400 Chestnut Street Tower II

October 2, 1979

Director of Nuclear Reactor Regulation
Attention: Mr. L. S. Rubenstein, Acting Chief
Light Water Reactors Branch No. 4
Division of Project Management
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Rubenstein:

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

Enclosed are responses to Reactor Systems Branch questions (items 212.104 through 212.114) on Loss of Coolant Accidents (LOCA) transmitted by your letter to H. G. Parris dated August 21, 1979. These responses will be incorporated into the Sequoyah Nuclear Plant Final Safety Analysis Report by Amendment 62.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills, Manager
Nuclear Regulat[illegible] nd Safety

Enclosures

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ENCLOSURE

RESPONSES TO REACTOR SYSTEMS BRANCH QUESTIONS

(TRANSMITTED BY AUGUST 21, 1979 LETTER
FROM L. S. RUBENSTEIN TO H. G. PARRIS)

15.33
(212.104) The response to Q15.25 has mentioned 1/7 and 1/5-scale model testing. Please identify where the 1/5-scale model test is documented, or when it will be submitted. Please identify when WCAP-9404 will be submitted.

Response

The 1/5-scale model test is documented in WCAP-9404. WCAP-9404 was submitted to the NRC on July 11, 1979; letter NS-TMA-2070 from T. M. Anderson, Westinghouse, to J. F. Stolz, NRC.

15.34 (212.105) The response to Q15.25 states that results of 1/7-scale tests show that flow of water from the core through the RCC guide tubes into the vessel upper head "causes the fluid temperature [in the upper head region] to become somewhat greater than the cold leg value." Please quantify "somewhat greater", and its basis.

Response

The 1/7 scale model test provided the pressure gradient which existed across the upper plenum. Based on these pressure gradients, analytical calculations were performed to estimate the vessel upper head region temperature. The analytical technique employed is presented in Section 2.2 of WCAP 9404.

This analytical model was compared to the 1/5-scale model test results, which are presented in Sections 3-19 through 3-21 of WCAP 9404. In addition, in-plant measurements, obtained from various plants, are compared to the predicted values in Table 4-4 of WCAP 9404.

WCAP 9404 provides actual upper head region fluid temperatures, in terms of percentage of the difference between T_{hot} and T_{cold}.

15.35

(212.106) The response to Q15.26 has discussed temperature measurement and modifications to Sequoyah reactor internals that would increase bypass flow to assure that 4 percent of the total cold leg flow would enter the upperhead region to maintain the upperhead region temperature at the cold leg value. The response has not addressed concerns about the impact of this increased bypass flow on Chapter 15 analyses. Please address this concern.

Response

An increase in core bypass flow affects the overtemperature ΔT protection setpoint calculation. The appropriate setpoint and Tech Spec revisions have been made. The overtemperature ΔT trip is required for protection during transients which approach DNB. This trip is most important during the slow rod withdrawal at power event.

The rod withdrawal at power accident has therefore been reanalyzed, using the adjusted overtemperature ΔT setpoints. The conclusions drawn in the FSAR are unchanged for this new analysis (i.e. the DNBR does not fall below 1.30). Appropriate revisions have been made to Section 15.2 to reflect this new analysis.

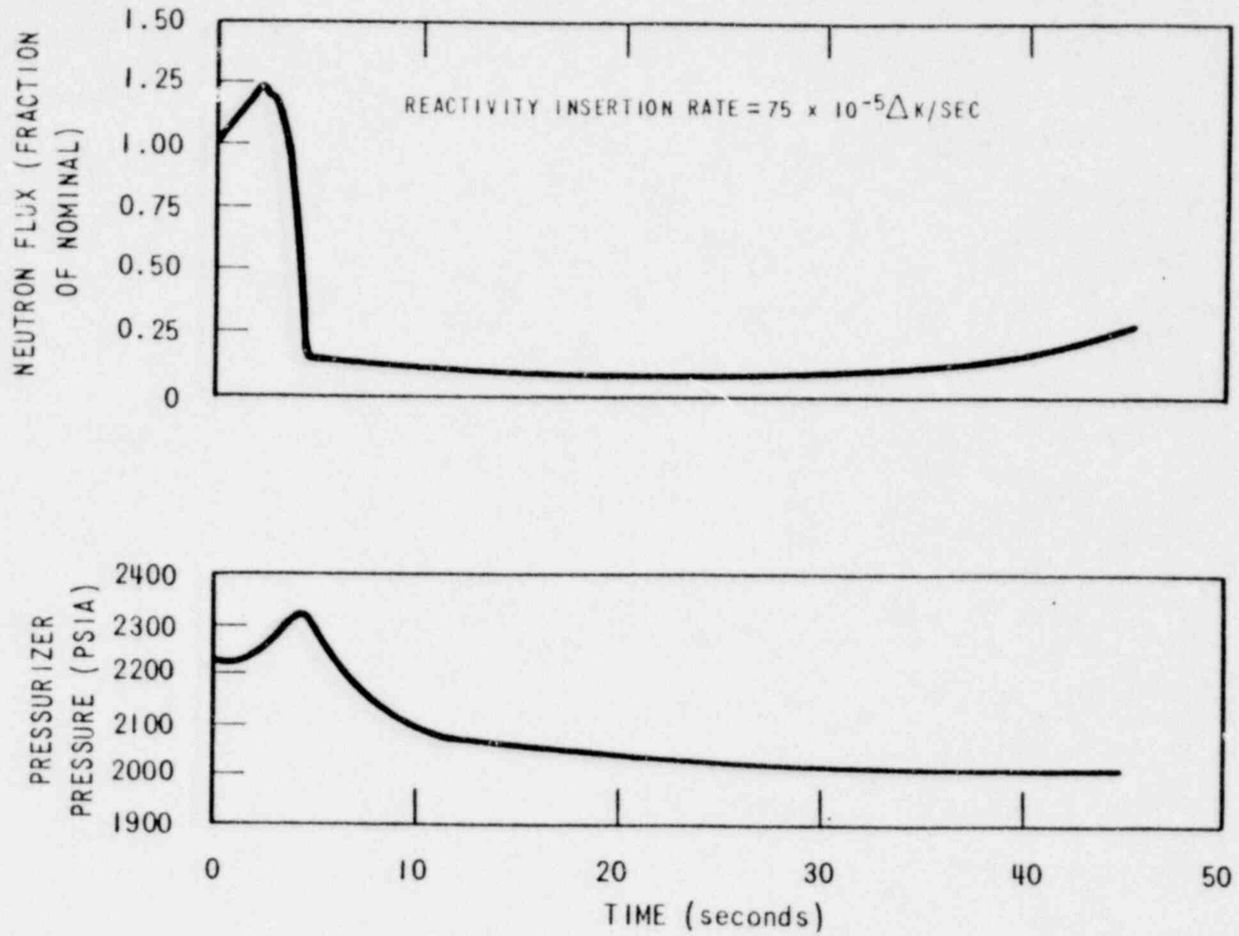

Figure 15.2-4 Typical Transient Response For Uncontrolled Rod Withdrawal From Full Power Terminated by High Neutron Flux Trip

