. Thirty minutes after the accident, the pressure between the defective steam generator and the primary system is equalized. The defective unit is isolated. No steam and fission product activities are released from the defective steam generator thereafter.

13. The 0-2 and 2-8 hour atmospheric diffusion factors given in Appendix 15A and the 0-8 hour breathing rate of 3.47×10^{-4} m^3/sec are applicable.

The steam releases to the atmosphere for the steam generator tube rupture are given in Table 15.5.5-1.

The gamma, ~~beta,~~ and thyroid doses for the steam generator tube rupture accident for the ~~realistic~~ conservative analysis at the exclusion area boundary are ~~2.2×10^{-4}~~ 5.5 rem, ~~5.1×10^{-4} rem~~ and ~~1.0×10^{-3}~~ 0.6 rem, respectively. The corresponding doses at the low population zone are ~~1.7×10^{-5}~~ 0.7 rem, ~~4.1×10^{-5} rem~~ and ~~8.0×10^{-5}~~ 0.06 rem, respectively. (Insert A) | 10

~~The gamma, beta, and thyroid doses for the postulated steam generator tube rupture accident for the conservative analysis at the exclusion area boundary and the low population zone as a function of primary to secondary leak rate are given in Figures 15.5.5-1 through 15.5.5-3.~~ The doses from this accident are well within the limits as defined in 10 CFR 100 (25 rem deep dose equivalent and 300 rem thyroid) ~~for the range of credible steam generator tube leakage.~~ | 10

It should be noted that making a more conservative assumption of no steam generator blowdown will have no significant effect on the offsite doses since the doses are primarily due to the primary water released to the secondary through the ruptured steam generator tube.

15.5.6 Environmental Consequences of a Postulated Fuel Handling Accident

Two analyses of a postulated fuel handling accident are performed: (1) a realistic analysis and (2) an analysis based on Regulatory Guide 1.25, Rev. 0. The parameters used for each of these analyses are listed in Table 15.5.6-1.

The bases for the Regulatory Guide 1.25 evaluation for a fuel handling accident in the Auxiliary Building are:

1. The accident occurs 100 hours after plant shutdown. Radioactive decay of the fission product inventory during the interval between shutdown and placement of the first spent fuel assembly into the spent fuel pit is taken into account.

Insert A

These doses are based upon 1 gpm primary to secondary leakage.

12. No credit is taken for natural decay either due to holdup in the Auxiliary Building or after the activity has been released to the atmosphere.

13. The thyroid, gamma, and beta doses from a postulated fuel handing accident in the Auxiliary Building are given in Table 15.5.6-2. These doses are much less than the 10 CFR 100 references values of 300 rem to the thyroid and 25 rem to the deep dose equivalent. | 10

If a fuel-handling accident should occur inside primary containment, the calculated offsite doses may differ from those calculated as above. At all times during refueling operations, the containment will either be isolated or ventilated through a single train of Reactor Building Purge Ventilation System (RBPVS). The RBPVS is described in Section 9.4.7.

The offsite dose analysis for a fuel handling accident inside the reactor containment is based primarily on the assumption of the Regulatory Guide 1.25 based fuel handling accident inside the Auxiliary Building, with the following exceptions:

1. The radial peaking factor is increased from 1.65 to 1.70 in order to bound the anticipated Cycle 8 core design.

2. Activity is assumed to be exhausted by one train of containment purge until isolation occurs, through the containment purge, directly to the environment, at a rate of 16,000 cfm (see page 5).

3. The containment purge is assumed to be isolated in 30 seconds by the purge line radiation monitor. This response time bounds the demonstrated response time; Reference 15.5.8.14.

4. No iodine removal credit is taken for the Reactor Building Purge Ventilation System filters. 10

5. The volume of containment assumed to participate in mixing is 32,550 ft^3 (less than 5% of the containment free volume).

6. The efficiency of the refueling canal perimeter exhaust is assumed to be 100%, i.e., none of the activity released from the pool is assumed to escape the mixing volume.

The resulting thyroid doses at the exclusion area boundary and low population zone outer boundary are 70 and 11 rem, respectively. The whole-body gamma doses are less than 1 and 0.15 rem, respectively. These doses are "well within" the 10CFR100 guideline. "Well within" is currently defined by NRC in the Standard Review Plan (NUREG-0800) as 75 rem thyroid and 6 rem whole-body. See Reference 15.5.8.14.

