MALUK

STEAM GENERATOR TUBE INTEGRITY

UNITED STATES NUCLEAR REGULATORY COMMISSION

December 7, 2000

Presentation to

Commissioner Nils J. Diaz

Commissioner Edward McGaffigan, Jr.

J/139

Presenters

Item	
5	Steven M. Long, Division of Systems Safety and Analysis, NRR (301) 415-1077
	Steven Arndt, Division of Safety Analysis and Regulatory Effectiveness, RES (301) 415-6502
6	John J. Hayes, Division of Systems Safety and Analysis, NRR (301) 415-3167
7	William M. Dean, Division of Inspection Program Management, NRR (301)414-1257
8	Robert G. Schaaf, Division of Licensing and Project Management, NRR (301) 415-1312
9	Richard B. Ennis, Division of Licensing and Project Management, NRR (301) 415-1420
	Edmund J. Sullivan, Jr., Division of Engineering, NRR (301) 415-2795

Risk Analysis for

Steam Generator Tube Degradation

(Agenda Item 5)

Accident Sequences Affected by SG Degradation

Risk is calculated for accidents sequences that result in substantial core damage.

Focus of the discussion is core damage with containment bypass due to SG tube failure

Sequences addressed:

Sequences initiated by spontaneous rupture of a degraded tube

Sequences initiated by secondary depressurization that induce rupture(s) of degraded tubes due to increased pressure differential

Sequences initiated by primary system over-pressurization that induce rupture(s) of degraded tubes due to increased pressure differential

Sequences that cause core damage by mechanisms unrelated to tube degradation, but which include conditions that would lead to containment bypass by inducing rupture of degraded tubes due to

increased pressure differential

increased tube temperatures

Severe Accidents Due to Spontaneous Steam Generator Tube Rupture

Not included in original Reactor Safety Study (WASH-1400)

First rupture event at Point Beach unit 1 in 1975 brought it into subsequent PRAs

Included in all IPEs with widely varying results:

Initiating event frequencies all about 10^{-2}/RY

CDF contributions vary from high 10^{-6} to low 10^{-9}/RY

Reasons for variations not fully understood by NRC

Dominant sequences for high results involve human error probabilities.

Dominant sequences for low results are usually hardware failures with non-recovery.

Human error probability estimation appears to be a major cause of differences.

These sequences are usually one of the dominant contributors to public health consequence results in current PRAs

Accidents Initiated By Secondary Depressurization Events

Potential Initiators:

- stuck MSL SVs
- steam dump control problems
- spontaneous breaks in MSL and associated valve headers
- SG overfills resulting in MSL or valve header breaks

Conditional probability of tube rupture depends on probability that susceptible flaw(s) will be in free-span of tubes in affected generator(s)

Successful mitigation requires cool down to cold shutdown unless secondary integrity can be restored.

Human errors are most important failures in dominant cutsets.

Mitigation of about 10 full ruptures appears possible, but human errors become much more probable as limit is reached.

We are not currently aware of mechanisms that appear to have a significant probability of creating such a large number of ruptures.

Accidents Initiated By Primary Over-Pressurization Events

Initiator : ATWS

ATWS events that exceed 3200 psi in primary system are usually considered to be core damage accidents, but this produces a tube ΔP of only about 2200 psi which is less than MSLB ΔP

For ATWS events reaching lower pressures, PI-SGTR is expected to have low conditional probabilities.

This makes potential complication of ruptures in multiple generators for the same event unlikely.

So, mitigation is expected to be similar to spontaneous tube rupture, following success in dealing with ATWS event.

For core damage events due to ATWS exceeding 3200 psi in the RCS, conditional probability of PI-SGTR creating containment bypass ranges from less than conditional probability of PI-SGTR during MSLB to a higher value, depending on ATWS peak pressure.

We do not have conditional probability curves for RCS pressure during ATWS for currently installed cores.

Rupture of tubes reduces ATWS pressure peak.

Due to low ATWS frequency and tube integrity requirements, ATWS is not estimated to make a substantial contribution to increased CDF or LERF due to induced SGTR

Core Damage Accidents That Induce Tube Rupture

Any core damage accident with the RCS at relatively high pressure and the SGs dry:

station blackout
loss of DC buses
small LOCAs with loss of secondary cooling

Concerns for inducing SGTR:

loss of secondary integrity causing rupture by increasing ΔP across tubes

transfer of heat from damaged core to tubes weakening tubes

Event trees for these sequences need to be expanded to account for various events that control the timing of primary and individual SG secondary depressurization and the rate of introduction of heat to the tubes and other RCS components

Core Damage Accidents That Induce Tube Rupture (continued)

Factors to incorporate include:

RCS leakage from RCP seals, SG tubes, Pressurizer valves partially or fully stuck open, etc.

SG leakage from stuck MSL SVs, TDAFW supply, MSIVs, MSIV bypass valves, blowdown valves and MFW check valves

Thermodynamics of RCS component heatup, including loop seal and downcomer skirt clearing, counter-current circulation flow mixing, etc.

Potential for tube failures depends on probability of occurrence of flawed tubes that cannot withstand conditions that occur during the various sequences.

ΔP may vary between generators

Temperatures may vary within generators

Risk Metrics

Steam generator degradation can have a major effect on the estimate of a plant's public risk contribution with only a minor effect on the plant's CDF, because releases (usually) bypass containment.

Depending on definition of LERF, actual core damage source terms with SGTR may or may not exactly meet definition.

However, they are expected to be orders of magnitude closer to LERF than to contained core damage event source terms. So they are treated as LERF in the regulatory process.

Use of full PRA level 3 consequence calculations for the various SGTR sequences is not yet feasible, due to the current lack of ability to predict accurately some physical phenomena of the accident progression.