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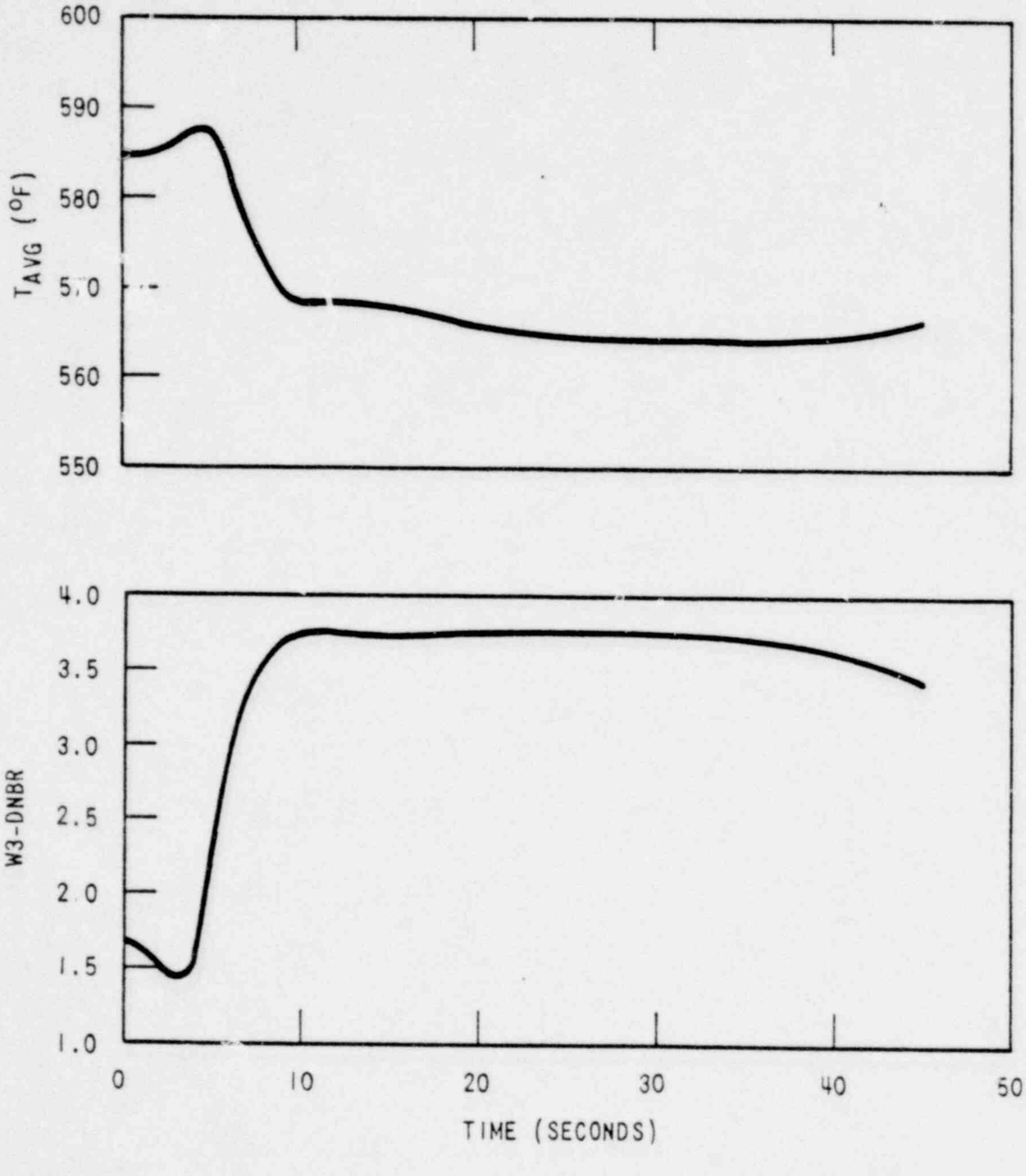

Figure 15.2-5 Typical Transient Response for Uncontrolled Rod Withdrawal from Full Power Terminated by High Neutron Flux Trip

