ENCLOSURE

SEABROOK STATION STEAM GENERATOR TUBE ~~RESPONSE~~ Rupture

DURING SEVERE ACCIDENTS

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DRAFT

Foreword

This report addresses the state of knowledge pertaining to Steam Generator Tube Rupture during postulated severe accidents (approach to core melt and core melt), and the application of this knowledge to the Seabrook Station nuclear power plant. This is an interim report, prepared with the assumption that the work and assessment will continue. The report does not cover all material received from Public Service of New Hampshire (PSNH) and its contractors. It also represents the view of the author and has not been subjected to comprehensive peer evaluation within the NRC.

Nomenclature

AC	Alternating current
BNL	Brookhaven National Laboratory
EPRI	Electric Power Research Institute
ICC	Inadequate core cooling
KW	Kilowatt
LM	Larson Miller parameter
LOCA	Loss of coolant accident
MW	Megawatt
NRC	Nuclear Regulatory Commission
NSSS	Nuclear steam supply system
PDS	Plant damage state (See below)
PORV	Pressure operated relief valve
PRA	Probabilistic Risk Assessment
PSNH	Public Service of New Hampshire
RAI	Request for additional information
RCP	Reactor coolant pump
RCS	Reactor coolant system
RWST	Refueling water storage tank
SG	Steam generator
SGTR	Steam generator tube rupture

Plant damage states are used to classify conditions as follows:

1 Early core melt, low RCS pressure at time of reactor vessel failure, RWST injection not initiated

2 Early core melt, low RCS pressure at time of reactor vessel failure, RWST injection initiated

3 Early core melt, high RCS pressure at time of reactor vessel failure, RWST injection not initiated

4 Early core melt, high RCS pressure at time of reactor vessel failure, RWST injection initiated

5 Late core melt, low RCS pressure at time of reactor vessel failure, RWST injection not initiated

6 Late core melt, low RCS pressure at time of reactor vessel failure, RWST injection initiated

7 Late core melt, high RCS pressure at time of reactor vessel failure, RWST injection not initiated

8 Late core melt, high RCS pressure at time of reactor vessel failure, RWST injection initiated

9 Core melt with non-isolated SGTR

A Containment intact at start of core melt, containment heat and fission product removal available

B Containment intact at start of core melt, containment heat removal only available

C Containment intact at start of core melt, containment fission product removal only available

D Containment intact at start of core melt, none of the containment functions available

E Containment not intact at start of core melt, activity release filtered

F Containment not intact at start of core melt, containment opening larger than three inch diameter

FP Containment not intact at start of core melt, containment opening smaller than three inch diameter

FA Aircraft crash

1. OVERVIEW AND SUMMARY

The Public Service of New Hampshire (PSNH) has presented information to show that the Seabrook Station containment is one of the strongest of any nuclear power plant. It also contains one of the largest volumes. This combination leads to a conclusion that the containment has the capability to either significantly delay or prevent the release of large quantities of radioactive material during and following a severe (core damage or core melt) accident. Based on this premise, any significant risk associated with Seabrook Station would likely be found in accidents which bypass containment.

A number of potential bypass possibilities exist, some of which have traditionally been recognized in Probabilistic Safety Assessments (PRAs), and some of which have not. Historically, conservative assumptions have been applied to those cases which have been recognized, and the conservatism has been assumed to be sufficiently large that the unrecognized possibilities became insignificant since they were believed small in comparison.

PSNH and its contractors have provided a comprehensive PRA with additional follow up investigations in which a better representation of nuclear plant behavior has been attempted. Some conservative, and thereby misleading, representations have been removed. This approach to accident analysis leads to the possibility that when conservatisms have been removed, previously neglected bypass paths which were masked may now be found to contribute to risk. Recognizing this, the Staff and PSNH have explored containment bypass possibilities. One possibility, the topic of this report, and a potential issue that has been under investigation by industry and the Staff for several years, is the loss of steam generator tube integrity due to generation of high temperatures at high pressure during a core melt accident. The potential concern involves movement of high temperature fluid from the region of the melting reactor core into the steam generator tubes, with a resultant overheating of the tubes which leads to their rupture. High pressure fluid containing radioactive material from the melting core would thereby be released to the secondary side of the steam generators, from where it could be released to the environment via the steam generator relief valves, thereby bypassing containment.

For steam generator tube rupture (SGTR) to be a concern as addressed here, one must have a core damage (or melt) condition in progress with no water on the steam generator secondary side. The principal contributor to this condition is estimated to be a loss of all AC power concurrent with a loss of all turbine driven feedwater to the steam generators. PSNH has investigated the possibility of encountering conditions which can contribute to SGTR and has determined the likelihood to be less than 4×10^{-5} per reactor year. This is sufficiently high, and the potential consequences of SGTR under severe accident conditions are sufficiently great, that further investigation has been necessary. This investigation is ongoing. This report provides an interim assessment of the status of the investigation, as well as a projection of expected results.

Study of SGTR due to severe accident conditions is difficult. The phenomena are complex, and most analysis techniques used to investigate nuclear power plant behavior have utilized assumptions which are not applicable here. The principal complication is the multidimensional character of fluid behavior in the reactor coolant system. Suitable computer programs are just beginning to become available. Suitable experimental information is just being developed. Hence, pioneering work, such as provided by PSNH in investigation of this issue, can be expected to have weaknesses as well as strengths. We have found this expectation to be true.

The work reported by PSNH and its contractors is highly informative and addresses most aspects of the SGTR issue. It is based upon knowledge of what takes place within the Nuclear Steam Supply System (NSSS), upon a major computer program that is under development and is being verified (MAAP), and upon information derived from an experimental program at Westinghouse. The following is a summary of the reported information and our assessment:

1. Mathematical modeling. Expected phenomena, experimental information pertinent to the phenomena, and modeling assumptions have been addressed for each of the major components of the NSSS which are affected. Multidimensional fluid flow and energy transport have been established as dominant over most of the conditions of interest. We consider this area

to be in a preliminary stage of development, and there are some potential difficulties, which include:

a. Certain modeling assumptions are overly optimistic. An example is the assumption of complete mixing in the steam generator inlet plenum which tends to reduce the temperature of fluid entering the steam generator tubes. This assumption is not supported by the available experimental evidence, and the effects of the assumption are not balanced by identifiable pessimistic assumptions elsewhere in the analysis.

b. Experimental evidence is preliminary. The experimental facility at Westinghouse is providing information pertinent to this issue. However, testing has been limited to conditions which are only roughly scaled to NSSS representation. This is due to a logical progression in the test planning and facility development. Data from apparently well scaled test conditions are just becoming available. No other test facility addresses certain aspects of this issue.

c. The computer program used as the basis for much of the work has not been verified, nor is documentation available. We understand a verification program and an effort to provide documentation are underway. (PSNH contractors have offered to discuss this information with us. Our review has not progressed to the stage where we can make use this offer.) Although the phenomena we understand to be modeled by the code appear adequate for the purposes needed here, and the code results appear reasonable subject to our concerns as expressed elsewhere in this report, this is not sufficient information to accept the analysis results.

2. <u>Seabrook Station Representation.</u> The basic analyses and sensitivity studies have been based upon a plant configuration in which the NSSS state is assumed. Most of the assumed state conditions are reasonable. There are exceptions. For example, the steam generator secondary side is

assumed to be at a pressure corresponding to secondary side relief valve settings, and creep rupture of tubes is reported for this state. The resulting conclusions are similarly based upon this state. We believe there is sufficient likelihood the secondary side will be depressurized that this case should be considered. Depressurization would roughly double tube stress since the secondary side pressure would be decreased from roughly 1100 psi to atmospheric pressure while the RCS pressure remained at approximately 2300 psi.