Effects of RCS blowdown through faulted SG during core damage are not yet evaluated:

- tube temperatures and additional creep damage
- jet heating and/or cutting damage
- primary-to-secondary flow rates as a function of time
- secondary side flow rates and velocities
- secondary structure temperatures
- effects of radionuclide deposition before release

So, source term for SGTR sequences not yet refined

Risk Metrics

Criteria for acceptance of consequences not clear, either.

Safety Goal numerical objectives apply only to close-in populations (1 mile and 10 miles)

But bulk of health consequences occur at greater distances, especially if evacuation is credited.

Integrated Decision Process

RG-1.174 delineates 5 "principles" to be addressed in making a risk-informed decision

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or a rule change.

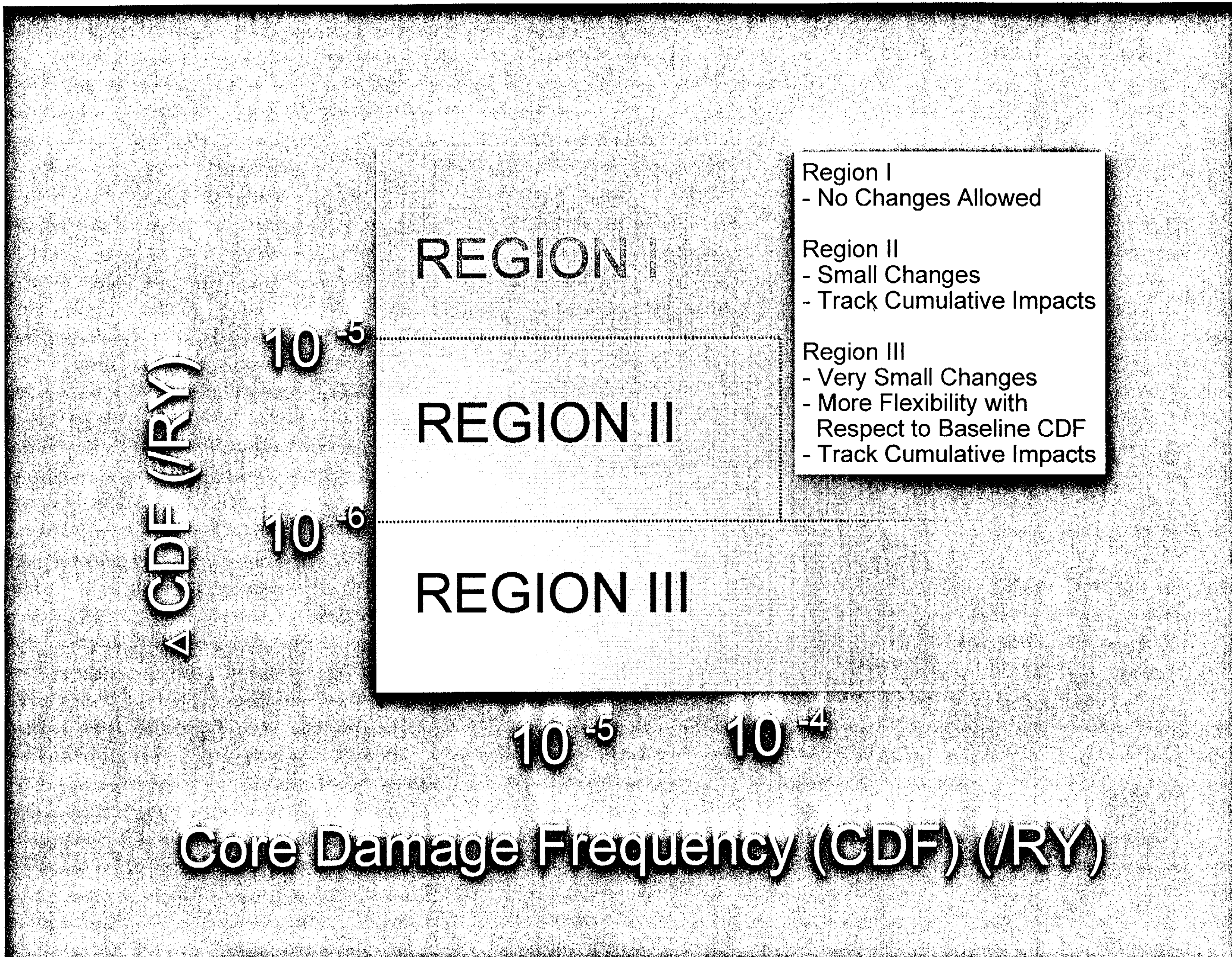
2. The proposed change is consistent with the defense in depth philosophy.