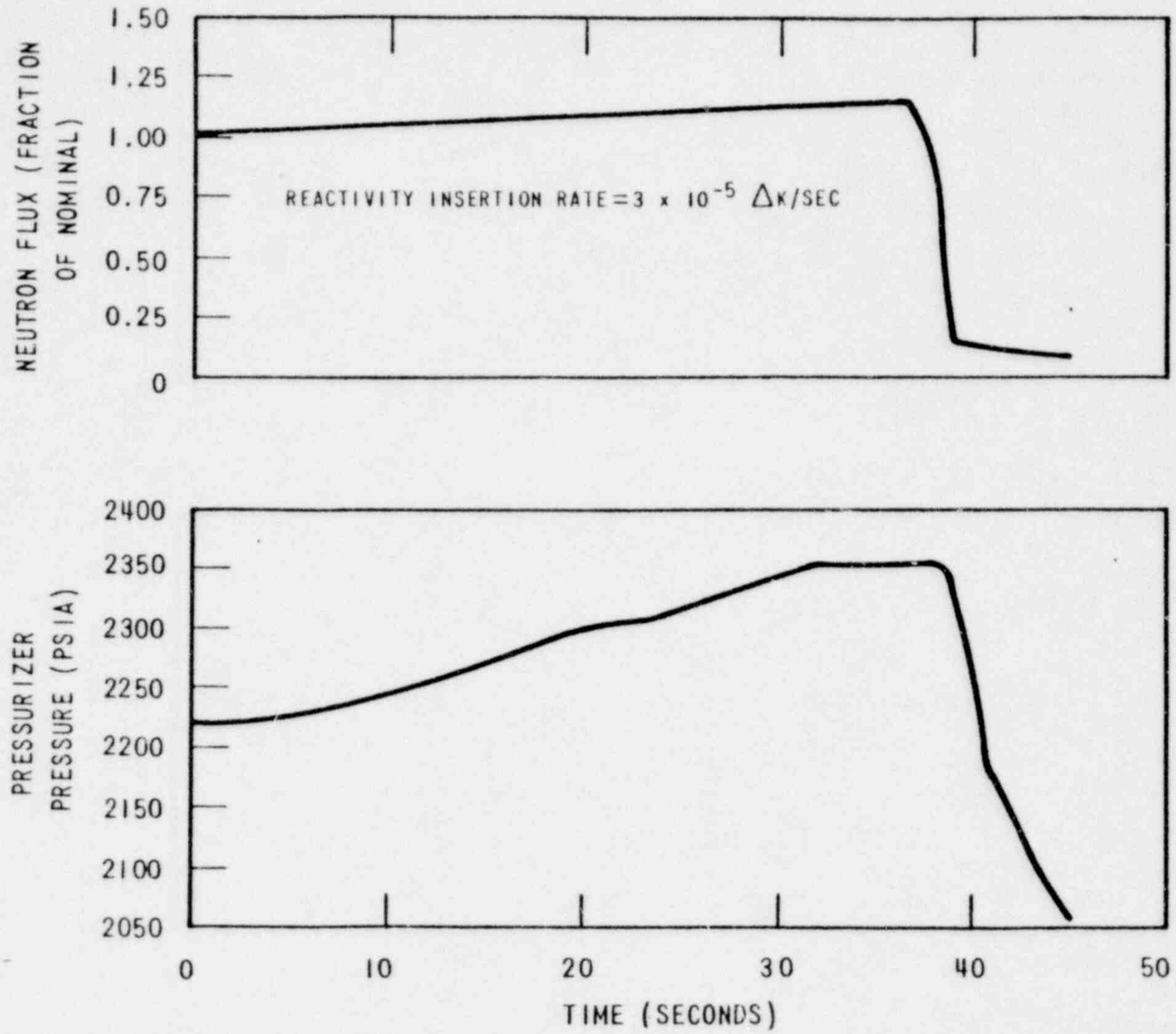

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Figure 15.2-6 Typical Transient Response for Uncontrolled Rod Withdrawal from Full Power Terminated by Overtemperature Δ T Trip

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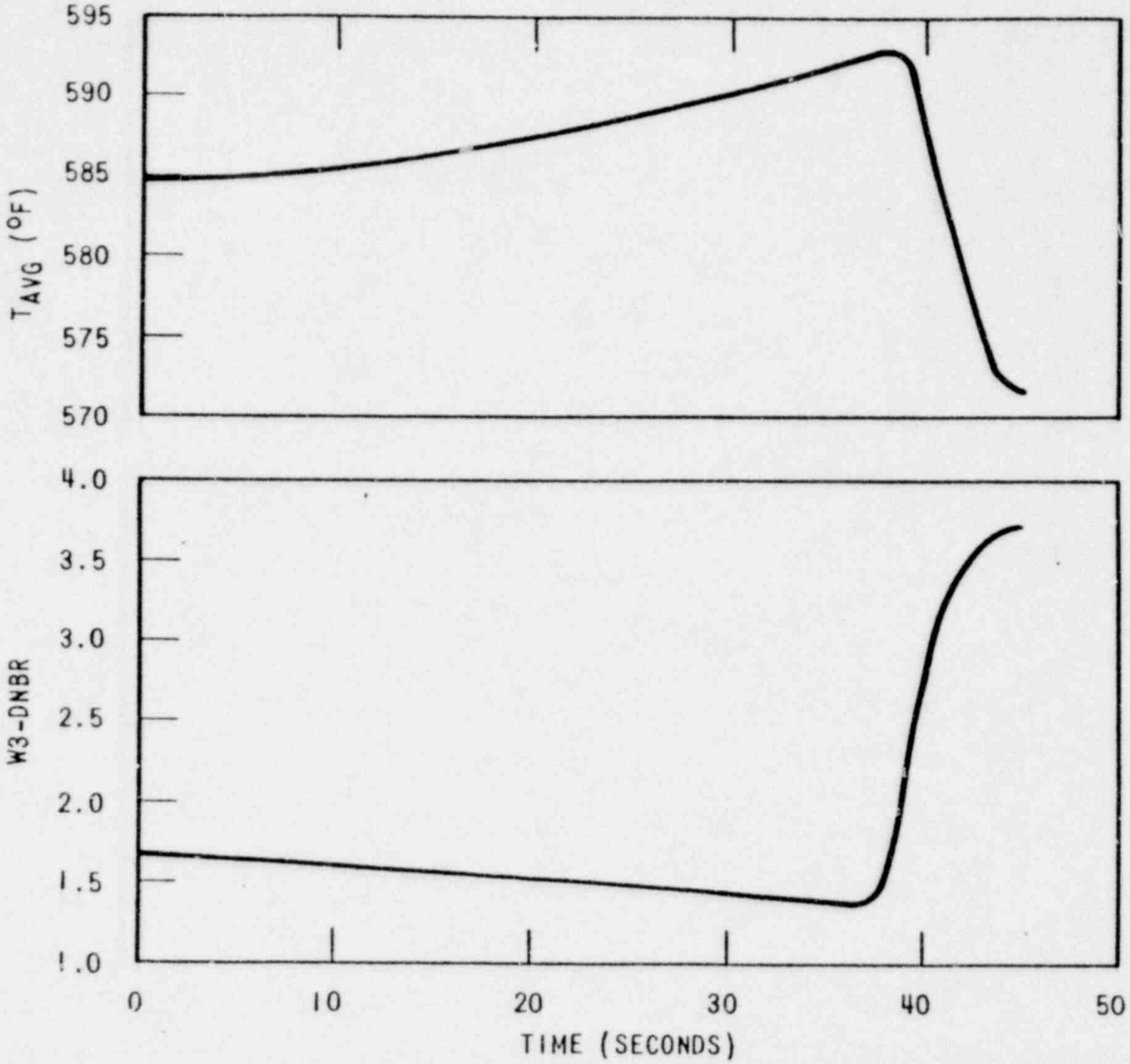

Figure 15.2-7 Typical Transient Response for Uncontrolled Rod Withdrawal from Full Power Terminated by Overtemperature ΔT Trip