3. Sensitivity Studies. PSNH and its contractors have performed a wide ranging sensitivity study as part of an assessment of the impact of various modeling assumptions and the state of the plant. Although this yields valuable information and insight, sensitivity studies should be approached with caution. They are only as good as the basic modeling. The impact of our difficulty with assumptions such as the behavior of the steam generator inlet plenum is not addressed in the sensitivity study, and could impact the results and conclusions.

4. Operator Actions. Plant response can be drastically altered by operator actions during a severe accident. SGTR is no exception. A number of operator responses have been discussed with PSNH. Although many of these were postulated actions, significant information has been developed from these postulations. Recognition that operator actions could depressurize the steam generator secondary side is one item raised during the review. Depressurization of the reactor coolant system via the pressurizer Pressure Operated Relief Valve (PORV) to avoid the SGTR problem is another.

We find that the topic of SGTR is in a developing state, with knowledge being rapidly accumulated. Further work is necessary to conclude that SGTR is unlikely under all conditions associated with a severe accident.

Existing knowledge can be used to support a conclusion that SGTR is not a problem if the RCS is depressurized. Consequently, reasonable assurance that progressions toward core melt would not occur at high RCS pressure, coupled

with supporting evidence in regard to steam generator tube response, would alleviate our concern regarding SGTR under severe accident conditions. We have not conducted an evaluation of the trade-offs associated with such an approach, nor have we been provided with information that would either support or negate RCS depressurization under severe accident conditions. We have not provided a recommendation regarding whether RCS depressurization is attractive when all pertinent factors are considered.

Our judgement is that a carefully conducted thorough evaluation on the part of PSNH can establish that the likelihood that a SGTR will result due to overheating during severe accidents which initiate from power operation is sufficiently small that the risk associated with this event can be shown to be negligible. Our judgement is preliminary and has not been substantiated.

Substantiation of a judgement regarding SGTR under severe accident conditions originating from power operation with the RCS at high pressure can be obtained through a combination of analytic and experimental investigations. The ongoing test at Westinghouse in which reasonably close similitude is claimed between the test facility and appropriate parts of a Westinghouse four loop NSSS will provide key data which can be applied to assist in the development and confirmation of analysis techniques. Use of selected test data from other facilities and further examination of the analysis techniques, coupled with necessary changes when they are uncovered, should provide sufficient confirmation that reasonable reliance can be placed upon accident analyses pertinent to this issue. Suitable analyses can then provide a sufficient foundation to resolve this issue.

2. INTRODUCTION

The Public Service of New Hampshire (PSNH) reporting of Seabrook response to accident conditions in References 1 - 4 represents one of the most comprehensive investigations of nuclear power plant accidents in a specific plant that we have encountered. Some accidents which have a significant impact upon risk are treated more comprehensively than previously reported by any investigator. For example, References 3 and 4 describe an investigation of LOCA outside of containment that is more comprehensive than any we have reviewed. Many of the commonly used conservatisms, which distort the perception of accident impact, have been removed. What results is a serious attempt to better represent plant response to severe accident conditions, with particular attention to items which have previously been identified as having a serious impact upon risk. Paradoxically, as will be seen, this attempt to better represent plant behavior requires a more careful review of certain aspects of severe accidents than required for previously reported PRA investigations.

PSNH has presented information to show that Seabrook Station has one of the strongest containments of any nuclear power plant. It is also one of the largest with respect to containment volume. The combination of large volume and strength leads PSNH to a conclusion that the containment can mitigate virtually every severe accident and, at the worst, can significantly delay release of meaningful quantities of radioactive material during and following core melt accidents. Most core melt accidents can be contained within the Seabrook Station containment, and, if this is accomplished, little radioactive material will escape. The full mitigative capability of the Seabrook containment will be realized if there are no "holes" in the containment. Such holes can exist if any of the following occur:

1. Containment is not properly closed (isolated), such as can occur if containment ventilation is not properly closed upon receipt of a containment isolation signal,

2. A failure occurs which allows the containment atmosphere to escape, such as failure of a containment penetration due to a combination of high pressure and high temperature, or

3. A failure occurs which allows material to move directly from the Nuclear Steam Supply System (NSSS), principally the Reactor Coolant System (RCS), to the environment, such as occurs with the traditional "Event V" (Ref. 5), with LOCA outside containment leading to core melt and the release of radioactive material via the LOCA flow pathway.

Clearly, if PSNH conclusions regarding containment strength are verified, there will be little risk associated with accidents at Seabrook Station unless containment is bypassed. Therefore, core damage accidents with containment bypass deserve careful attention. PSNH has reported studying some bypass accidents in significant detail (Refs. 3, 4, 12, 17, and 18). Such studies have led them to conclude that certain bypass accidents at Seabrook, such as LOCA outside containment, engender significantly less risk than previously believed.

In PRA investigations, one may neglect some small contributors to risk since they are negligible in comparison to major contributors. If major contributors are found to be significantly smaller, then one must check the previously neglected contributors to assure they are still negligible or, conversely, they must be included in the contribution to risk if they are now significant. This is the situation that is typified by the Seabrook Station PRA investigation (Refs. 1-4). A major contributor to risk associated with containment bypass, Event "V", has been analyzed by PSNH, and they have concluded that it is not the significant contributor to risk that it was previously believed to be.

This situation has led us to ask "Are there any containment bypass risk contributors which have been missed or which require further consideration?" One potential area for bypass, as identified above, involves a path between the RCS and the environment. One way of searching for such paths is to ask "Are there any phenomena which may occur, and which have not been adequately addressed in past searches for accident possibilities?" We and PSNH, among others, have asked that question, and found that certain phenomena have been neglected in past PRA investigations because their contribution to risk is small in comparison to other contributors. The phenomena of potential significance involve multidimensional fluid behavior and fission product transport

within the RCS during the approach to core melt and during the core melt process. Consideration of these phenomena has a significant impact upon RCS response, including potentially the location of RCS failure. There are many possible implications, including the possibility that the impact of RCS failure on containment may have been overestimated in past analyses. The implication of interest here is that failure to accurately model RCS fluid and fission product heating behavior might result in an RCS failure which bypasses containment. The only area discovered where this is of immediate concern involves the Steam Generator (SG) tubes. If these fail during a core melt accident while the RCS is at high pressure, there is a high potential of a major release via the SG relief valves or the SG Pressure Operated Relief Valves (PORVs), which vent directly to the environment.

The general concern addressed in this report is the rupture of SG multiple tubes in response to high temperature, which in turn is a result of core uncovery. This accident sequence should be of concern any time there is a core melt with the RCS at high pressure in combination with no water in the SG secondary sides. These conditions lead to a potential for natural circulation transport phenomena to significantly heat the tubes prior to breach of the reactor vessel. If this occurs, the resulting loss of tube strength could lead to tube rupture. If tube rupture occurs, and any of the secondary side valves are open, the secondary side is breached outside containment. Alternatively, if the RCS pressure is above the SG relief valve setpoints, containment is similarly bypassed. This has not been adequately investigated, and is not recognized as a release path in the early Pickard, Lowe and Garrick work on risk investigation at Seabrook Station (Refs. 1-4). It has been addressed in more recent work (Refs. 12, 17, and 18).