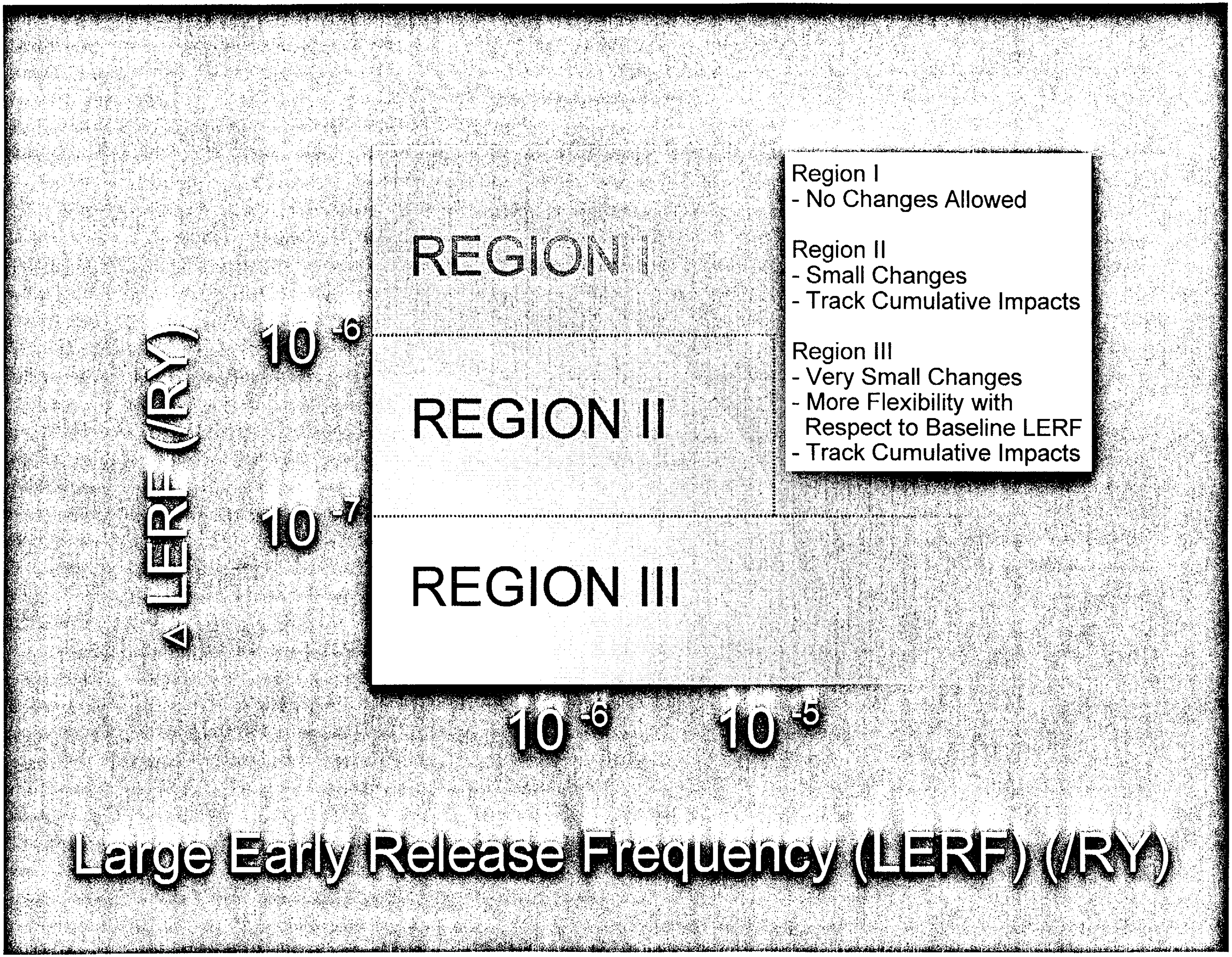
3. The proposed change maintains sufficient safety margins.

4. When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

5. The impacts of the proposed change should be monitored using performance measurement strategies.

PLUS: Consideration of uncertainties and their potential effects on the decision.


Region I
- No Changes Allowed
Region II
- Small Changes
- Track Cumulative Impacts
Region III
- Very Small Changes
- More Flexibility with Respect to Baseline CDF
- Track Cumulative Impacts
REGION I
REGION II
REGION III
10 -5
10 -6
Δ CDF (/RY)
10 -5
10 -4
Core Damage Frequency (CDF) (/RY)


Region I
- No Changes Allowed
Region II
- Small Changes
- Track Cumulative Impacts
Region III
- Very Small Changes
- More Flexibility with Respect to Baseline LERF
- Track Cumulative Impacts
REGION I
REGION II
REGION III
10 -6
10 -7
Δ LERF (/RY)
10 -6
10 -5
Large Early Release Frequency (LERF) (/RY)

Analyses Required to Quantify Risk

Initiating event frequencies

Conditional probabilities

Thermal-hydraulic modeling

Human error probability estimation

Initiating Event Probabilities

Spontaneous tube rupture frequency is taken from experience.

Steam-side depressurization events have not been tabulated, especially with respect to the maximum ΔP experienced.

ATWS sequence peak ΔP frequencies have not been developed.

"High/dry" fractions of CDF have not been determined for most PRAs, and the logic is not set up to make retrospective determination simple.

Conditional Probabilities

In addition to the usual equipment failure probabilities routinely used in PRAs, modeling of risk as a function of SG tube degradation raises additional equipment issues that are not addressed by current data:

Steam leak rates from SGs are not measured, but are important in determining if a dry SG will depressurize.

Steam lines are not designed for flooding and steam line relief valves are not designed to discharge saturated water, but conditional failure probabilities are not available for over-fill conditions.

Definition of Human Failure Events

Current PRA practice is to define human failure events associated with those responses that, if not performed in a timely manner, lead to a failure of a function or system required to respond to an initiating event

The consequences of performing actions that have a detrimental impact on mitigation are typically not modeled

The plant conditions implied by the accident sequence provide the contextual definition of the HFE (cues, time available, equipment available, etc.)

Quantification of Human Error Probabilities

There is no consensus on how to estimate the probabilities of human failure events

In general, operator performance is recognized as being a function of a number of factors, including

a. clarity and definitiveness of indications
b. time available for completion of response
c. experience and/or training
d. procedural guidance and completeness
e. plant ergonomics

Methods differ in how these factors are addressed and there is significant variability between results from different methods

There is little, if any, actuarial data to calibrate the probabilities

ATHEANA Project

Additional Human Reliability Analysis of SGTR events will be done using "A Technique for Human Event Analysis" (ATHEANA)

- ATHEANA provides a improved method for dealing with "Error Force Contexts" (EFCs) into Human Reliability Analysis.
    - ATHEANA provides more realistic identification and modeling of errors of commission and dependencies.
    - ATHEANA provides a structure for understanding how unexpected plant conditions and unfavorable performance shaping factors (PSFs) can effect error rates.
- ATHEANA has been developed at Sandia National Laboratories and is currently being demonstrated at part of the NRC human reliability assessment of the pressurized thermal shock (PTS) reevaluation project.
- The SGTR ATHEANA analysis is currently planed to be conducted at the Idaho National Engineering and Environmental Laboratories use personnel that have been training in ATHEANA and are familiar with reactor human reliability analysis. This work will be starting the latter haft of FY2001.