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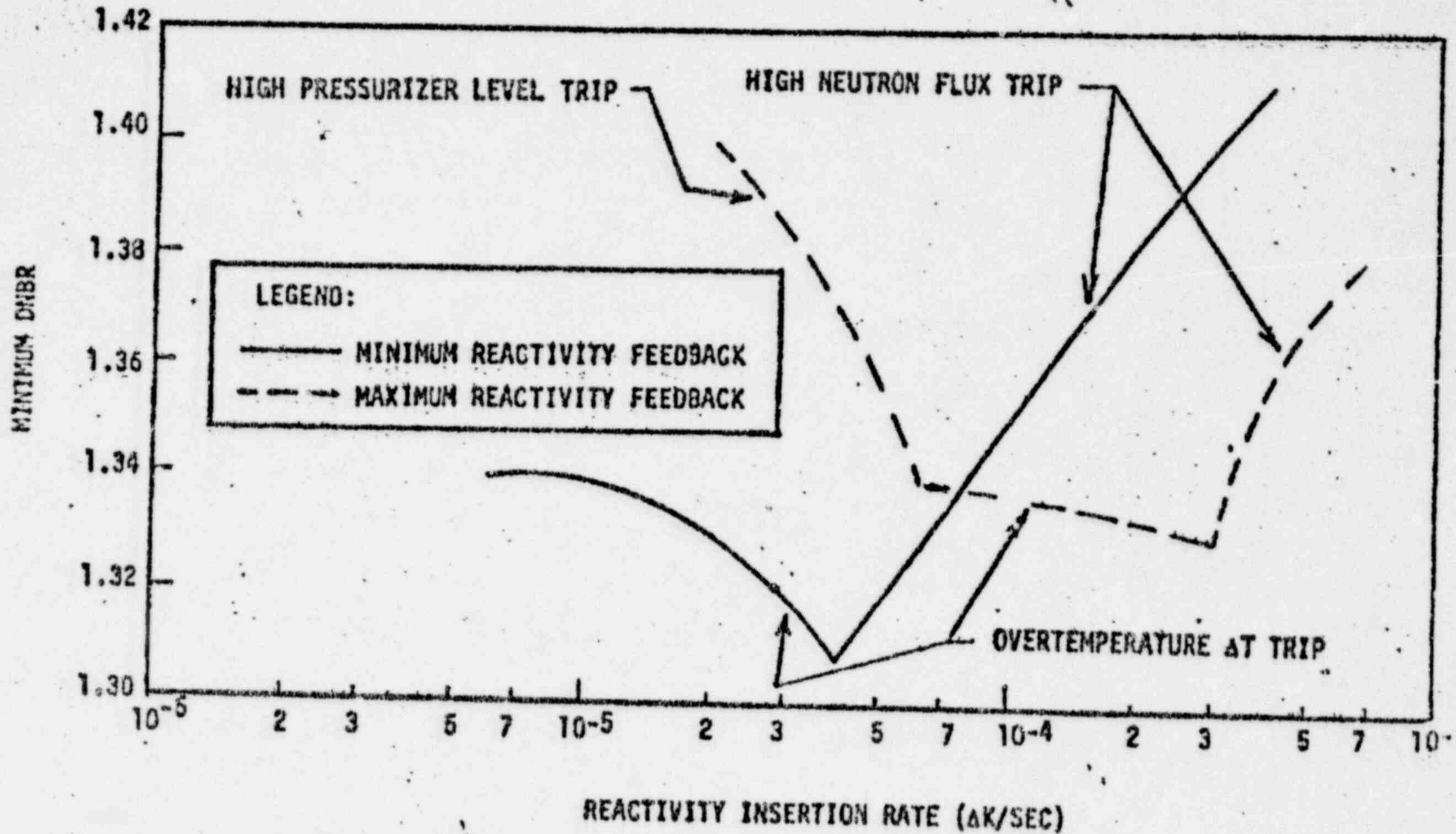

Figure 15.2-8 Effect of Reactivity Insertion Rate on Minimum DNBR for a Rod Withdrawal Accident from 100% Power

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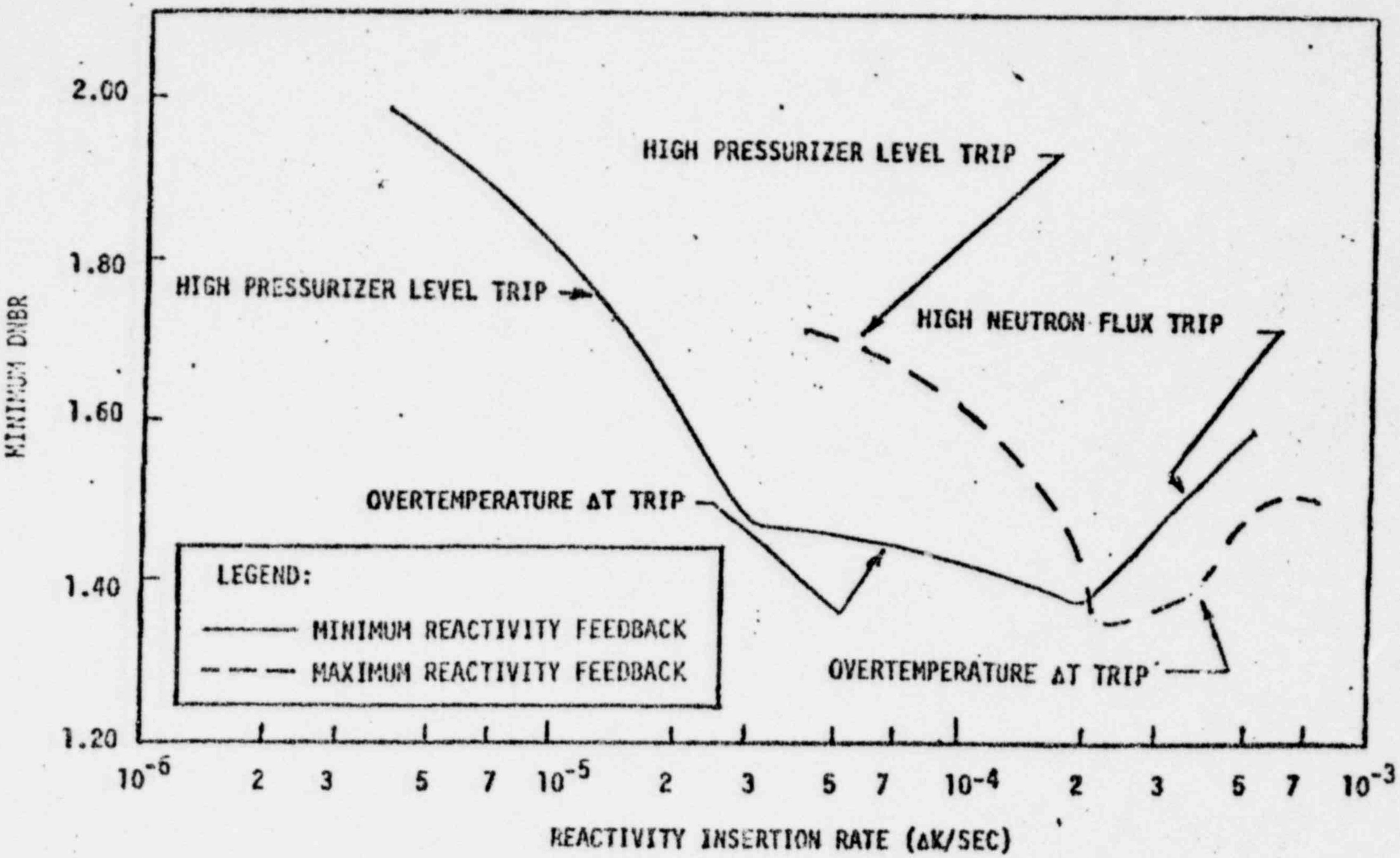