15.5.7 Environmental Consequences of a Postulated Rod Ejection Accident

~~A Regulatory Guide 1.77 analysis of a postulated rod ejection accident is performed. The parameters used for the analysis are listed in Table 15.5.7-1.~~

~~The conservative analysis of the doses resulting from a rod ejection accident is based on the analysis given previously in this chapter which~~

The consequences of a postulated rod ejection accident are bounded by the results of the loss-of-coolant accident analysis evaluated in Section 15.5.3.

demonstrates a conservative fission product release of the gap activity from 10 percent of the fuel rods in the core. In addition, fission products are released from the fraction of the core which is assumed to melt.

The fraction of fuel melting was conservatively assumed to be one quarter percent of the core as determined by the following method:

a. A conservative upper limit of 50 percent of the rods experiencing clad damage may experience centerline melting (a total of 5 percent of the core).

b. Of rods experiencing centerline melting, only a conservative maximum of the innermost 10 percent of the rod volume will actually melt (equivalent to 0.5 percent of the core that could experience melting).

c. A conservative maximum of 50 percent of the axial length of the rod will experience melting due to the power distribution (.5 of the 0.5 percent of the core = 0.25 percent of the core).

The following assumptions were used in this analysis. 2

1. Prior to the accident, the plant is assumed to be operating at full power with coincident fuel defects and steam generator tube leakage. Further, it is assumed that the fuel defects are present in fuel rods generating 0.12 percent of the core thermal power. The steam generator tube leakage rate was assumed to be 1.0 gpm. The activities in the primary coolant due to operation with 0.12 percent fuel defects is given in Table 11.1.2-2.

2. 100 percent of the noble gases and iodines in the fuel-clad gaps of the fuel rods experiencing clad damage (assumed to be 10 percent of the rods in the core) are available for release.

3. The amount of activity in the fuel-clad gap is assumed to be 10 percent of the noble gases and iodines accumulated at the end of core life.

4. 100 percent of the noble gases and 25 percent of the iodines in the melted fuel (assumed to be 0.25 percent of the core) are available for release.

5. The activity released is assumed to be simultaneously and instanteously mixed in the containment atmosphere and reactor coolant volume.

6. No credit is assumed for removal of iodine in the containment due to the ice condenser.

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7. The containment leaks for the first 24 hours at its design leak rate as specified in the technical specifications of 0.25 percent per day. Thereafter the containment leak rate is 0.125 per day.

8. 25 percent of the total primary containment leakage is assumed to go to the Auxiliary Building. The remaining 75 percent of the containment leakage is assumed to go to the Shield Building.

9. For the conservative analysis, Auxiliary Building Gas Treatment System (ABGTS) filter efficiencies of 95 percent for the removal of elemental, methyl, and particulate iodines are used. For the conservative analysis, Emergency Gas Treatment System (EGTS) filter efficiencies of 95 percent for the removal of elemental, methyl, and particulate iodines are used.

10. Primary and secondary system pressures are equalized after 750 seconds, thus terminating primary to secondary leakage in the steam generators.

11. For the case of loss of offsite power, 81,000 pounds of steam are discharged from the secondary system through the relief valves the first 350 seconds following the accident. Steam dump is terminated after 350 seconds.

12. The specific activity of this 81,000 lbs. of steam is based on the equilibrium activity of assumption 1 with the addition of post- accident reactor coolant to the secondary at the design leak rate of 1 gpm for 350 seconds.

13. All releases are assumed to be at ground level.

14. Dose models and isotopic data used in the analysis are presented in Appendix 15A of this report.

Results

The gamma, beta and thyroid doses at the exclusion area boundary and low population zone for the rod ejection accident for the Regulatory Guide 1.77 analysis are given on Table 15.5.7-2. For the two hour and 30-day periods after a postulated rod ejection accident the doses at the exclusion area boundary and LPZ respectively are well within the limits as defined in 10 CFR 100 (25 rem deep dose equivalent 300 rem thyroid at the exclusion area boundary and LPZ for the 2 hour and 30-day periods after the accident, respectively). 10 10

15.5.8 References

1. Styrikovich, M. A., Martynova, O. I., Katkovska, K. Ya., Dubrovski, I. Ya., Smrinova, I. N., Transfer of Iodine from Aqueous Solutions to Saturated Vapor, Translated from Atomnaya Energiya, Vol. 17, No. 1, pp. 45-49, July 1964.