The concern was expressed as the rupture of multiple steam generator tubes. We do not believe single tube ruptures will occur under the severe accident conditions of interest. The reason for this is that if one tube ruptures, or even begins to leak significantly, this will induce flow of hot RCS fluid toward the leak. Therefore, the location of tube rupture will probably quickly become hotter. If high temperature is what led to the break, a higher temperature can only make it worse. Tubes in the vicinity of the break will be exposed to the high velocity break flow, in additional to high temperature,

weakening them and, we believe, quickly leading to their failure. We believe this cascading effect would rapidly propagate to multiple tube rupture, stopping only when sufficient RCS depressurization has occurred that tubes are no longer stressed by a significant pressure differential across their walls.

Although this report is limited to SG tube rupture, there are other SG components which separate RCS fluid from the SG secondary side. These components, such as the SG tube sheet, must be investigated to achieve completeness in the investigation of containment bypass via the steam generator.

An initial consideration in investigation of the SG tube rupture issue is "What is the likelihood of attaining conditions where SG tube response could be of concern?" Principally, the conditions are loss of all SG feedwater with a simultaneous loss of RCS makeup capability; conditions which result, for example, from a loss of all AC electrical power with the simultaneous loss of the turbine driven auxiliary feedwater pump. PSNH estimated this condition to have a mean annual frequency of 4.5×10^{-5} per reactor year (Ref. 17), a value sufficiently high that tube response must be considered.

3. STEAM GENERATOR TUBE RUPTURE (SGTR) UNDER SEVERE ACCIDENT CONDITIONS

3.1. Description of Phenomena and Potential Concern.

The RCS is generally modeled with a one dimensional representation of fluid flow, and in some cases with parallel one dimensional modeling in regions such as the reactor vessel. This has been particularly true for PRAs, where to our knowledge, all have been based upon computer code analyses which incorporated single dimensional representations of fluid behavior within the RCS. Additionally, movement of the source of heat due to fission product migration is seldom modeled.

The possibility of RCS behavior being different from what is generally represented during severe accidents has been recognized for some time. Winters (Ref. 6) identified aspects of the problem in 1982 and Denny and Sehgal (Ref. 7) provided preliminary multidimensional analysis results in 1983. It was the subject of an NRC/Industry meeting (Ref. 8), and a formal request for work within NRC (Ref. 9), in 1984. Potential impact upon SGTR was estimated on a preliminary basis (Ref. 10), and experimental data were presented from an ongoing series of tests (Ref. 11), in 1985. Numerous analysis results have been published since the early publications of Denny and Sehgal which represent work sponsored by both industry and the NRC. However, there is no published analysis of overall NSSS response to a broad range of severe accident conditions which includes these phenomena, and which is based upon accident analysis methods which have been subjected to broad peer review and acceptance. This introduces a difficulty into review of SGTR during severe accidents with respect to the impact upon the Seabrook Station risk evaluation. As will be seen, sufficient work has been accomplished that what appear to be reasonable conclusions can be formulated, although confirmation will require additional effort. As will further be seen, there appear to be operational methods which can negate the problem, although the impact on other aspects of plant operation has not been evaluated.

The one dimensional analysis approach appears adequate for approximation of the time to core uncovery following accident initiation. Whether it is adequate to represent behavior following core uncovery depends on a number of

factors, such as the conservatism associated with RCS modeling, system response to accident conditions, and the type of accident. In general, one should question the adequacy if two, and sometimes one, of the following conditions exists:

1. RCS pressure is in the vicinity of normal operating pressure

2. A liquid "plug" exists at RCS low points (the lower reactor vessel or the crossover legs between the SG exit and the Reactor Coolant Pump (RCP) suction connection), and the remainder of the RCS is filled with vapor or gas

3. Accident contributors generally found to be major contributors to risk have a lower probability of occurrence than found in most PRA investigations

In the case of Seabrook Station, all three conditions apply with respect to SG response. A number of conditions potentially lead to core melt with the RCS at high pressure (including, for example, a loss of all AC electrical power with loss of feedwater), such conditions are calculated to leave liquid plugs in RCS low points for Westinghouse designed NSSS's, and the Seabrook Station PRA work represents comprehensive modeling with removal of some unrealistic conservatisms.

The potential misrepresentation of system response of concern here stems from the fluid flow behavior inherent in one dimensional modeling. Such modeling typically represents flow through the reactor core as determined by the water boiloff rate from the lower core or lower plenum. This rate becomes small as the water level approaches the bottom of the core. Typical calculations (see historical references which were previously discussed) indicate that the flow rate due to natural convection which occurs in a multidimensional manner is of the order of ten or more times that of the flow due to boiloff. Hence, the calculations are typically based on a minor contributor to flow, and the major contributor is neglected.

The modeling difficulty also applies to upper plenum behavior. One dimensional modeling of any fluid (liquid, vapor, or gas) that passes through the core is typically assumed to flow through the upper plenum and out the hot leg.

This modeling is incorrect under severe accident conditions where a major portion of the core has been uncovered or the core is being vapor or gas cooled since strong recirculation patterns will develop which thermally link the core and upper plenum. At pressures in the range of 2250 psi, the linkage is strong, and some of the upper plenum component temperatures can be expected to closely follow core temperature during the early stages of the approach to core melt. The strength of the linkage diminishes with decreasing pressure. Information also exists which illustrates a decrease in linkage with increasing hydrogen concentration and core damage (although initial production of hydrogen may enhance circulation due to the buoyant gas "pushing" its way toward upper regions of the reactor vessel).

Similarly, correct consideration of the hot leg and steam generator behavior leads to calculation of significantly different behavior when contrasted to one dimensional modeling. Hot fluid, at a temperature far greater than predicted via a one dimensional model, will enter the upper portion of the hot legs from the reactor vessel, and flow toward the inlet plenum of the steam generators. Displaced colder fluid will return to the reactor vessel upper plenum along the bottom of the hot legs. Circulatory patterns will become established in the steam generator inlet plena in which some of the hot incoming fluid is mixed with plenum fluid. Fluid from the steam generator inlet plena will flow into some of the steam generator tubes in the nominal forward direction, displacing fluid in the steam generator outlet plena. This displaced fluid will flow through other tubes in a nominal reverse direction, reentering the steam generator inlet plena. (All of these flows have been observed experimentally as described in References 11, 13, and 14). This mechanism has the potential to transport hot fluid from the reactor vessel into the steam generator tubes during core heatup and melt, with the result of creating the potential of overheating the tubes if there is no water on the steam generator secondary side.

There are other possibilities which could challenge tube integrity as well. For example, many plant Inadequate Core Cooling (ICC) emergency procedures specify RCP operation if conditions exist which indicate an approach to core melt, and alternate mitigative measures have failed. Such a step could circulate hot fluid through the RCS, including the tubes. Although this may

slightly extend the time to core melt, it may be an unattractive approach if it also introduces a high likelihood of loss of tube integrity. To our knowledge, these contrasting responses and the impact upon risk have not been studied. (Note the likelihood of encountering the situation is small.)

A final phenomenon that has received inadequate attention during conditions leading to core melt is fission product movement. Typical one dimension accident code calculations take such movement into account from the viewpoint of radiological hazard, but do not include the influence upon heat generation. Approximately a quarter of the heat producing radioisotopes probably has left the core under the conditions of interest, and substantial deposits can be expected in the upper plenum structure. This could have a significant influence upon thermal response, particularly if some of this material leaves the reactor vessel and enters the hot legs.

As will be seen in the following sections, PSNH has addressed many of these issues in the most comprehensive study of this problem that we have encountered.