Figure 15.2-7 Effect of Reactivity Insertion Rate on Minimum DNBR for a Rod Withdrawal Accident from 60% Power

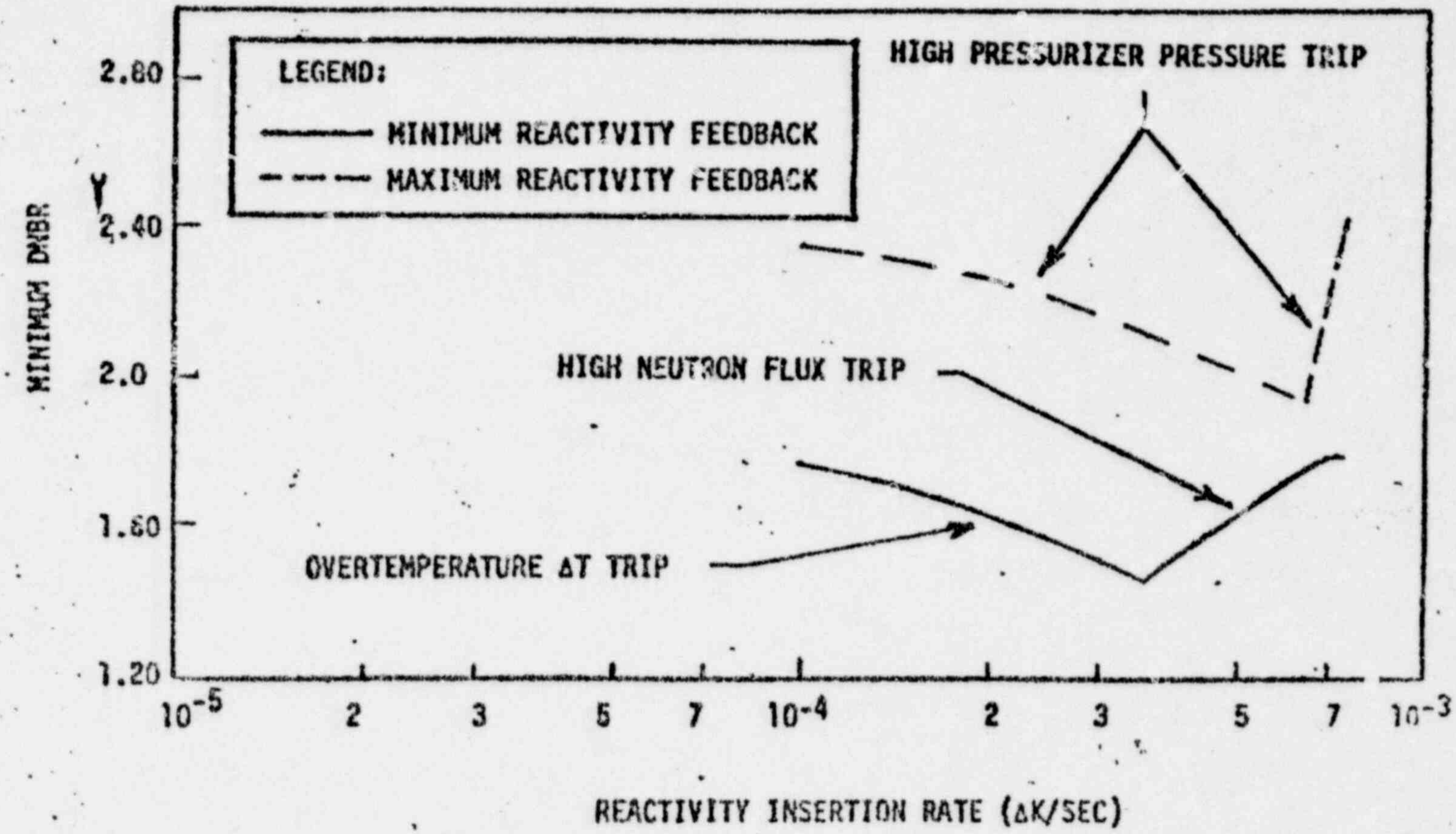

Figure 15.2-10 Effect of Reactivity Insertion Rate on Minimum DNBR for Rod Withdrawal Accident from 10% Power

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15.36 (212.107) The response to Q15.26 states that "Sensitivity studies have shown that a 100 psi reduction in UHI accumulator pressure will not alter the conclusion drawn from the FSAR non-LOCA analysis." Please identify the sensitivity studies and justify their applicability to Sequoyah. Also justify the discrepancy between statements in the response:

(a) ". . . set between 1200 and 1500 psia."

(b) ". . . assumed for the FSAR non-LOCA accident analysis is 1300 psia." and

(c) . . . difference. . . no greater than 100 psi."

Response

The setpoint for the Sequoyah UHI system may be set anywhere between 1200 and 1300 psia. The setpoint assumed for the FSAR non-LOCA analyses was 1300 psia. Therefore, the safety analysis assumption is at the high end of the Sequoyah setpoint range. The maximum difference between the safety analysis UHI setpoint assumption and the actual Sequoyah UHI setpoint will be 100 psi.

Sensitivity studies were performed, using the Sequoyah model, to determine the effect of raising (or lowering) the UHI setpoint assumed in the steam line rupture analysis. A high UHI setpoint results in a relatively early actuation of the UHI system during the reactor coolant system depressurization caused by the steam line rupture. The UHI addition then would tend to retard the RCS depressurization and thereby reduce the safety injection water delivered (due to the relatively higher backpressure). The net result is that slightly higher peak power levels are attained, following the return to criticality during a steam line rupture cooldown.

Since the safety analysis is at the top of the UHI setpoint range for Sequoyah, the analysis is conservative.

15.37
(212.108) The response to Q15.27 provides an analysis of the impact of a 4^oF reduction in core inlet temperature on the FSAR limiting case LOCA. Describe technical specifications which will ensure that the temperature assumption(s) is (are) met.

Response

The position of Westinghouse is that a technical specification on minimum T_{in} is not required. Reasons for this are: 1) ECCS analysis, performed according to 10CFR50 Appendix K requirements, are conservative, and 2) the effect on PCT caused by a variation in T_{in} is small ($<20^o$) and will be outweighed by analysis conservatism. For more information on the effects of T_{in} on PCT, see the Westinghouse response to NRC question 212.144 (212.120).