TABLE 15.5.5-1 (Sheet 1)

PARAMETERS USED IN STEAM GENERATOR TUBE RUPTURE ANALYSES

— Delete

	Realistic Analysis	~~Conservative Analysis~~
Core thermal power	3582 MWt	3582 MWt
Steam generator tube leak rate prior to and during accident	0.009 gpm	0.01 to 1.0 gpm
Off-site power	Available	Lost
Fuel defects	0.12%[1]	1%
Iodine partition factor in non-defective steam generators prior to and during accident	0.0[illegible]	0.1
Iodine partition factor in defective steam generator prior to accident	0.0[illegible]	0.1
Iodine partition factor in condenser prior to accident	0.0001	0.0001
Iodine partition factor in condenser during accident	0.0001	-
Blowdown rate per steam generator prior to accident	12.5 gpm	12.5 gpm
Time to isolate defective steam generator	30 min	30 min
Duration of plant cooldown by Secondary System after accident	8 hour	8 hour

TABLE 15.5.5-1 (Sheet 2)
(Continued)

PARAMETERS USED IN STEAM GENERATOR TUBE RUPTURE ANALYSES

	~~Realistic Analysis~~	~~Conservative Analysis~~ 46,800
Steam release from defective steam generator	~~36,000 lbs (0-30 min)~~	~~36,000~~ lbs (0-30 min)
Steam release from 3 non-defective steam generators	~~429,000 lbs (0-2 hr) 1,080,000 lbs (2-8 hr)~~	429,000 lbs (0-2 hr) 1,080,000 lbs (2-8 hr)
Feedwater flow to 3 non-defective steam generators	~~438,000 lbs (0-2 hr) 1,170,000 lbs (2-8 hr)~~	438,000 lbs (0-2 hr) 1,170,000 lbs (2-8 hr)
Reactor coolant released to the defective steam generator	~~125,000 lbs~~	~~125,000 lbs~~ 131,250 lbs
Meteorology	~~Annual Average averaged over all sectors[(2)]~~	Accident (see Appendix 15A)

Delete

~~(1) Based on operating experience of Westinghouse PWR's.~~

~~(2) 7.46 x 10^{-8} sec/m^3 at exclusion area boundary based on condenser vacuum exhaust as release zone; 5.93 x 10^{-7} sec/m^3 low population zone. Onsite meteorology for April 2, 1971 - March 31, 1972.~~ 10

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TABLE 15.5.7-1 (Sheet 1)

PARAMETERS USED IN ROD EJECTION ACCIDENT ANALYSIS

	Reg. Guide 1.77 Analysis
Core thermal power	3582 MWt
Containment free volume	1.241 x $10^6 ft^3$
Fuel defects	0.12%
Steam generator tube leakage rate prior to and during steam dump	1.0 gpm
Failed fuel	10% of fuel rods in core
Activity released to reactor coolant from failed fuel and available for release Noble gases	1% of core inventory
Iodines	1% of core inventory
Melted fuel	0.25% of core
Activity released to reactor coolant from melted fuel and available for release Noble gases	0.25% of core inventory
Iodines	0.0625 of core inventory
Iodine partition factor in steam generators prior to and during accident	0.01
Form of iodine activity in containment available for release	
Elemental iodine	91%
Methyl iodine	4%
Particulate iodine	5%

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2

TABLE 15.5.7-1 (Sheet 2)
(Continued)

PARAMETERS USED IN ROD EJECTION ACCIDENT ANALYSIS

	Reg. Guide 1.77 Analysis	
Primary containment leak rate	0.25% per day (0-24 hrs) 0.1[illegible]5% per day (1-30 days)	
Percent of containment leakage to Auxiliary Building	25%	
Auxiliary building gas treatment system filter efficiencies		
Elemental iodine	95%	
Methyl iodine	95%	
Particulate iodine	95%	
Emergency gas treatment system filter efficiencies		2
Elemental iodine	95%	
Methyl iodine	95%	
Particulate iodine	95%	
Percent of annulus free volume available for mixing of activity	50%	
Delay time of ABGTS operation	5 minutes	
EGTS flow rates	See table 15.5.3-3	
Offsite power	Lost	
Steam dump from relief valves	81,000 lbs	
Duration of dump from relief valves	350 seconds	

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TABLE 15.5.7-1 (Sheet 3)
(Continued)