3.2. Seabrook Station Steam Generator Integrity

3.2.1 Issues Addressed By PSNH

The PSNH has addressed many of the issues applicable to SG tube response to severe accident conditions (Refs. 12, 17, and 18). Analysis results were summarized which were intended to determine the thermal response of SG tubes under severe accident conditions. Basic analysis assumptions pertinent to the state of the plant were:

1. The steam generators must be dry to experience a significant thermal transient since, if the SG secondary side contains water, the tubes cannot overheat.

2. Station blackout conditions (Loss of all AC power) exist.

Analyses were conducted for the following:

1. Station blackout without operator actions or RCP seal LOCA

2. Station blackout with a 50 gpm RCP LOCA (each RCP) and no operator actions

3. Station blackout with operator actions

4. Uncertainty evaluation

Possible operator actions considered included:

1. Start steam turbine driven auxiliary feed water flow

2. Restore emergency AC power (diesels and/or switchgear)

3. Shed nonessential loads

4. Open RCS PORVs when core exit temperatures exceed 1200°F.

A number of other operator actions one might expect were discussed during a meeting with the PSNH at BNL on October 17, 1986, including:

5. SG blowdown and depressurization to enable filling the SGs by the condensate booster pumps or from fire water systems. (There are two diesel driven pumps and one electrically driven pump at Seabrook Station. The ability to use these for injection into the SGs has not been confirmed.)

6. RCP operation, a step that is not possible unless off site electrical power has been restored. (PSNH felt the likelihood was sufficiently low that there would be negligible effect on risk.)

3.2.2 Likelihood of Conditions Leading to Tube Failure

PSNH addressed the question of conditions necessary for SGTR in the response to the Staff Request for Additional Information (RAI) 47 (Ref.

17). In this response, PSNH stated the risk to be small for the following reasons:

1. The frequency of high pressure core melt with dry steam generators is very small.

2. Given the postulated occurrence of a high pressure core melt with dry steam generators, creep rupture of the SG tubes is not a credible failure mode.

3. A large number of tubes must fail to produce an early large containment bypass.

4. All three of the following must occur in order for there to be a containment bypass:

   a. Failure to recover water to the SG
   b. Failure to depressurize the RCS
   c. SG tube creep failure

3.2.3 PORV Considerations

PORV operation as identified in item 4, above, is not specifically contained in Seabrook Station emergency procedures, but is believed by PSNH to be a logical operator response as an attempt to depressurize and obtain water from the accumulators. (Operator monitoring of the temperatures is specifically identified in the procedures for loss of all AC power conditions.) In addition to potential core cooling via the accumulator water, opening the PORVs is claimed to have the following effects:

1. It reduces stresses in all primary system components

2. PORV flow overrides natural circulation such that high fluid temperatures are not attained in the SGs, including the tubes.

In response to a staff question, PSNH indicated that the likelihood of being able to open the PORVs under loss of AC and ICC conditions was high (See Section 2.2.9). They also indicated that one PORV was sufficient since its "worth" is about 50 MW of energy removal in the form of steam, and have presented blowdown rate information in Reference 18. (Note Seabrook is equipped with two PORVs.)

Although we consider the EPRI funded Westinghouse tests pertinent to this issue to be somewhat preliminary with respect to scaling to NSSS conditions, some interesting effects have been observed that are worth noting which pertain to PORV operation. These include:

1. Natural circulation flow restores itself readily to the pre-opening condition in the hot legs, core, and communication paths between the upper plenum and the upper head following PORV closure.

2. Heat transfer in steam generators between the primary and secondary side fluids increases 50% to 75% with periodic venting.

3. The core is little affected except for the boundary with the hot leg that connects to the pressurizer surge line.

Item 2 is of particular interest since it carries an implication that flow in the steam generator tubes is enhanced by PORV operation (as well as by opening and closing of RCS safety valves). Hence, if one visualizes opening and closing a pressurizer PORV when degraded conditions are well established with the steam generator secondary side depressurized, there may be a tendency to enhance flow of hot RCS fluid through the tubes, with the potential of causing tube rupture.

3.2.4 Loop Seals

Loss of RCS inventory under natural circulation conditions (RCPs not running) is expected to leave the RCS in a condition where water is trapped at low elevations. According to a number of preliminary analyses. these exist at the cross over leg between the SG exit and the RCP

inlet, and in the lower region of the reactor pressure vessel. The absence of these water seals could significantly change circulatory conditions during ICC conditions, with the potential for changing SG tube response. Although we expect a careful examination of behavior in the Seabrook RCS would establish that the seals will remain under most boil down conditions, this expectation needs to be substantiated by suitable analyses which address the range of conditions which can exist during severe accidents.

Complete loss of the RCS liquid inventory with the RCPs running, followed by loss of the RCPs, could result in a homogeneous fluid condition in the RCS. Under this condition, fluid heated in the core would flow into the upper plenum, through the hot legs, the steam generators, the RCPs, and back into the reactor vessel and the core via the cold legs. Although multidimensional fluid flow conditions probably exist in the reactor vessel after RCPs are lost, one may estimate that thermal response is still reasonably realistic if modeling is restricted to one dimension provided the natural convection flow rates are high. For this case, existing analysis codes could be applied to roughly estimate steam generator tube response. If the response was not clear, then multidimensional analyses could be applied to estimate the influence. In such a case, uncertainty in the multidimensional analyses might not be of as great a concern as for the situation of multidimensional behavior dominating system response. However, nonexistence of the loop seal due to continuous RCP operation is an unlikely situation since the majority of conditions during which steam generator tube integrity is of concern will involve loss of off site AC power, and RCPs will be unavailable. To our knowledge, a complete, accurate, analysis of a four loop Westinghouse NSSS has not been performed for these conditions. In additional to an analysis approach, closure of consideration of this aspect of SG tube behavior could be obtained if the probability of occurrence of the RCS homogeneous fluid condition was established as negligibly small in contrast to other situations where SG tubes were shown to lose integrity, or if the risk associated with the condition was established as negligible when compared to other Seabrook Station risks.

A second situation involving free circulation in the RCS might be obtained if one considers the RCPs as being restarted in response to high core temperatures, as prescribed in the emergency procedures. For this case, sufficient head might be developed to clear the loop seals of water, and rehomogenize the RCS fluid, thereby generating the condition described in the previous paragraph. To our knowledge, rehomogenization under these conditions has not been established to occur at Seabrook. Insofar as SGTR at Seabrook is concerned, the issue can be dealt with as outlined in the previous paragraph.

A third situation of removal of loop seals also potentially exists during boil down of the RCS inventory. One may postulate that the ICC condition occurs with the loop seals in place, and that some other mechanism causes their disruption. This could occur if a sufficient pressure difference occurred across the seals that they were forced out of the low regions. Several analyses have been conducted which include consideration of this behavior, and none showed loss of the seals. One would expect that consideration of this condition could be closed if analyses applicable to Seabrook could reasonably establish that the seals remain.

A final condition can be visualized if one considers a LOCA to have occurred in the RCS. For example, a small cold leg LOCA (or an RCP seal LOCA) could be located between the two natural seal regions of the crossover leg and the reactor vessel lower plenum. Removal of RCS mass might occur under conditions such that the seal water was forced out of the RCS via the break. Elimination of consideration of this effect with respect to impact upon risk could be considered on the basis of a thermal-hydraulic investigation of RCS behavior, establishing that the potential impact on risk of the behavior is negligible in comparison to other established risk contributors, or both.