Sequoyah Nuclear Plant standard technical specifications 3.1.1.4, "Minimum Temperature for Criticality," requires that the reactor coolant system lowest operating loop temperature (Tavg) shall be greater than or equal to 541 F. If the temperature is less than 541 F with the reactor critical, it will be restored to within its limit within 15 minutes or the reactor will be brought to hot standby within the next 15 minutes.

15.38
(212.109) The response to Q15.28 indicates that the worst case, imperfect mixing Cd = 0.6. LOCA analysis, which was run included a reduction in initial gap pressure to correspond to Sequoyah as built fuel parameters. This reduction is identified to be a benefit. Please explain how reduction in gap pressure causes a lower calculated peak cladding temperature. How does burnup come into play in assessing this benefit?

Response

The reduction in initial fuel rod gap pressure has an implicit affect on peak clad temperature through its direct affect on fuel rod burst and ensuring hot channel flow blockage. An earlier time of fuel rod burst results in a longer period of reduced heat transfer immediately downstream of the burst location leading to higher clad temperatures in that region. Depending on the particular transient this may lead to a higher overall peak clad temperature. Burst is calculated to occur if the clad temperature exceeds the burst temperature, where the burst temperature is the clad temperature required to burst the fuel rod at a given clad differential pressure. Based on the functional relationship of burst temperature and fuel rod gap pressure given in WCAP-8301, a reduction in the initial fill pressure tends to increase the transient burst temperature and delay the occurrence of burst. This relationship then leads to a lower calculated peak clad temperature for rods with a lower initial fuel rod fill pressure.

The effect of burnup on peak clad temperature is addressed on a generic basis in WCAP 8963-P-A. This report shows the sensitivity of peak clad temperature to rod internal pressure and fuel burnup. The reduction in initial rod internal pressure does not alter the conclusions of this report and the most limiting fuel conditions for the ECCS evaluation model remain unchanged.

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15.39
(212.110) Your responses to questions 15.28 and 212.120 discussed the sensitivity of LOCA analyses to cold leg accumulator assumptions. Please explain how cold leg accumulator behavior impacts ECCS calculations for perfect and for imperfect mixing cases. Include discussion of the effect of fast/slow accumulator delivery time on peak cladding temperature, and why.

Response

Since cold leg accumulator performance is affected by initial conditions and the geometric description of the system as reflected in ECCS analysis input, a sensitivity to calculated peak clad temperature as predicted by the analysis is expected due to variations in assumed initial conditions. This situation is not unique to cold leg accumulator input but exists for almost all ECCS code input. Since the overall conservatism of the peak clad result is the prime issue for all ECCS calculations, the cold leg accumulator input should be assessed in relation to all other ECCS input and to the total effect of all ECCS input on the conservatism of the calculated peak clad temperature.

The intent of Apendix K is to ensure that margin exists in the ECCS calculation. It accomplishes this by carefully specifying criteria that provide for conservatism in the calculational model. However, Appendix K does not address itself to the requirement for conservatism in plant specific code input. Therefore, a set of input values at "most expected" or nominal plant conditions would represent no change to the margin the ECCS calculation. Applying nominal input would not alter the margin specified in Appendix K.

Deviations from this most expected condition would result in different calculated peak clad temperatures due to various thermal-hydraulic phenomena which can affect both the perfect and imperfect mixing cases to a differing degree. The initial conditions and the geometric description of the cold leg accumulator can be combined in a number of different ways resulting in a range of accumulator empty times. This range of delivery times then, has an associated resulting range in peak clad temperatures.

As stated above, the fast/slow accumulator delivery rates impact the perfect and imperfect mixing cases differently. For the fast delivery rate, the perfect mixing case is impacted through the time required to refill the lower plenum. If, due to the high delivery rate the accumulator empties of water before the filling of the lower plenum, a longer time period is required to fill this region (with safety injection only) before flooding the core. Higher peak clad temperatures will then result due to the lengthening of the core heatup time during this refill period.

The imperfect mixing case is affected quite differently. Because the imperfect mixing blowdown transient is much shorter than the perfect mixing case, there is no possibility that the cold leg accumulators would exhaust prior to lower plenum refill. Due to the overall nature of the imperfect mixing case (i.e. earlier upper head drain time, earlier end of bypass, etc.) the transient response of this high delivery rate is a faster filling of the lower plenum and an earlier bottom of core recovery time. This earlier core recovery time results in a shorter clad heatup period with lower calculated peak clad temperatures.

For the case where cold leg accumulator assumed initial conditions lead tc a longer accumulator delivery period, assurance is provided for both perfect and imperfect mixing cases that the lower plenum and downcomer can be refilled rapidly with cold leg accumulator injection. However, as cold leg accumulator delivery is lengthened, the effect of the steam/water mixing ΔP assumed during accumulator injection in the reflood phase of the transient is enhanced. As required by Appendix K, a conservatively high ΔP is added at the cold leg injection location during the period of cold leg accumulat delivery. This additional ΔP serves as a substantial increase in loop resistance, and results in a significant reduction in calculated flooding rate. As the accumulator delivery period is extended, the resulting reduced core flooding rate is also prolonged; and tends to increase the calculated peak clad temperature.

The contribution of each of the twc effects described above on the calculated pea' clad temperature is dependent on the nature of the individual transient. For perfect mixing cases, the most important consideration is that adequate cold leg accumulator inventory exist at the end of blowdown to ensure refill of the lower plenum and downcomer before core reflood. Therefore, the maximum delivery rate (earliest accumulator empty time) of the cold leg accumulators is of primary interest for this case.

Conversely, for cases where imperfect UHI mixing is assumed, the minimum accumulator delivery rate case (latest accumulator empty time) provides the largest potential impact on calculated peak clad temperature. This is the result of extended application of the cold leg accumulator steam/water mixing ΔP on core reflood rate during the injection period.

15.40
(212.111) Identify provisions (technical specification, etc...) which will assure that cold leg accumulator operating conditions (pressure, water volume, etc...) will be maintained within the range assumed for the sensitivity analyses which have been provided. Identify how and when accumulator levels and pressures will be verified and adjusted.

<u>Response</u>

Technical specification 3.5.1.1 for the Sequoyah Nuclear Plant has the requirements for cold leg accumulator operating pressure, water volume and boron concentration. Surveillance requirement 4.5.1.1.1 lists the requirements for demonstrating the cold leg accumulator operable. These requirements include provisions and the frequency for verifying cold leg accumulator pressure, water volume and boron concentration.