PARAMETERS USED IN ROD EJECTION ACCIDENT ANALYSIS

	Reg. Guide 1.77 Analysis
Time between accident and equilization of primary and secondary system pressures	750 seconds
Steam dump to condenser	0.0
Meteorology[(1) or (2)]	Accident (see Appendix 15A)

2

[(1) or (2)] Onsite meteorology for April 2, 1971 - March 31, 1972. 6

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Table 15.5.7-2

OFFSITE DOSES DUE TO A REGULATORY GUIDE 1.77
CONTROL ROD EJECTION ACCIDENT

~~2-hour Exclusion Area Boundary~~ 10

~~Gamma Dose~~	~~1.52×10^{-1} Rem~~
~~Beta Dose~~	~~7.79×10^{-2} Rem~~
~~Inhalation Dose~~	~~8.18 Rem~~

~~30-day Low Population Zone~~

~~Gamma Dose~~	~~2.53×10^{-2} Rem~~
~~Beta Dose~~	~~1.94×10^{-2} Rem~~
~~Inhalation Dose~~	~~1.86 Rem~~

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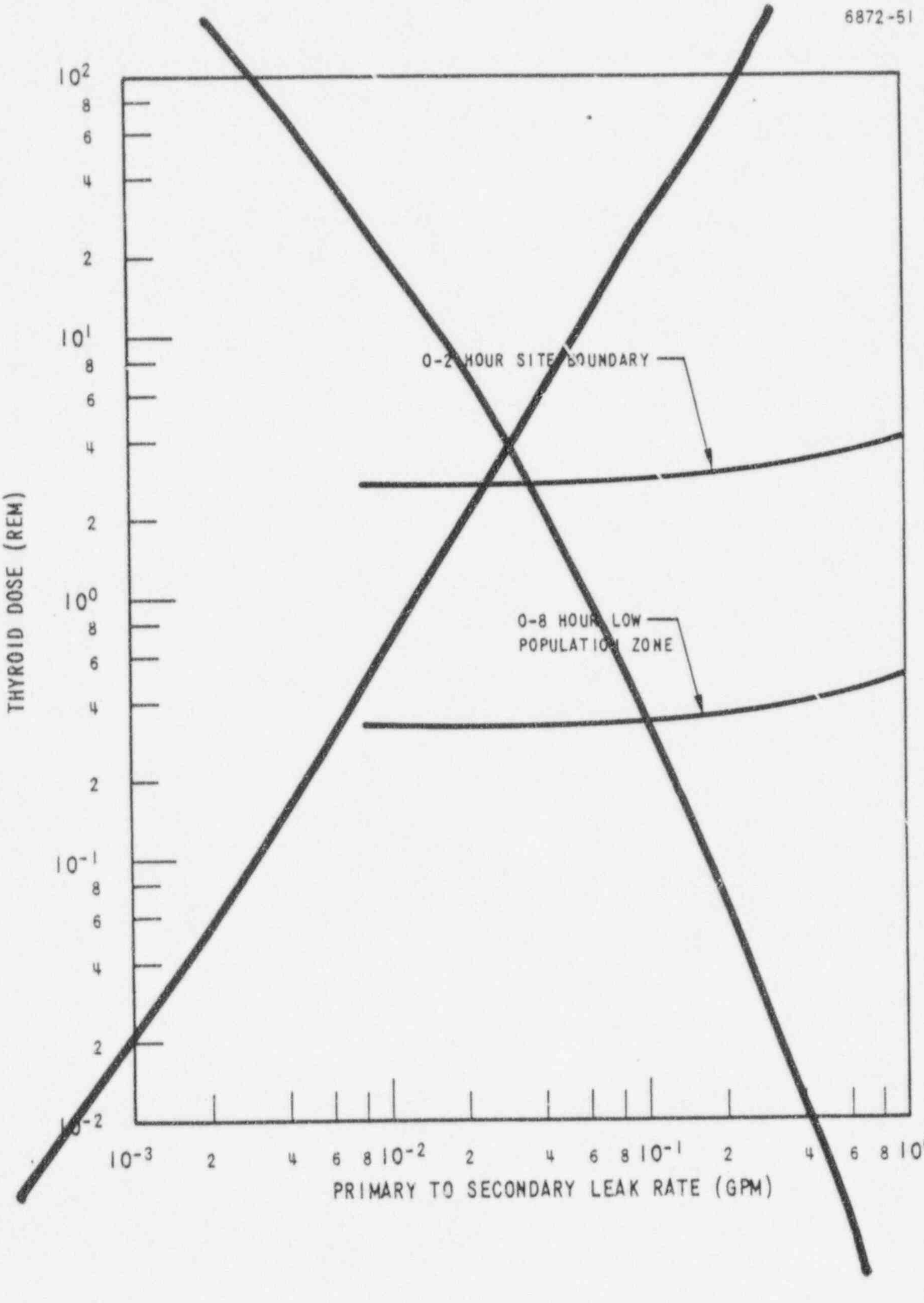