3.2.5 PSNH Modeling Considerations

The PSNH has reported application of the MAAP 3.0 code to investigation of natural circulation flow in Seabrook (Refs. 12 and 17). This code treats the major phenomena, including approximations of multidimensional

flow and fission product (heating) movement, and is applied to the regions of the RCS which are affected by the SGTR issue.

Quasi-steady momentum balances and continuity equations are used to represent natural circulation flow, and the steam generator inlet plenum behavior is represented by quasi-steady mixing models. The modeling represents gas and wall temperatures using conventional lumped parameter models, with 15 gas control volumes and 17 two dimensional heat sinks. (Several volumes are subdivided into further volumes for some types of calculations. The core, for example, contains 70 nodes which comprise the core volume node.) The control volumes are based upon approximations of the flow patterns which were seen in the Westinghouse experiments on a scaled NSSS (Refs. 11, 13, and 14). This basis for definition of control volumes means that deviations from the assumed flow pattern and flow instabilities may not be represented in the model. Experimental evidence shows that there are asymmetric flow patterns, for example, which are not modeled, and which could lead to tube heating conditions which would not be calculated. Further, although instabilities have not been experimentally observed at the Westinghouse test facility, one must accept this evidence with care since testing with fluid conditions which closely simulate those expected in an NSSS are just being initiated.

Use of the lumped parameter model requires further discussion. Unlike computer codes such as COMMIX, which can determine flow patterns within certain bounds provided the configuration is properly modeled, a lumped parameter model is based more strongly upon a presupposed flow behavior. Although such representation can be valuable and accurate under certain conditions, such assumed behavior must be verified before it can be accepted. The preliminary Westinghouse experiments, as discussed briefly in the next section of this report, and some COMMIX and MELPROG calculations (Refs. 15 and 16), represent steps in this direction, but further evidence is necessary before we can accept the assumption as verified. (The experiments are somewhat preliminary, and the COMMIX and MELPROG calculations have not, to our knowledge, been carefully checked against experimental evidence.) We further note that, to our knowledge, there has been no independent study of the version of the MAAP code used for

the analyses. At a minimum, we believe a reasonable knowledge of code modeling and logic, in addition to a verification program, are necessary for acceptance of the calculated results. (We note that EPRI has a MAAP verification program underway.)

One aspect of the modeling appears worthy of further consideration. The steam generator inlet and outlet plena are assumed to be completely mixed in the PSNH studies being reviewed here, and they are represented by single nodes with uniform properties. The Westinghouse facility test data indicate a partially stratified, partially mixed SG inlet plenum (Ref. 14), and modeling for the test facility is based upon a quasi-steady state model in which partial mixing is assumed at various (limited) locations between streams of different origins. Reference 14 describes the situation as follows:

> "The flow in from the hot leg rises rapidly in a plume in the inlet plenum and induces mixing. Some of the cold return flow from the tube bundle does avoid mixing, particularly near the divider which is furthest from the hot leg. Much of the cold return tubes' flow plunges through the hotter stratified fluid layer that spreads across the bottom of the tube sheet. The mixing flows could be observed from dye injection and from observation of light through the density gradients that resulted. Temperature measurements in the inlet plenum are indicative of mixing. The tubes carrying hot fluid from the inlet plenum were generally concentrated in the area above the hot leg entrance and scattered in the regions further away. Cold return tubes were also scattered and were found in the area above the hot leg inlet also."

Test facility modeling of the phenomena uses a six equation approximation which contains an experimentally determined mixing parameter.

We believe the assumption of complete mixing used for the PSNH investigations will reduce SG tube temperatures when contrasted to the experimentally identified situation. This modeling and its implications need further consideration. (This comment is repeated a number of times in

the discussion of calculated NSSS response in the following sections of this report.)

3.2.6 Comparisons of Calculations to Experimental Data

Several comparisons between MAAP code calculations and experimental data have been briefly described by PSNH and its contractors to the BNL and NRC staffs (Refs. 12, 17, and 18). These are discussed below.

1. Core and upper plenum flow rates. The following comparison of experimental and calculated values was presented:

Test Condition	Experimental Flow Rate	Calculated Flow Rate
28 KW Water Test	0.54	0.50
0.9 KW SF_6 Test	0.016	0.017

2. Hot leg and steam generator natural circulation. Comparison of several parameters was provided:

Item	Experimental Value	Calculated Values for Indicated Number of Steam Generator Tubes Carrying Flow in the Out Direction		
		6	12	24
Heat Transfer Rate, KW	2.43	2.0	2.6	2.9
Entering Fluid, ^oC	30	30.7	29.2	28.4
Exiting Fluid, ^oC	19	24.2	21.7	18.8
Coolant, ^oC	10 - 11	9.4	11.2	12.8

where the entering fluid is flowing into the steam generator inlet plenum from the upper portion of the simulated hot leg, and the exiting fluid is flowing from the lower portion of the steam generator inlet plenum back toward the simulated reactor vessel along the bottom of the hot leg. The

coolant temperature is that of the water leaving the secondary side of the simulated steam generator, and thus, can be related to the heat transfer rate from the primary to the secondary sides.

These results are clearly promising. Continuation of the comparisons with a wide range of experimental conditions in the same test facility, and with no changes in the modeling except for the change of experimental conditions and fluid properties, would be helpful in code verification. Extension of the same modeling approach to other experimental data (such as flow in ducts and components) would provide further confirmation. Completion of confirmation of modeling adequacy could typically include comparisons of existing data obtained in large facilities, selected contrasting of alternate calculational methods to portions of the code under consideration here, and establishment that scaling is adequately represented by the code.

3.2.7 Calculated Seabrook Thermal Response to Severe Accidents

Calculated behavior to selected accident conditions has been summarized by PSNH. Principal results and our comments are as follows:

1. Peak Steam Generator Temperature for Loss of AC Power and Loss of Feed Water Flow. The following temperatures and flow rates were calculated at the indicated condition:

Location	Temperature, ^{o}K	Flow Rate, kg/sec
Core (Peak)	1800	18 (recirculating between upper plenum and core)
Upper Plenum	1160	
Hot Leg	760 (wall)	2.4 (countercurrent)
SG Inlet Plenum	850	-
SG Tube	700 (wall maximum)	3.3 (total in each direction)
SG Outlet Plenum	640	-

PSNH indicated that the hottest core node would melt at about 30 seconds from the time of these values, and that the generated hydrogen and blockage due to relocated core material would cause natural circulation between the core and the upper plenum to almost stop. At this point, the upper plenum would begin to cool due to energy transfer to the hot legs.

Plys (Ref. 18) presents additional information which shows temperatures continue to increase after vessel blowdown, with the peak upper plenum temperature exceeding 1200°K for a short time. The tube temperature continues to increase for the time of the calculation (20,000 sec, with vessel rupture at 11,600 sec), reaching a maximum of about 1020°K. We would be interested in seeing plots of other parameters over the span of the calculations, including the hot leg and SG plena temperatures, to better understand the interactions and modeling.

In response to a question, PSNH indicated they had not performed a detailed analysis of reactor vessel hot leg nozzle thermal behavior, but felt a temperature of the order of 1000°K was necessary to cause failure. Discussion also identified that there was significant steam circulatory flow in the secondary side of the steam generator tubes, and that this steam, which was at a pressure corresponding to the steam generator safety valve settings, represented a significant heat sink. Further, it was an effective medium for transferring heat from hot tubes to colder tubes, thus tending to reduce the maximum tube temperature. This raises a question of what results would be obtained if the steam generators were depressurized to atmospheric pressure, thus maximizing pressure differential across the tubes and simultaneously removing a heat sink which could influence temperatures throughout the NSSS. (A sensitivity analysis was conducted in which this was one of the parameters.)