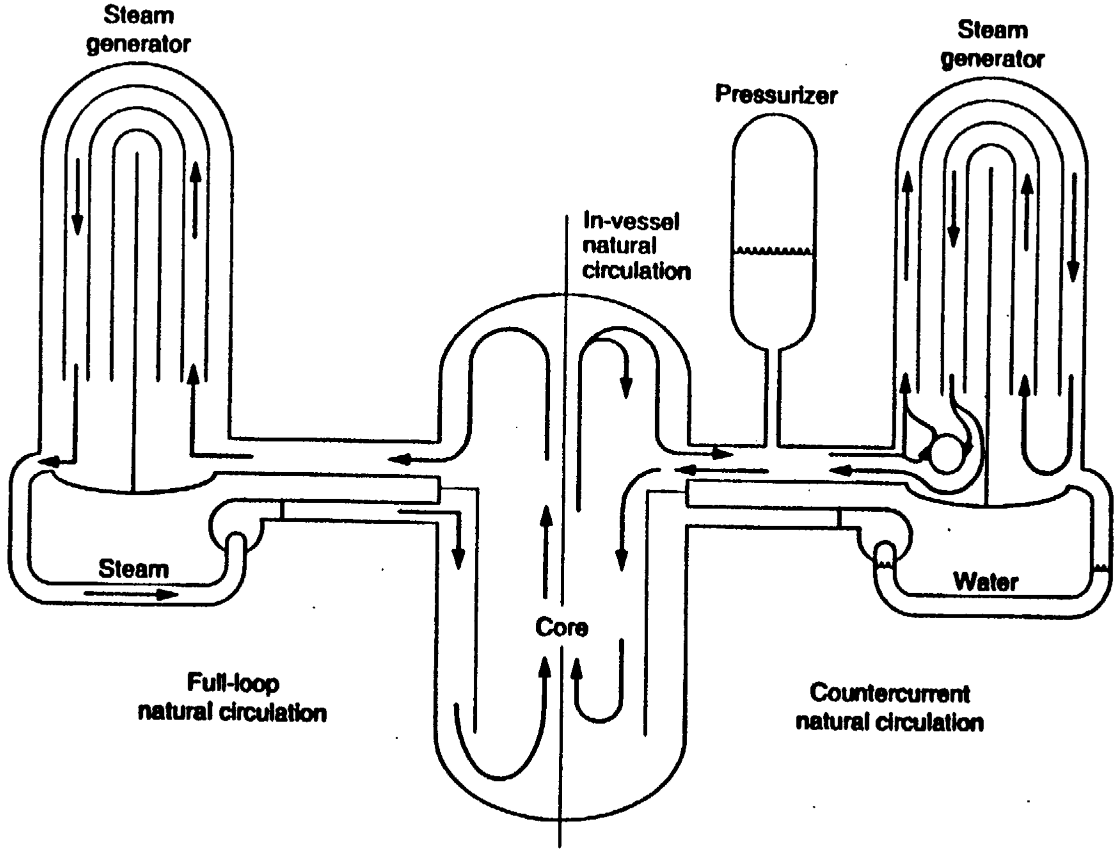
Thermal-Hydraulic Calculations for Core Damage Accident Progression and Prediction of Phenomena Affecting Tubes

Steve Arndt

Item 5.4

Example Scenario

- Station Blackout (TMLB')
- Initiated by the loss of off-site power, with the unavailability of on-site AC power.
- Main feedwater valves and turbine stop valves close, effectively isolating the SG secondaries.
- SG pressures increases until relief valves open. SG pressure is maintained between opening and closing pressure for the relief values. One SG relief valve fails open on first challenge.
- As water in the secondaries is boiled away the SG's no longer remove significant amounts of heat and the RCS heats, resulting in system pressurization controlled by cycling pressurizer SRV's
- After the RCS is heated to saturated conditions, a high pressure boiloff begins, leading to core uncovery and heatup.



Natural circulation flow patterns that can develop during severe accidents in PWRs