Figure 15.5.5-1 Steam Generator Tube Rupture - Thyroid Dose

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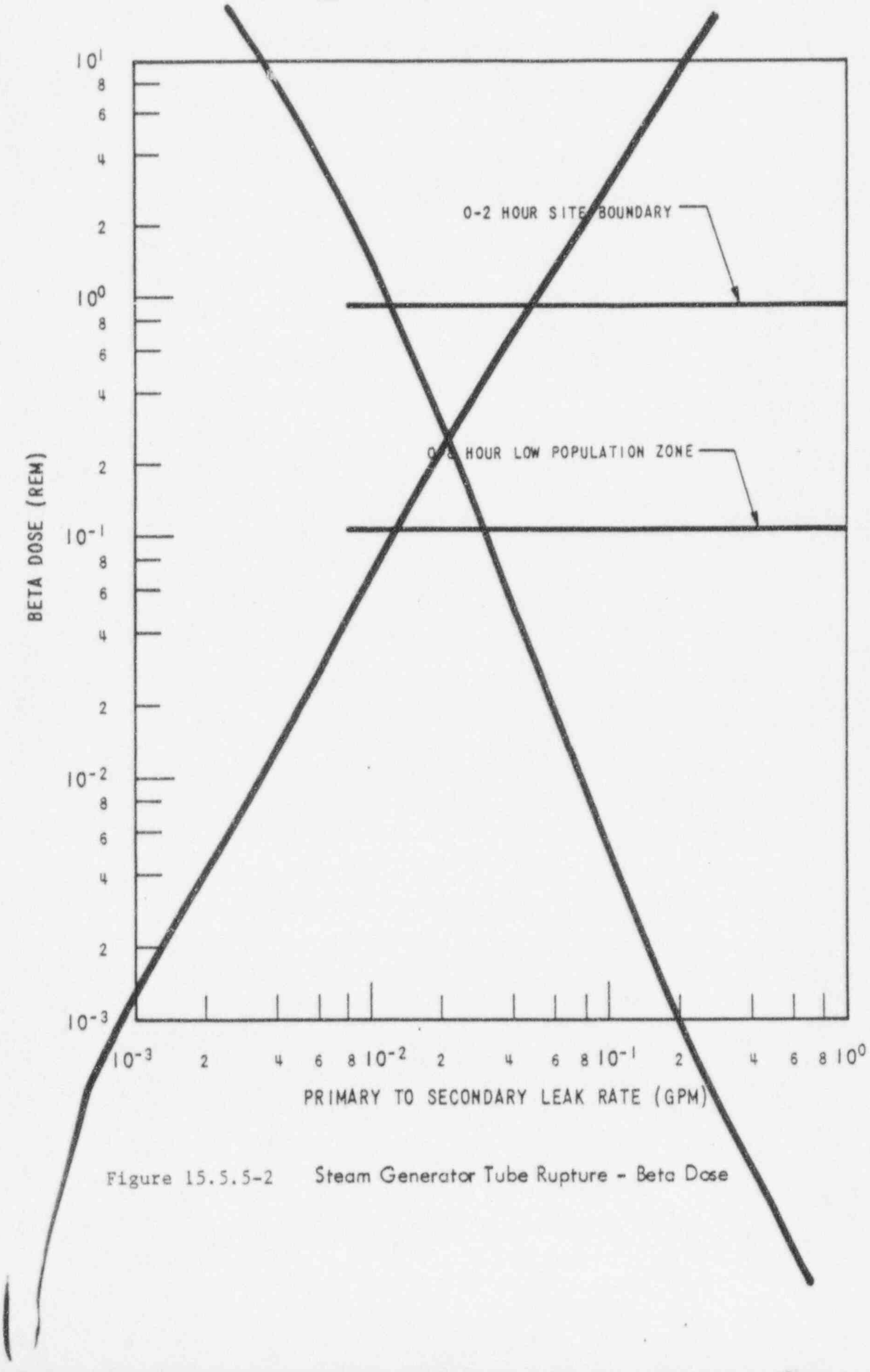