Information presented in Reference 12 and the above summary table shows fluid flow rates in the hot leg of roughly 2 kg/sec as contrasted with a rate above 3 kg/sec in the SG tubes for the time after effective boiloff of water from the core until melt through of the reactor vessel. Cooling via steam contained in the SG secondary side is thus an effective medium for cooling the SG inlet plenum. The total mixing assumption pertinent

to fluid in the plenum is, in turn, effective in preventing hot fluid from reaching the tubes. This high tube flow rate is also effective in transferring heat from the reactor vessel to the SG secondary side, thus helping to limit fluid temperature in the hot legs as well.

We believe a study would be beneficial of behavior with the SG secondary side depressurized after SG dry out. Now there would be no heat sink on the secondary side, and tube flow rates may be lower due to less of a driving force for natural convection flow in the SG. Further, we would expect to see further stratification in both the hot leg and the SG inlet plenum (the latter not being allowed in the PSNH supported analyses due to the modeling assumption of complete mixing). We pose the question of whether temperatures may be significantly above what was calculated by PSNH and its contractors under these conditions.

2. Operator Induced Depressurization. This calculation was based on the assumption that the operator would open a PORV when the core exit thermocouples indicated 1200°F. The calculations indicated accumulator discharge approximately 1400 sec after opening the PORV, with the RCS depressurized prior to vessel failure. The accumulators were emptied at about 10,600 sec, and vessel failure occurred 2000 sec later. Accumulator water was found to cause a small additional amount of hydrogen production. Phenomena associated with depressurization and hydrogen decreased the effectiveness of heat transfer between the core and other regions of the NSSS. Steam generator inlet plenum temperature reached a peak of roughly 850°K during the depressurization, then cooled, and remained below 650°K for the remainder of the calculation (20,000 sec total calculation time, with PORV opening at approximately 8000 sec). Maximum tube temperature was about 650°K, and was reached at 20,000 sec, being identical to the inlet plenum temperature at that time. (Note RCS pressure is that of the containment following depressurization earlier in the calculation.)

We note that RCS pressure behavior (Ref. 18, Figure 4-4) is different for the base case and the PORV opening case prior to the time of opening of the PORV. We would like to discuss these differences for all parameters

and we would like to understand the reasons they exist. (We note there is little difference in temperature over the range in question, and temperature is the important parameter for the SGTR issue.)

Volatile fission products represent about 20% of the decay heat, and the behavior of this energy source is calculated in the MAAP code. The calculations illustrated movement of the decay heat source. About 10% of the decay heat was associated with fission products which were in the upper plenum at the time of vessel failure. A small amount was in the hot legs, as was also the case for the pressurizer. The amount in the steam generator tubes was not significant. (Most of the CsI was in the upper plenum at the time of vessel failure, with about 10% of the CsI in the hot legs.)

3. Other Variations and Uncertainty. Several sensitivity calculations were performed to obtain a better understanding of behavior. These included:

a. Higher core melt temperature
b. RCP seal failure
c. SG secondary side blowdown
d. Core resistance variation
e. Reduced SG tube circulation
f. Core blockage changes.

These are discussed below.

a. Higher Core melt temperature. A case was run in which core melt temperature was assumed to be 3000°K as contrasted to the base case 2500°K. This was intended to delay the onset of core geometry degradation, which in turn provides more time to heat other portions of the RCS. The 500°K change in melt temperature was found to cause only a few degrees change in SG tube temperatures, which was attributed to the extremely rapid temperature increase rate in the core as melt temperature is approached, and a concomitant small increase in the time available for heat transport to the steam generators.

The model is based upon assumed symmetric behavior, whereas some asymmetries have been found experimentally. If these contributed to a preferential flow of hot fluid near one of the hot legs, that leg might transport hot fluid toward a steam generator and provide higher temperatures than determined in the calculation. This could increase the computed impact of the sensitivity calculation.

A second aspect of the modeling that would act to reduce the calculated impact of the sensitivity run is the assumption of mixing within the steam generator inlet plenum. We believe an assessment of this effect is needed, as previously identified.

b. RCP seal failure. RCP seal failure, if it were to occur, was felt to be a leak in the range of 50 gpm (water) per seal. This was modeled, with the break occurring in all four RCPs at 45 minutes after initiation of the accident. This was found to have an insignificant impact on the results (Refs. 12 and 18).

PSNH also addressed preexisting leaks in SG tubes which are within technical specifications. These were stated to be small in comparison to the 50 gpm flow rate associated with seal leaks, and consequently were argued as being negligible (Ref. 17).

We believe the preexisting leak situation has a negligible impact on NSSS behavior as long as the leak remains small, but do not accept the argument advanced by PSNH as the reason. A comparison of the velocity associated with flow in a tube due to natural circulation with that associated with the leak, with establishing that the latter was negligible, would be more convincing. Similarly, a comparison of flow rate induced by the RCP seal rupture to that expected for natural convection flow would be helpful.

Provision of temperature information pertinent to fluid passing through the RCP seals would be helpful.

c. SG secondary side blowdown. Plys (Ref. 18) reports a calculation to investigate the effect of reduced cooling on the SG secondary side in which the steam generator PORVs are assumed to stick open, thus depleting the secondary side of a high pressure steam atmosphere. Drastic differences were discovered early in the accident due to cooling as the steam generators blew down. Sufficient cooling was provided that the pressurizer emptied due to primary fluid contraction. Reactor vessel failure occurred slightly earlier in this case as contrasted to the base case due to less heat removal from the primary system following removal of the secondary side heat sink. An initial peak in SG inlet plenum temperature of 860°K is identical to that of the base case, but occurs about 500 sec earlier. Following the initial peak, the plenum temperature behavior is similar to the base case, although displaced in time, but is 50 to 100°K higher over the remainder of the transient.

We suggest the calculation be conducted by assuming the PORV is stuck open after all water has been vaporized. This avoids the situation of overcooling associated with the early opening, and may be more compatible with some postulated operator actions associated with late attempts to deal with approaching core melt.

Again, we are concerned with the influence of assumed mixing in the steam generator inlet plenum and the impact upon calculated results.

d. Core resistance variation. Variation of the resistance of the core to flow was evaluated by lowering the axial and cross flow core friction factors in one calculation. This slightly increased heat transfer to the steam generators and correspondingly increased time to vessel failure. There was a slight tube temperature increase, but in general, the calculation showed little sensitivity of tube temperature to the change in core friction factors.

e. Reduced SG tube circulation. Selection of lower limit values of the number of steam generator tubes participating in flow from the inlet to the outlet plena was used for another sensitivity calculation.

This provided lower values of steam generator natural circulation flow relative to the hot leg natural circulation flow rate, and reduced cooling of the steam generator inlet plenum due to flow from the outlet plenum. Slightly less heat was removed from the reactor vessel due to the lowered flow rates, and vessel failure occurred slightly earlier. These changes were insignificant. However, the steam generator inlet plenum was found to be about 150°K higher than for the base case, reaching a temperature of 980°K for a short time. Steam generator tube temperature was relatively unaffected.

Comparison of inlet plenum and tube temperature transient behavior (References 17 and 18's Figures 4-11 and 4-12) appears to indicate a significant thermal inertial associated with the tubes, which do not increase in temperature to a significant degree in contrast to the temperature of the source fluid in the steam generator inlet plenum. We believe this needs further discussion. For example, what is the location of the tube temperature and does this location correspond to the highest tube temperature?