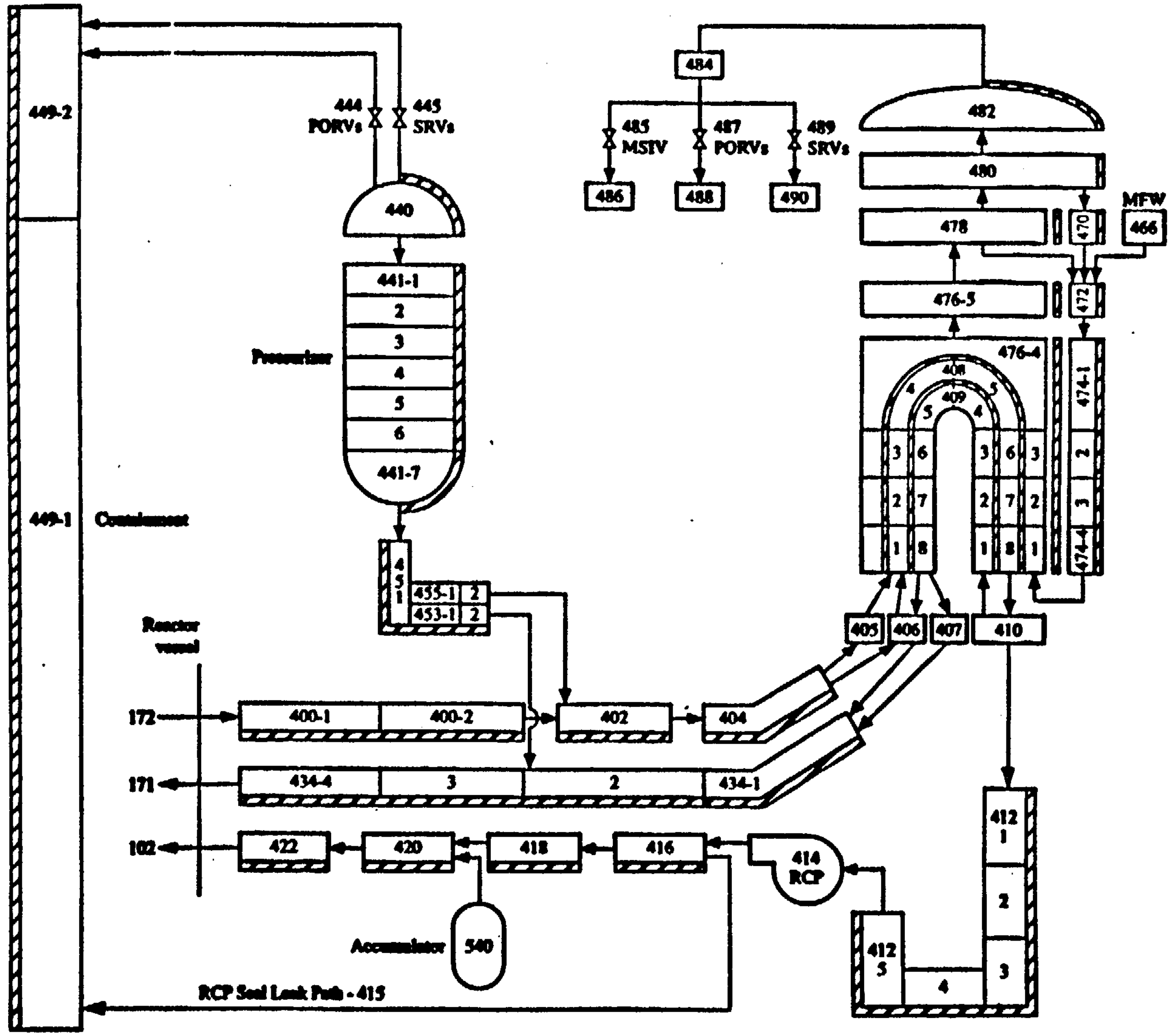
Summary

- Analysis using SCDAP/RELAP5 have been performed for representative plants for potentially risk-significant scenarios (high pressure TMLB' sequences with depressurized secondary side) to estimate effects of high temp fluid circulation.
  - SR5 analyses predict failure of hot leg or surge line before unflawed SG tubes.
  - Sensitivities on T-H Modeling did not alter conclusion on tube integrity but margins are relatively small.

Example Calculation

Application of SR5

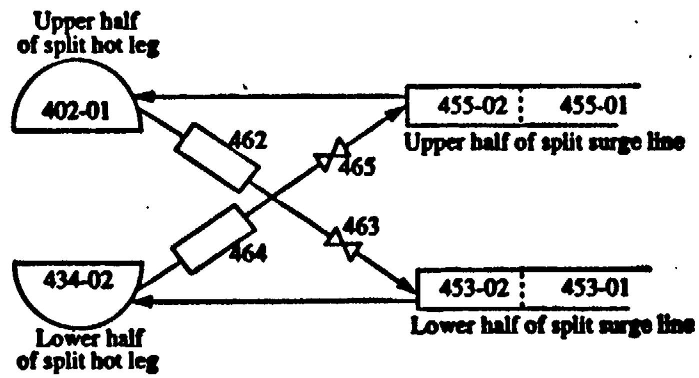
- Surry Plant calculation
  - TMLB' transient with SG secondary side depressurization.
  - Base case: #SG tubes participating in forward flow → 53%.
    mixing fraction = 0.87
    recirculation ratio = 1.9



Surry pressurizer loop (Loop C) nodalization with provisions for hot leg countercurrent natural circulation.