15.41 (212.112) In response to Q15.30 you have provided DECLG break analyses for cases with Cd of 0.4 and 1.0 and a split break with Cd = 0.6. The DECLG analyses assured ?eak factor of 2.32 which is not consistent with the 2.25 factor assumed for other analyses. Justify that these analyses satisfy spectrum requirements and perform DECLG, Cd = 0.8. analyses for both perfect and imperfect mixing to complete the spectrum.

The DECLG break spectrum was performed to show the sensitivity of peak clad temeprature to break size and identify the limiting case break. This spectrum of breaks is typically performed with all breaks at the same peaking factor so that a direct comparison of the break size affect on peak clad temperature is readily apparent. For the Sequoyah plant however, a reduction in peaking factor for the limiting break was required to meet the Appendix K requirement of a peak clad temperature less than 2200^{o}F. A similar reduction in peaking factor was not performed on the balance of the break spectrum since it would only reduce the associated peak clad temperature.

TABLE Q15.41-1

Results	DECLG (C_d=0.8, Per)	DECLG (C_d=0.8, Imp)
Peak Clad Temp. °F	1962	1908
Peak Clad Location Ft.	5.0	7.5
Local Zr/H_2O Rxn (Max.) %	2.3	3.0
Local Zr/H_2O Location Ft.	7.5	7.5
Total Zr/H_2O Rxn %	<0.3	<0.3
Hot Rod Burst Time, Sec.	91.5	67.5
Hot Rod Burst Location, Ft.	5.5	6.5

Calculational Assumptions	
Core Power, Mwt, 102% of	3411
Peak Linear Power, kw/ft, 102%	12.25
Peaking Factor (at License Rating)	2.25
Accumulator Water Volume (cold leg, nominal setpoint value, ft^3 per accumulator)	1078
Accumulator Water Volume (UHI, nominal delivered volume, ft^3 per assumulator, perfect mix)	1056
Accumulator Water Volume (UHI, nominal delivered volume, ft^3 per assumulator, imperfect mix)	900

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15.42
(212.113) Your response to question 15.31 discussed your treatment of statistical uncertainties associated with UHI operation. Please address the following comments about the discussion by our Applied Statistics Branch.

1. The assumption of independence of the five volume errors must be investigated carefully. If the errors are not independent, the consequence of the wrong assumption must be addressed.

2. The size of the Monte Carlo simulation (number of runs) is not stated. Clearly a small size would not be of any value. Also, a documentation of the simulation ought to be made available for review.

3. How are the five errors combined? (The expression used in Sequoyah's response is "combined probabilisically," which is somewhat ambiguous.) If this is a linear combination, how are the coefficients determined?

4. The resultant distribution of the water volume uncertainty claims that water level exceeds (nominal + $25ft^3$) with .05 probability and falls short of (nominal - $25ft^3$) with .05 probability. This suggests a symmetric distribution for the sum of the water volume uncertainties. Now, whereas the sum of normal errors is also normal and, hence, symmetric, the sum of uniformly distributed variables is, generally, neither uniformly distributed nor symmetric.

5. Replace 3 by $\sqrt{3}$ in the fifth line from the bottom, page 1 of Sequoyah's response to question 15.31.

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Response

1. The five volume errors presented in Table 15.4-4e. and associated with the statistical treatment of UHI delivered volume uncertainties are independent in that they result from volumetric error introduced by different and separate pieces of equipment or in calibration of this equipment. A brief explanation of each of the UHI volumetric errors is provided below.

   A. Tank Level Instrumentation Accuracy

   This error is based on the full plus and minus manufacturer's quoted accuracy of the UHI level switches, each of which initiates the closure of its corresponding UHI hydraulic isolation valve.

   B. Tank Volume Tolerance

   The volume of water contained in the UHI system is determined by calculation using the manufacturer's "as built" dimensions of the specific UHI water accumulator and the actual piping drawings of the UHI piping from which water drains during UHI injection. Since the normal specification for tank construction specifies that the contained volume be $\pm$ 1%, and since the UHI contained water volume is approximately 1900 ft^3, a $\pm$ 19 ft^3 error is assumed.

   C. Instrument Setting Tolerance

   The actuation setpoint of the UHI level switches is initially set manually in the field and is periodically verified. The allowable tolerance on variation between the desired trip setpoint and that obtained during the initial adjustment or during verification is specified as $\pm$ 1/4 inch. This tolerance is included as a volumetric error in addition to the level switch accuracy discussed in Item A above.

   D. Hydraulic Isolation Valve Stroking Time

   The variation in hydraulic isolation valve stroking time, due to variations in the nitrogen pressure and nitrogen volume (within the alarm setpoints) of the valve's hydraulic actuator, has been determined. The effect of this variation in closing time (or delivered volume) has been determined assuming all four isolation valves close "fast" of close "slow".

   E. Tank Level Reading Accuracy

   This volumetric variation accounts for inaccuracy in determining the final tank level (the level used to establish the final UHI level setpoint) following the pre-operational test blowdown from full pressure. In actual practice the Sequoyah UHI final level was determined using a high pressure sight glass in which tank level could be directly determined.

From the above discussion it is evident that each of these five UHI volumetric errors are based on independent parameters.

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2. The size of the Monte Carlo simulation is 40,000 trials, sufficient to generate the uncertainty distribution of the overall delivered volume due to the five sources of error discussed in the answer to part 1 of this question.

   The Monte Carlo technique is used to generate statistical variations on the digital computer in a way that is similar to using a large number of trials in a sampling experiment of manufactured parts or of a physical process. Consider an assembled product made up of component parts, each with a specified design tolerance or uncertainty. Given that each component has a statistical distribution, the Monte Carlo model simulates the combining or the functional utilization of the component parts and generates information about the resultant statistical variation of the key item(s) of the assembly after processing the specified number of trials (or histories). This information includes the resulting statistical distribution(s), as well as the estimated values of the mean, standard deviation, skewness and kurtosis.

   The program is formulated in a general way so that the assembly model could be any mathematical function of independent variables having statistical uncertainty. The assembly model selected for the UHI delivery volume is the sum of component uncertainties for each trial.

3. The five sources of variation in UHI delivery associated with measurement error and system performance in Table 15.4-4e. are expressed as maximum contributions to the delivery volume uncertainty. These values are used as the limits for the corresponding uniform distributions. Referring to the discussion of the model given in response to Part 2 of this question, for each Monte Carlo trial a particular value of uncertainty for each of the five components are selected at random from the appropriate uniform distributions and are combined directly as an algebraic sum for the delivery volume uncertainty for that trial. Since the error attributed to each source is expressed as a volumetric error, equal weights are applied to the components. The probability distribution of the delivery volume uncertainty for the UHI accumulator is generated after processing a large number of trials.