Again, as previously stated, the influence of the assumption of complete mixing in the steam generator inlet plenum will impact the results. A portion of the concern is that reduced flow rates may lead to greater stratification and less mixing in the SG plena, a phenomenon that is not modeled in the PSNH reported evaluations, and a phenomenon with the potential to increase tube temperatures over what was reported.

f. Core blockage. In this calculation, a delay of blockage in the core at the time of core melt to the time the node was completely filled with refrozen eutectic was assumed. This was done to continue core oxidation and core/upper plenum flow for a longer time. For this case, the maximum sustained SG inlet plenum temperature is roughly 1060°K, with a short time (less than 50 seconds) temperature "spike" to about 1120°K.

We again reiterate the concern with SG inlet plenum modeling and its impact upon the results.

g. Sensitivity Summary. An approximate comparison of the results of the sensitivity study is provided in Figure 1. The major early effect on increased tube temperature is due to changing the SG tube flow characteristics. Later, and with the greatest impact, is the effect of delaying formation of blockage in the core, which allows continued circulation of hot fluid through the core where the temperature is increased, as opposed to a drastic reduction in heat transport between the core and other RCS components when a core geometry change occurs.

4. Steam Generator Tube Strength. Plys, in Reference 18, Appendix B, addresses SG tube integrity. The presentation is based upon the SG secondary side pressure being at the SG safety or relief valve setpoints which, as previously discussed, may not be the case. We note that Plys identifies nominal hoop stresses of 9300 to 10000 psi for the assumed conditions. Hence, the case of the SG secondary being depressurized will result in a nominal hoop stress of roughly 19,000 psi. This stress, substituted into Reference 18's Figure B-6, results in a Larson Miller parameter of about 37. The Larson Miller parameter is defined as:

$$LM = T(20+\log t_r) \times 10^{-3}$$

where:

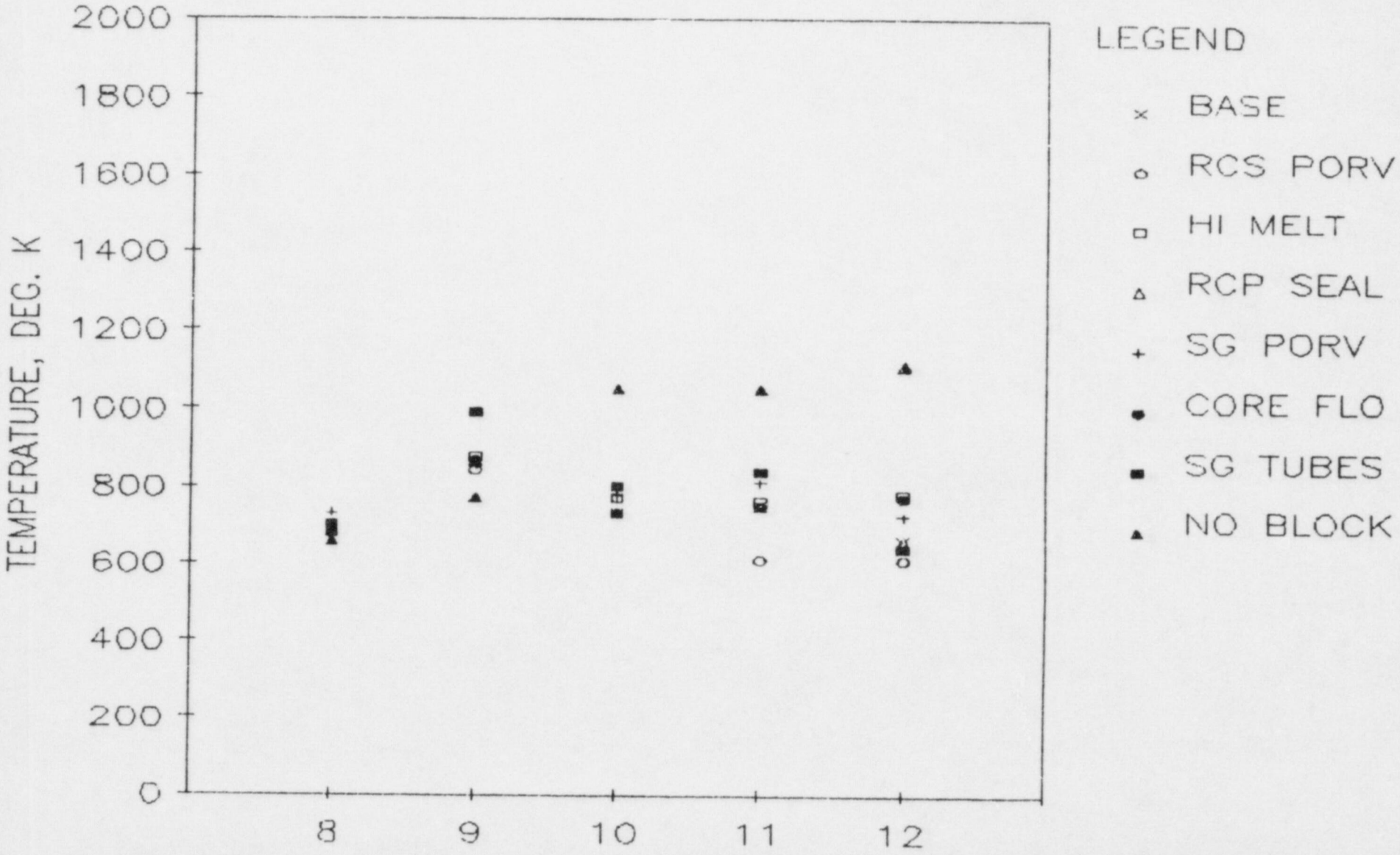
T = temperature, ^{o}R

t_r = time to rupture, hrs.

Substituting a temperature of $1090^{o}K$ (the value used by Plys to conclude the rupture time would be greater than 2.5 hrs) yields a time to rupture of about 5 minutes, a significant change from the Plys value.

Plys could have selected $1090^{o}K$ as conservative, with no need to consider an alternate since the no tube rupture position was supported by the

FIG. 1 STEAM GENERATOR INLET PLENUM TEMPERATURE COMPARISONS



result. If we recognize this possibility, and select a less conservative $1000^{o}K$, we find a rupture time of about 3.5 hours. These temperatures can be contrasted to the SG inlet plenum temperatures provided in Figure 1, with recognition that these are not tube temperatures, but also with the recognition that some of the parameters contributing to the temperatures remain to be evaluated.

Clearly, we are in a temperature region where relatively small changes have a significant impact upon creep rupture time. Equally clearly, tube stress could be roughly a factor of two higher than the value used to justify that tubes would not rupture. We conclude the picture is not as clear as presented in Reference 18, which presented a conclusion that tubes would not be ruptured.

3.2.8. Other Considerations

In Reference 17, PSNH stated that if one postulated creep rupture failure of steam generator tubes, the pressure inside the previously dried out and isolated steam generator secondary side would increase until the steam generator PORVs setpoint was reached, at which time the valves would lift and modulate until reactor vessel melt through and RCS depressurization into the containment. During the periods of SG PORV opening, there would be a high leak rate bypass condition directly from the RCS to outside the containment. They further stated that after vessel melt through, the leak rate out this path would be low and would correspond to any low pressure leakage through the reclosed PORV. They note this leak path could be enhanced if the SG safety valves also lift and fail to reseat properly; however, they believe it unlikely that the safety valve setpoint would be reached.