Steve Arndt

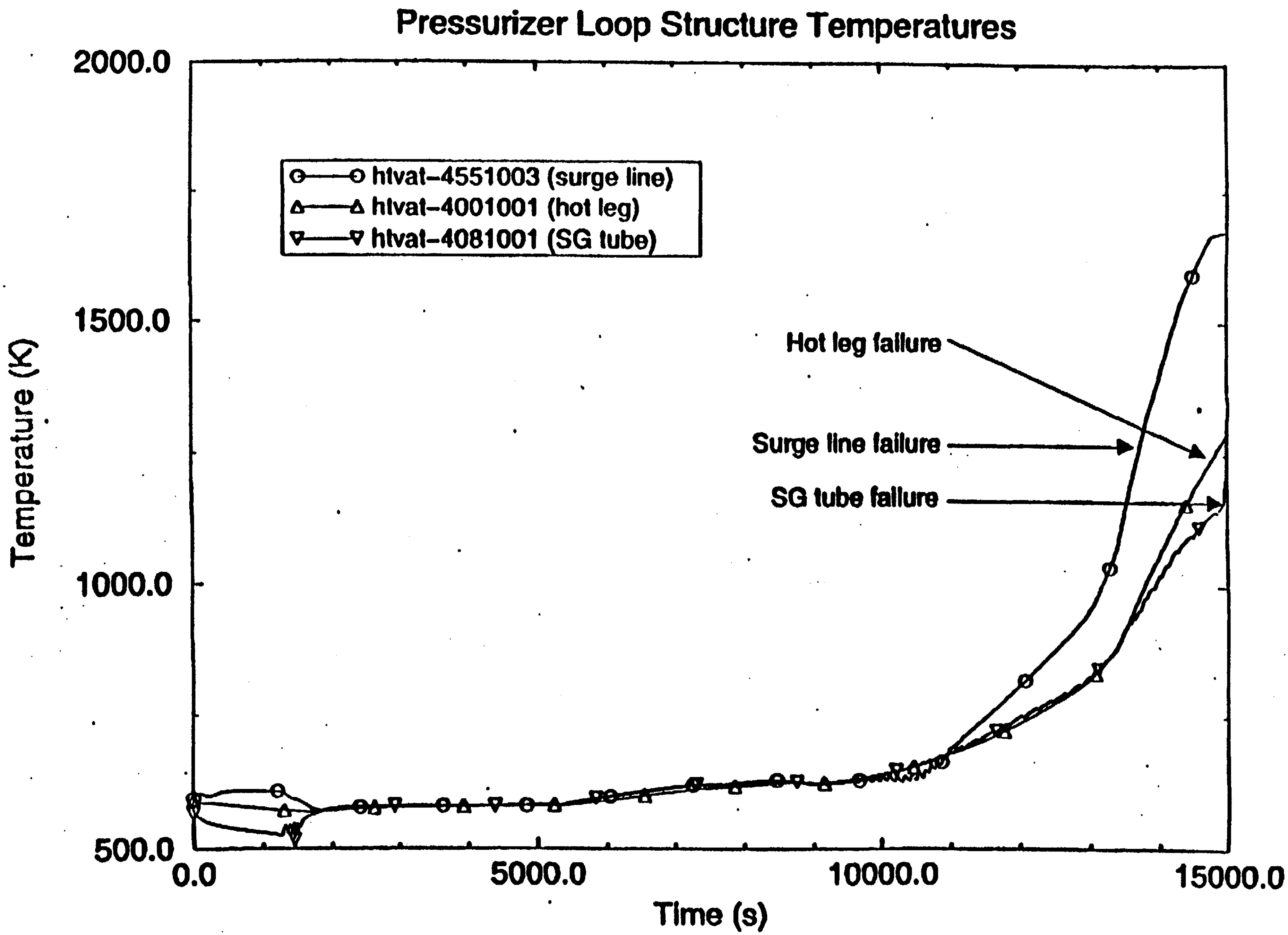
Item 5.4



Nodalization detail showing connections between the split hot leg and the split surge line.

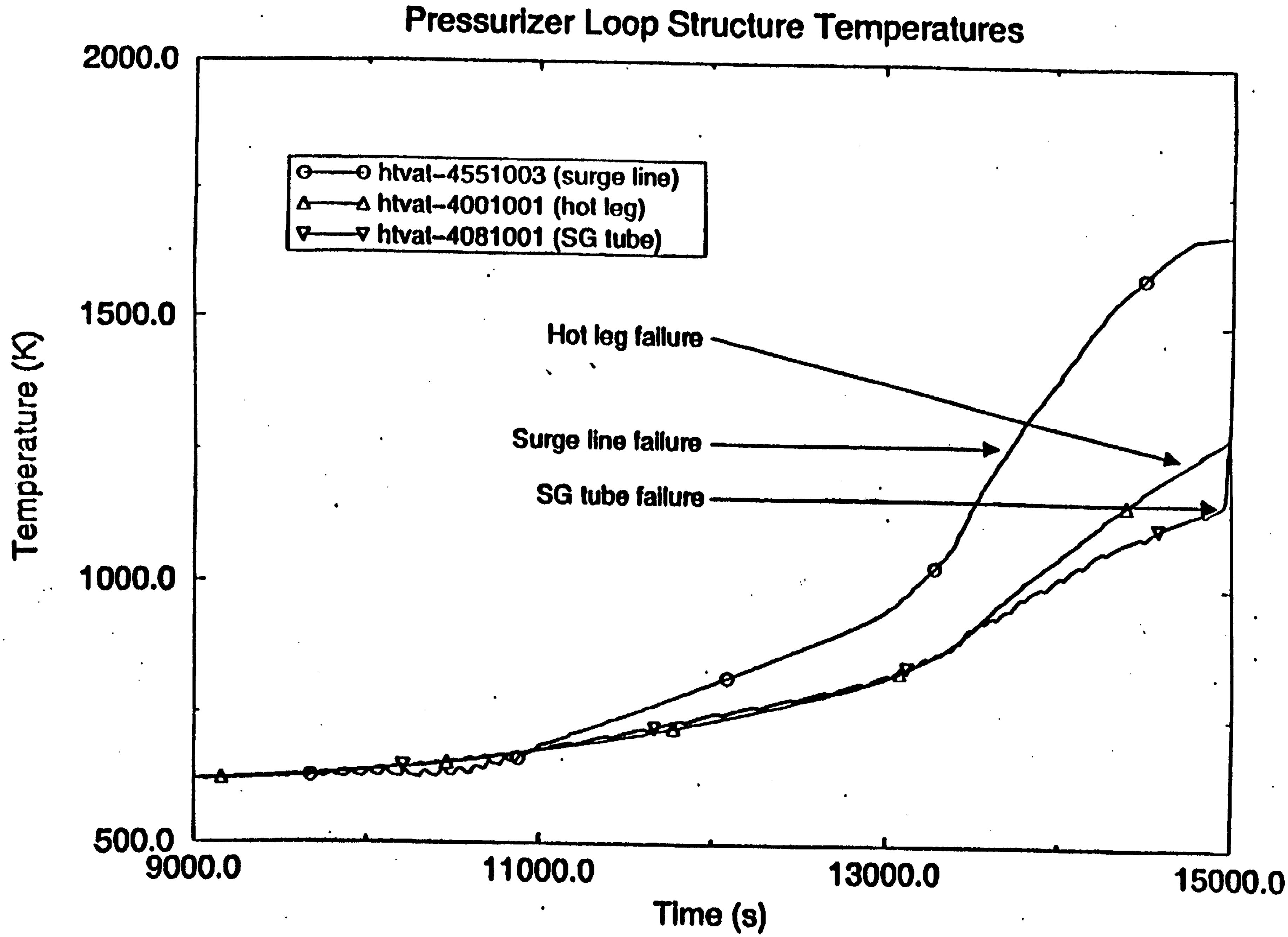
Steve Arndt

Item 5.4



Steve Arndt

Item 5.4



Steve Arndt

Item 5.4

Example Calculation (continued)

Application of SR5 (continued)

- Surry Plant calculations

  Conclusions:

  – pressurizer surge line creep rupture was the first RCS pressure boundary failure

    - Surge line rupture occurred at 13,730 sec
    - SG tube temp at this time 957 °K

  – pressurizer surge line failed in early phase of core damage (before the onset of fuel melting

  – if pressurizer surge line and hot leg failures are ignored AND a SG secondary ADV fails open, SG tube rupture occurs ~ 20 minutes after the first RCS pressure boundary failure.