4. The probability distribution of the UHI delivered volume uncertainty as obtained from the Monte Carlo trials showed approximate symmetry about the nominal, as indicated by the reported values at 5% probability. This distribution was not uniform, in agreement with the last statement in Part 4 of this question.

5. Refer to the revision of Q15.31.

15.31 Statistical uncertainties - Your discussion of uncertainties that some uncertainties were combined statistically using a Monte Carlo evaluation. Discuss and justify the distributions chosen for these uncertainties and indicate the sensitivity of total UHI delivery to these assumptions.

Response

In the table presented on UHI water volume uncertainties, the five volume errors associated with (1) tank level instrumentation, (2) instrument setting, (3) tank volume, (4) hydraulic isolation valve stroking time, and (5) tank level reading are interpreted as the maximum acceptable tolerance. For example, a tank would be rejected that does not meet the specified volume tolerance.

Among these five sources, some individual errors are expected to be high, others, low. Therefore (excluding the single failure errors), the probability distribution of the UHI Water volume uncertainty results from a statistical combination of the individual component uncertainties.

The distributions of the component errors were assumed to follow a uniform (or rectangular) probability distribution within the limits specified in the table. The individual errors for these five sources were combined probabilitistically using Monte Carlo techniques and assuming independence. The resultant distribution of the water volume uncertainty provides the probability of exceeding a specified range. From this distribution it was ascertained that (excluding single failures) the probability of the water volume being smaller than 25 ft^3 below nominal was less than five percent and the probability of being more than 25 ft^3 above nominal was less than five percent. Adding this range to the uncertainty range associated with single failure modes produces an uncertainty range of 156 ft^3 as indicated in the referenced table.

Note: Two bounding UHI water volume delivery cases are considered in the UHI analysis. For each case the probability of being within the analysis value is 95 percent.

The effects of distributions other than rectangular will now be discussed. If these same five sources of error were each at their respective lower (or upper) limit, then total contribution to UHI volume uncertainty would be 46.2 ft^3 below (or above) nominal. The combined uncertainty range of 92.4 ft^3 is unrealistic since all of the five sources of error will not be at the same limit as the Nonte Carlo results support.

The standard deviation (defined from the second moment of a distribution) of the rectangular distribution equals the half-range divided by the square root of three. If a normal distribution is assumed for each of the five sources of error with a standard deviation equivalent to that of

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the respective rectangular distribution, then the resultant statistical combination is also a normal distribution with standard deviation equal to the square root of the sum of squares of the component standard deviations. The error associated with the five 5 percent tails can be determined from standard tables. Assuming a normal distribution for the five components results in a probability of five percent that the combined uncertainty exceeds 24 ft^3 above nominal, and similarly below nominal. These results approximated those obtained by Monte Carlo for the rectangular distribution. However, in this case the mathematical errors are permitted to exceed the maximum tolerance limits, which would be unrealistic.

A truncated normal distribution could be assumed for the individual sources of error, however, the standard deviation is not uniquely defined. As an example, a standard deviation could be specified that generates a normal distribution truncated by the tolerance range at $\pm 2\sigma$. Since the normal distribution (including truncated) gives a higher probability near the nominal than near the tolerance limits, the use of truncated normal distributions for the five sources of error would result in a water distribution that is more peaked than that obtained from the rectangular distributions. Therefore, the uncertainty range determined above for the rectangular distributions is more conservative than a truncated normal distribution.

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15.43

(212.114) Please submit a discussion to document information provided at the meeting of 3/29/79 and provide any additional information according to your commitment at that meeting.

Response

At the March 29, 1979 meeting, Westinghouse provided details of the basis for establishing steady state operation coolant temperatures used in the loss of coolant accident (LOCA) analysis computer code initialization process. The following is a summary of that information.

The primary objective of the LOCA analysis is to provide assurance that the plant meets the Emergency Core Cooling System acceptance criteria in 10CFR50.46. This objective to be met by performing an analysis which satisfies the requirements of Appendix K (of 10CFR50) and includes suitable conservatism in the plant parameters used in the analysis.

The impact on LOCA/ECCS analysis results of varying the steady state operation primary coolant temperature has recently been reevaluated. In the past, it was assumed (and verified using previous analytical models) that increasing primary coolant temperature resulted in an increase in the calculated peak clad temperature (PCT). However, the recent studies indicate that the sensitivity on PCT of changes in the primary coolant temperatures can vary in both magnitude and direction depending on the type of plant and the break size analyzed. This variation in sensitivity can be explained by the effec that changes in primary coolant temperature has on break flow, upper head temperature, and ultimately on core flow during the blowdown phase of the LOCA transient.

Although this effect on PCT of varying initial coolant temperature is not constant, it is relatively small. The following table shows results of these studies:

Case Analyzed	ΔPCT Per - 1.0°F T_{in}
2 Loop Plant	+2.0
4 Loop Plant - case a	+4.0
- case b	-2.0
4 Loop UHI Plant	+5.0

In performing these sensitivity studies, an important requirement is that the plant be modeled in a steady state operation equilibrium condition. Requirements of 10CFR50 Appendix K include the use of 102 percent of the license power rating and conservative assumptions with respect to pump performance and primary coolant system hydraulic resistance. The assumptions result in a primary coolant flow rate which is much lower than that expected to actually exist. Conservative factors such as these cause the primary coolant temperature distribution to be altered somewhat from that which is actually expected during full power steady state operation.

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Considering, however, that the system must be initialized in an equilibrium condition, determining a sensitivity to changes in core inlet temperature also requires a change in some other system parameter to remain in that equilibrium condition (e.g. RCS flow, steam generator overall heat transfer coefficient, etc.). Therefore, assuring conservatism in some parameters precludes variation of some other parameters if the requirement of steady state equilibrium is to be met. Since the magnitude of conservatism (in terms of ΔPCT) introduced by Appendix K requirements (such as use of 102 percent power) outweighs the effect of likely variations in primary system temperature, it can confidently be stated that the final result is conservative.

The current procedure for performing LOCA/ECCS analysis is to use the nominal, or expected value of core inlet temperature where nominal is determined as follows.

For a given steam pressure, Tav will be calculated based on 100% licensed core power and best estimate RCS flow. Using the Tav output from that calculation, thermal design RCS flow and 102 percent core power, the appropriate steam generator secondary temperature/pressure and the primary system temperature distribution will be calculated. This method of determining steady state operation conditions for the LOCA analysis initialization provides an appropriate value of Tav for use in licensing calculations. That is, the actual primary coolant temperatures measured at full power operation conditions will be sufficiently close to the temperature values assumed in the LOCA analysis such that applying the established PCT sensitivity would not result in a significant (20oF) increase in PCT. That primary coolant temperature comparison will be verified after plant startup and any change in the nominal temperature will be factored into subsequent LOCA analyses.