Figure 15.5.5-2 Steam Generator Tube Rupture - Beta Dose

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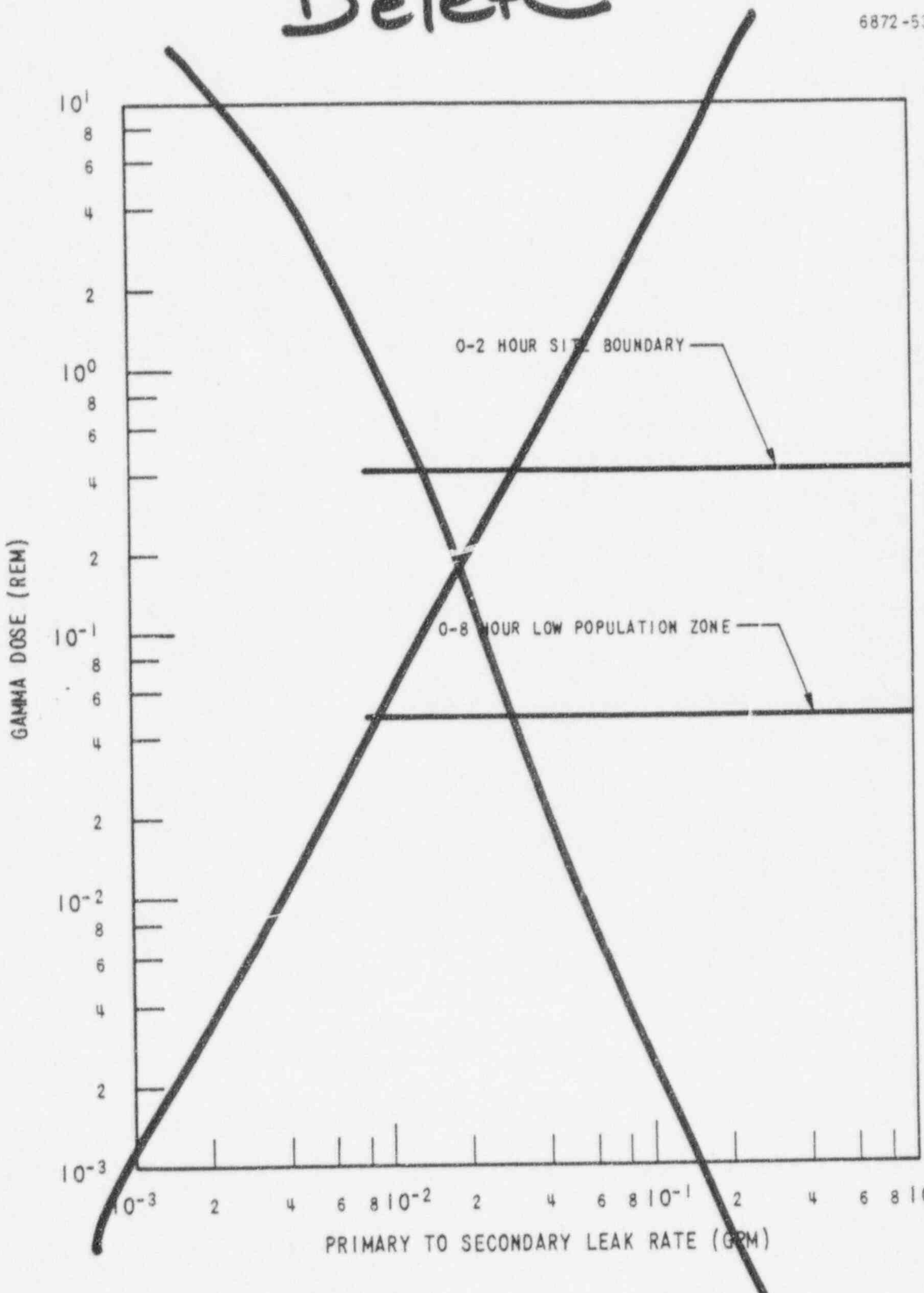

Figure 15.5.5-3 Steam Generator Tube Rupture - Gamma Dose