Conclusions

- SG tube T/H boundary conditions are most directly influenced by variations in the accident sequence which determine pressurization/depressurization of the primary and secondary system
  - Failure of primary and secondary relief valves
  - Primary side PORV operation (Accident Management)
  - Pump seal leakage
- SG tube T/H boundary conditions are also influenced by phenomenological issues associated with counter current natural circulation and RCS T/H.
  - SG inlet plenum mixing
  - heat transfer modeling
  - loop seal clearing
- Variations in the treatment of these issues within reasonable ranges did not significantly worsen SG tube boundary conditions.

Code Validation

- SCDAP/RELAP5 code was used to assess fluid heating of SG tubes.
  - Benchmarked against W 1/7-scale test data
  - Considerable experience developed in similar applications (DCH, unintentional depressurization) over nearly 10 years
  - Peer reviewed for this specific application (counter current natural circulation and SG tube analysis.

Code Validation (continued)

- Integral experiments were conducted by W at 1/7 scale under an EPRI/NRC cooperative program to investigate severe accident natural circulation in PWRs with U-tube SGs.
  - Several series of tests conducted, using water, low pressure SF_6.
- Low pressure tests showed (by using dye in the fluid), that a stable countercurrent flow was present in the hot legs. Flow patterns were consistent over a wide range of conditions.
- High pressure SF_6 tests provided data for validation of codes. Five series of experiments with high pressure SF_6 were conducted. Temperature measurements in the SG inlet plenum and tube inlets indicated that the fluid in the inlet plenum was well mixed.

SCDAP/RELAP5 Analysis

- Additional SR5 calculations were performed to address recommendations made by ACRS and/or peer-reviewers:

  - Variations in heat transfer coefficients in the surge line, hot leg, reactor vessel upper plenum, and SGs.

  - Account for radiation heat exchange between the hot and cold streams of steams in the hot leg, and the circumferential wall heat conduction in the hot leg.

  - Synergistic effects (by changing two or more key parameters, for example the number of SG tubes carrying hot flow, mixing fraction, and recirculation flow simultaneously).

SCDAP/RELAP5 Analysis

Case	First tube failure minus first failure time(s)	Pressurizer-loop SG tube structure temperature @ surge line failure time(K)
6B ("h" x 0.8)	1180 (~ 20 mins)	964
6D ("h" x 1.3 in SG tube entrance)	1220 (~ 20 mins)	957
6 (nominal "h")	1230 (~ 21 mins)	957
6C ("h" x 1.3) in all entrances	1250 (~ 21 mins)	944
6Z ("h" x 1.2)	1310 (~ 22 mins)	938
6E fluid to fluid h.t. and circumferen tial wall conduction in hot leg)	1510 (~ 25 mins)	937

SCDAP/RELAP5 Analysis (continued)

- To explore synergistic effects (change the number of SG tubes carrying hot flow, mixing fraction, and recirculation flow simultaneously assuming that there is no relationship among these factors), use 5% confidence level from W 1/7 scale high-pressure transient tests.

		Case 6	Case 6F
hot/cold SG tube split	→	53%/47%	43%/57%
mixing fraction	→	0.87	0.73
recirculation ratio	→	1.9	1.78

Case	First tube failure minus first failure time(s)	Pressurizer-loop SG tube structure temperature @ surge line failure time(K)
6 (53%/47% hot/cold SG tube split)	1230 (~ 21 mins)	957
6F (5% confidence level for W high-pressure transient tests)	800s (~ 13 mins)	1007

Effect of Leakage on SG Inlet Plenum Mixing

Concern: SG tube leakage during severe accident could alter mixing in inlet plenum.

- 1/7 Scale tests did not simulate tube leakage.

Staff's evaluation of issue.

- Tube leakage effect on inlet plenum mixing likely to be dispersed (allowable leakage is an aggregate)
- Leak area equivalent to 1 gpm would produce, during severe accident, flow rate which is a very small fraction (<<1%) of tube bundle flow.
- CFD code analysis should be able to provide additional insights.

Fission Product Deposition

- Objective of the analysis

  To determine the effect of fission product transport and deposition in the RCS on SG tube integrity.

- Summary of approach

  Used the VICTORIA fission product code with SCDAP/RELAP5 calculated T/H conditions as input.

- Conclusion

  Fission product transport and deposition in the RCS have a negligible effect on SG tube integrity, because the fission product release is relatively small and late in the transient.

  - Volatile fission product release represents 5–10% of decay heat.

  - Fission products spread among upper plenum, hot leg, SG plena and tubes.

Conclusions

- Tube heating during severe accidents has been analyzed using benchmarked models (validated against scaled experimental data), undergone peer review, and sensitivities examined through parametric variations.
- Evaluation of tube performance during severe accidents would benefit from resolution of uncertainty regarding T/H conditions.
- Confirmation of temperature variation/uncertainty underway.
  - More rigorous consideration of uncertainties in SG inlet plenum mixing.
  - Additional sequence/plant variation.
  - More detailed CFD modeling.

Uncertainties In Risk Assessments

There are numerous sources of uncertainty in the calculation of risk due to steam generator tube degradation.

Experience has pointed to a few areas that are major contributors to uncertainty:

human error probabilities

(leading to about 10^4 variation in results)

NDE detection of flaws (probability of detection)

(POD claimed 99%+ *vs* <40% demonstrated)

Tube strength estimates based on NDE characterization of flaws

(95% confidence strength ~1300 psi below best estimate value at 3ΔP pressures)

Difficulties In Risk Assessments

Materials and structural issues:

Creep models for RCS components assume infinitely long, thin-wall, straight tubes, although bends and welds are expected to dominate failure behavior.