As previously discussed, we do not believe an individual tube would rupture, but instead believe there would be a massive failure in one steam generator. (Once the failure initiated, we would expect the RCS to depressurize rapidly, which would reduce stress on tubes in other steam generators.) It is difficult to postulate a PORV modulating this condition. It is further difficult to postulate the PORV or the safety valves

would not be damaged when exposed to these conditions, and therefore their reclosing may be questionable. Finally, if the conditions which led to the accident sequence involve a loss of all AC power, which is one of the likely situations given a severe accident scenario, we pose the question of how long the PORVs can be expected to modulate pressure assuming they are not damaged by the fluid being modulated.

Plys (Ref. 18) has identified that the MAAP code does not model certain aspects of SG tube temperature, and a method of obtaining temperature was discussed. Aside from the impact of secondary side steam as a cooling medium, we are concerned about local heating due to small leaks. Such a leak could cause a small amount of hot fluid to pass through a localized area into the SG secondary side, with different heat transfer characteristics and tube temperatures than one would encounter with the treatment of overall inside to outside heat flow utilized by Plys in their estimation. Whether this is important to localized tube temperature over a sufficient area to be of concern should be addressed. (Note the effect could also be concentrated in an adjoining tube. This can be visualized by picturing a tube with a small hole which directs hot RCS fluid onto the secondary side surface of an adjoining tube, while the inside surface of that same tube is exposed to hot RCS fluid.)

3.3 Accident Likelihood

PSNH has estimated the mean annual frequency of accidents in which the core melts with the RCS at high pressure and the SGs dry as bounded by a value of 4.5×10^{-5} per reactor year (Ref. 17). This is composed of the following plant damage states:

DRAFT

Plant Damage State (PDS)	Mean Annual Frequency
3D	1.5×10^{-5}
3FP	8.9×10^{-6}
4A	1.4×10^{-5}
4C	1.7×10^{-7}
4D	2.8×10^{-6}
4E	2.2×10^{-11}
4FP	1.2×10^{-7}
8A	3.9×10^{-6}
Total	4.5×10^{-5}

The accident sequences which comprise the PDSs include transient and loss of off site power sequences with failure of all emergency feedwater, failure of feed and bleed with loss of all emergency feedwater, and transients without scram. PDS 8A consists of eight sequences which involve station blackout and emergency feed water failure with recovery of containment heat removal.

PSNH also addresses the potential impact of tube rupture on this information. They have assigned a high chance of no containment failure to PDSs A. PDSs C and D are considered as leading to a high likelihood of long term containment overpressure failure. PDSs FP are a high chance of small bypass, and PDS E is a high chance of large bypass. Hence, PDSs A, C, and D would be impacted by SGTR, and FP may represent some impact. Addition of the appropriate values indicates that the likelihood of being in a condition where SGTR could affect the results is about 4×10^{-5} (as contrasted to the assumption of no SGTR).

PSNH considers these values to be bounding because some of the values include states with water on the steam generator secondary side, for which SGTR is not a concern, certain operator recovery actions have been neglected, and RCS depressurizations prior to core melt have not been considered. As previously discussed, operator depressurization is one of the potential steps which one could consider to mitigate SGTR. PSNH estimates the frequence of operator failure to depressurize as less than 10^{-2} to 10^{-3} per demand, provided procedures are modified and adequate operator training is provided. These values lead to a conclusion that the frequency of obtaining conditions under which SGTR would be of concern can be reduced to of the order of 10^{-7} to 10^{-8} per reactor year.

Although these values appear reasonable, we note that the conditions which led to the factor of 10^{-2} to 10^{-3} reduction do not presently exist. We further would need substantiation for these values prior to acceptance.

Discussion is also provided concerning the likelihood of SGTR if exposed to high pressure core melt conditions (Ref. 17). PSNH points out that their calculations show SG tube temperatures that are roughly 200 to 300°F below what would be required for creep rupture, and this is identified as principally due to cooling by steam on the SG secondary side. Several things are necessary for acceptance of the tube temperature conclusions, including, as discussed elsewhere, substantiation of the calculational technique and investigation of the likelihood of the SG secondary side having a significant steam inventory (which also means having a significant pressure).

Finally, PSNH estimates a 99% chance that failure of SG tubes will not occur before reactor vessel melt through or piping nozzle failure. This value, combined with the prior estimates of frequencies, appears sufficient to establish that SGTR is not of concern as a significant contributor to risk. Therefore, one can reasonably anticipate that substantiation of the various items which led to the conclusion, as discussed in this communication, should provide substantiation of the above preliminary conclusion.

3.4 Additional Reviewer Observations

A number of observations and comments have been made in the previous discussion. We offer the following additional comments:

1. Much of the modeling utilized in the calculations has not been documented. We understand this is underway. Such documentation will be helpful in the continuation of the review.

2. The outside of the hot legs is assumed to be adiabatic. This probably introduces a small conservatism into the results with respect to hot leg temperature. The impact on other parameters is probably negligible. With respect to the hot legs, the parameter of interest may involve a relatively thin wall connecting pipe that is exposed to high fluid

temperature, and whose temperature will follow fluid temperature more closely than is the case with the relatively massive hot leg; or the vessel nozzle region of the hot leg, which will be more closely allied with fluid circulating rapidly within the upper plenum. Thermal response of these regions may be critical in determination of the failure point of the RCS pressure boundary.

3. Although the limited experimental evidence reveals some symmetry in flow behavior within the reactor vessel, there are also unsymmetrical flows and temperatures. We understand the MAAP calculations are based upon modeling the upper plenum fluid as a single volume. This appears to be a nonconservative approach.

4. STEAM GENERATOR TUBE RUPTURE CONCLUSIONS

The above discussed considerations lead us to the conclusion that this topic is in a developing state, with knowledge being rapidly accumulated. Insufficient information is presently available for one to conclude that SGTR cannot occur as a result of severe accident conditions.

Our judgement, at this juncture, is that a carefully conducted and thorough evaluation on the part of PSNH, that utilizes information which either exists or will be available within the near future, can establish that the likelihood is small that a SGTR will result due to overheating during severe accidents. Further, our judgement is that the risk associated with SGTR can be shown to be negligible for these conditions. Our judgement needs to be substantiated. We have encountered too many unanswered questions, unsubstantiated assumptions, and potential conditions which could lead to calculation of increased temperature to accept a conclusion that SGTR will not occur under circumstances such that the associated risk can be neglected. We note, as a qualifier to these conclusions, that our review is not complete, and, in addition, work is ongoing to provide further information.

Existing knowledge would support a conclusion that SGTR is not a problem if the RCS is depressurized. Consequently, reasonable assurance that progressions toward core melt would not occur at high RCS pressure, coupled with suitable technical backup for a conclusion that low pressure is not of concern, would eliminate our concern regarding SGTR under severe accident conditions. We have not conducted an evaluation of the trade-offs associated with such an approach, nor have we been provided with information that would either support or negate RCS depressurization under severe accident conditions. We have not provided a recommendation regarding whether RCS depressurization is attractive when all pertinent factors are considered.

Substantiation of a judgement that SGTR is not a concern under severe accident conditions with the RCS at high pressure can be obtained through a combination of analytic and experimental investigations. The ongoing test at Westinghouse in which reasonably close similitude is claimed between the test facility and appropriate parts of a Westinghouse four loop NSSS should provide key data which can be applied to assist in the confirmation of analysis techniques. Selected test data from other facilities and further examination of the analysis techniques, coupled with necessary changes when they are uncovered, should provide sufficient confirmation that reasonable reliance can be placed upon accident analyses pertinent to this issue. Application of a reliable analysis technique to issue investigation should then provide the necessary background to resolve this issue.

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