Differential thermal growth of different temperature tubes is not considered as a possible mechanism for displacing degraded tube sections from confining structural supports.

Creep of tubes at high temperatures may result in tubes contacting each other before failure.

High-temperature of components capable of making small RCS leakage paths has not been studied sufficiently to predict efects on RCS pressure.

Difficulties In Risk Assessments
(Continued)

Thermal-hydraulic modeling:

RELAP/SCDAP models one *average* hot tube, but we need to know *hottest* tube temperature to assure survival of all tubes

We need *distribution* of tube temperatures to predict failure probability of significant flaws and high temperatures coinciding in same tube.

Mixing of counter-current flows in SG inlet plenum is not well enough understood to take into account:

different plenum geometry of different reactor vendor types

effects of leakage in tubes

Sequences with small leakages (elsewhere) in RCS appear to be more challenging than no leakage or large leakage, but effects of variations in leak rate and initiation timing are not explored for small leakage scenarios.

Consistent differences in results between MAAP and SCDAP/RELAP not fully understood.

Effects of radiative heat transfer between fluids and walls.

New Research Efforts Related to Steam Generator Tube Integrity

- NRR User Need Memorandum dated 2/8/00
- RES Response Memorandum dated 9/7/00
- Four primary areas of research
    - Probabilistic Risk Assessment
    - Thermal-Hydraulic Accident Conditions and Dose Consequences
    - Structural Behavior of Steam Generator Tubes
    - Structural Behavior of Primary System Components other than Steam Generator Tubes

Probabilistic Risk Assessment

- Develop PRA for selected plants to identify accident sequences which could lead to steam generator tube failure and potential containment bypass.

  - Review of NUREG-1150, NUREG-1570 and other PRAs to identify initiators, accident sequence, and key design characteristics.

  - Review of operation data for initiators or conditions which are not seen in previous PRAs but could be significant.

  - Develop PRA models and initial accident sequence frequency estimates.

  - Refine models and frequence estimates based on detailed T/H and engineering analysis.

  - Perform sensitivity analyses on key design characteristics to expand methods and results beyond those of the specific plants studied.

- Develop an integrated framework to assess how the individual parts of the research effort in Steam Generator Tube Integrity will contribute to the associated uncertainties in the overall analysis of these events.

Thermal Hydraulics and Dose Consequences

- Accident Sequence Variations
  Analysis of potentially important variations in the baseline station blackout sequence

- Plant Design Differences
  Review of plant differences and areas of possible concern in the review of plant applications for several different plant designs including, Surry, Zion and ANO.

- Inlet Plenum Mixing
  New, more formal uncertainty analysis using a Monte-Carlo sampling method of parameter distributions.

- Tube to Tube Variations
  Detailed review of inlet plenum and steam generator tubes using CFD, as well as a new review of 1/7 scale data.

- Core Melt Progression
  Review of uncertainly associated with core oxidation and possible core blockage.

- Dose Consequences
  Review new research in the areas of transport and deposition of fission products.

Inlet Plenum Mixing

- The SCDAP/RELAP5 code will continue to be used as the principal tool for analysis of the tube T/H boundary conditions.
- Phenomenological uncertainty in the natural circulation calculation has centered on the issue of mixing in the SG inlet plenum.
- Additional uncertainty relates to heat transfer assumptions.
- Earlier work considered single and multiple simultaneous variations in inlet plenum mixing characteristics as well as variations in heat transfer coefficients.

Inlet Plenum Mixing (continued)

- New analysis performed under this plan will involve a more rigorous treatment of uncertainties.

- Distributions will be developed for the individual mixing parameters and heat transfer coefficients and sampled using Monte-Carlo techniques. SCDAP/RELAP5 analysis will then be performed for sampled points to develop a probabilistically weighted picture of SG tube temperatures.

- The current plan is to use the following parameters for the analysis: mixing fraction, recirculation ratio, number of tubes carrying flow forward, SG tube heat transfer coefficients, hot leg and surge line heat transfer coefficients.

Effects of Leakage and Tube to Tube Variations

- SCDAP/RELAP5 analysis limited in ability to resolve variation in T/H effects among tubes.
- To estimate tube-to-tube variations, re-examine the experimental basis for the modeling (i.e., the 1/7th scale test data) to determine the appropriate variability for plant conditions.
- Use computational fluid dynamics (CFD) codes to predict inlet plenum mixing and tube to tube variations, including the effects of leakage.
- CFD code will need benchmarking against experimental data, but fundamentally, CFD codes have greater inherent capabilities for solving this type of fluid flow problem.

CFD Work as Part of Inlet Plenum Mixing and Tube to Tube Variation Tasks

- As part of the severe accident tube integrity investigation, RES has initiated a CFD study of steam generator inlet plenum mixing.
- The Westinghouse 1/7th scale test data will be used to assess the CFD technique
  - The assessment will be challenging due to the limited nature of this data
- Next CFD will be applied to a full scale steam generator geometry and sensitivity studies will be completed.
- 2D model will be studied for grid dependence and modeling option sensitivies
- 3D model is being developed
- Mixing parameters will be computed from the CFD predictions. Sensitivity studies will be completed.
- Assuming success with W 1/7th evaluation, a full scale plant will be modeled.

Current Status of Work

- Probabilistic Risk Assessment
  - This work is just getting underway
- Thermal-Hydraulic Accident Conditions and Dose Consequences
  - SCDAP/RELAP5 work will be done at INEEL. The contract is in place and the work was stared in July 2000.
  - The work to date has included establishing a new baseline calculation using the newest version of SCDAP/RELAP5, MOD 3.3, and beginning the development of parameter distributions for the inlet plenum mixing task.
  - Work on this part of the user need is now planned to be complete in March 2002, with intermediate products available as work on individual tasks are completed.
- Structural Behavior of Steam Generator Tubes
  - Work complete, reports in perperation
- Structural Behavior of Primary System Components other than Steam Generator Tubes
  - This work is being planed