UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

July 14, 2011

MEMORANDUM TO: ACRS Members

FROM: Christopher L. Brown, Senior Staff Engineer /RA/
Technical Support Branch, ACRS

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS **OPEN** MATERIAL, METALLURGY, AND REACTOR FUELS SUBCOMMITTEE ON CONSEQUENTIAL STEAM GENERATOR TUBE RUPTURE (C-SGTR) APRIL 6, 2011 IN ROCKVILLE, MARYLAND

The minutes of the subject meeting were certified on July 14, 2011, as the official record of the proceedings of that meeting. A copy of the certified Minutes is attached.

Attachment: As stated

cc w/o Attachment: E. Hackett
C. Santos

UNITED STATES
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WASHINGTON, DC 20555 - 0001

July 14, 2011

MEMORANDUM TO: Christopher Brown, Senior Staff Engineer
ACRS

FROM: Joy Rempe, Chair
Materials, Metallurgy, and Reactor Fuels Subcommittee

SUBJECT: CERTIFICATION OF MINUTES OF THE ACRS OPEN MATERIALS, METALLURGY, AND REACTOR FUELS SUBCOMMITTEE ON CONSEQUENTIAL STEAM GENERATOR TUBE RUPTURE (C-SGTR) APRIL 6, 2011 IN ROCKVILLE, MARYLAND

I hereby certify, to the best of my knowledge and belief, that the Minutes of the subject meeting on April 6, 2011, are an accurate record of the proceedings for that meeting.

RA 7/ 14 /11

Joy Rempe, Date
C-SGTR Subcommittee Chair

Certified by: J. Rempe
Certified: July 14, 2011

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF ACRS SUBCOMMITTEE MEETING ON CONSEQUENTIAL STEAM GENERATOR TUBE RUPTURE
APRIL 6, 2011
ROCKVILLE, MARYLAND

INTRODUCTION

The Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on Materials, Metallurgy, and Reactor Fuels met in room T-2B1 at the Headquarters of the U.S. Nuclear Regulatory Commission (NRC), located at 11545 Rockville Pike, Rockville, Maryland, on April 6, 2011. The Subcommittee was briefed by representatives of NRC's Office of Nuclear Regulatory Research (RES) and Nuclear Reactor Regulations (NRR) on Consequential Steam Generator Tube Rupture (C-SGTR).

The Subcommittee planned to gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the Full Committee of the ACRS at a later date.

The Chair for this ACRS Subcommittee meeting was Dr. Joy Rempe. Mr. Christopher Brown was the ACRS staff cognizant engineer for this topic and served as the Designated Federal Official for this meeting. This meeting was open to public attendance and no proprietary information was discussed. The Subcommittee received no written comments or requests for time to make oral statements from any members of the public concerning the subject of this meeting. The meeting convened at approximately 1:30 pm and adjourned at 4:41pm.

The detailed agenda identifying the specific presentation topics comprising this meeting can be found in Attachment 1. Both during and following the scheduled presentations, the speakers responded to specific questions and comments from the ACRS Subcommittee members. The scope of the questions, comments, and answers thereto, and the speaker's responses thereto, have been captured in the verbatim meeting transcript. As a result of Member questions and comments, and speaker responses (answers), several follow-up items were identified for further discussion at subsequent Subcommittee meetings. These follow-up actions will be tracked by the ACRS staff.

ACRS Subcommittee meeting transcript for this meeting can be found at the following NRC Internet website location*: http://www.nrc.gov/ reading-rm/doc-collections/acrs/tr/subcommittee/*.

ATTENDEES:

The following list of individuals (and their affiliations) attending this meeting was compiled using both the sign-in sheets and the Subcommittee meeting transcript.

ACRS Members

J. Rempe, Subcommittee Chair	S. Abdel-Khalik
J. S. Armijo	J. Stetkar
J. Sieber	M. Corradini
D. Powers	H. Ray
D. Bley	W. Shack

ACRS Staff

C. L. Brown, Designated Federal Official	C. Santos
K. D. Weaver, ACRS staff	

NRC Staff

R. Iyengar, RES	A. Zoulis, NRR
C. Harris, RES	S. Wong, RES
A. Csontos, RES	R. Lee, RES
E. Fuller, NRO	E. Murphy, NRR
K. Coyne, RES	S. Sancaktar, RES
M. Gavrilas, RES	

SCHEDULED PRESENTATIONS:

The scheduled briefing provided an update on additional work as part of the closure of NRC's Steam Generator Action Plan (SGAP). The phenomenological aspects of C-SGTR and the technical approach were also presented. Specifically, guidance and tools developed to support future risk assessments of C-SGTR. The tools (i.e., software package) and guidance will assist NRR staff in its evaluation of proposed modifications to existing requirements (e.g., alternate tube repair criteria) and in evaluating the risk significance of SG tube degradation in the reactor oversight process.

The scheduled presentations focused on RES project plan to address NRR's user need. The agenda is attached to these Minutes.

OPENING REMARKS AND OBJECTIVES:

Dr. Joy Rempe, Chair of the ACRS Subcommittee on C-SGTR, convened the meeting at 1:30 p.m. Dr. Rempe indicated in her opening remarks that the staff and industry have expended considerable resources in the last ten years to better understand the safety implications and risks associated with C-SGTR events. She briefly summarized some of the key research activities that have been done in this area thus far.

Dr. Raj Iyengar, lead C-SGTR project manager made an opening statement. He provided an overview of the C-SGTR project plan. He indicated that the staff was seeking insight and feedback from the Subcommittee to ensure that they were on the correct path regarding the project.

INDIVIDUAL PRESENTATIONS:

Origin of User Need, User Need Details & Regulatory Implications

Antonios Zoulis from NRR, Division of Risk Assessment provided background and discussed the tasks associated with the user need that was developed by NRR. He indicated that there was a need to continue to follow-up work on C-SGTR, in particular, the need for further thermal hydraulics analyses to address the Combustion Engineering (CE) plant issues. Furthermore, he said therewas a need to update SG flaw distributions and to develop an enhanced Reactor Coolant System (RCS) structural analysis. He indicated that all of the research and analysis for this endeavor would be documented for knowledge management purposes.

Mr. Zoulis discussed the work being done by NRR and RES relating to thermal hydraulics. The thermal hydraulic analysis would focus on updating the CFD codes and address the issues associated with CE designed plants. Further, he said that the staff wants to investigate the impact of in-core instrument tube failure on natural circulation for both Westinghouse and CE plants. A plan is underway to update the SG flaw distributions for the current population of SGs and a structural analysis for both Westinghouse and CE RCS components to establish confidence in the prediction of RCS piping failure.

In summary, Mr. Zoulis said that the staff would like to better understand C-SGTR phenomena and its implication to risk assessments and develop efficient tools to be used by risk analysts.

RES Project Plan to Address NRR User Need

Dr. Iyengar discussed the user need issued by NRR and how staff internally communicates regarding this effort. He presented and explained a project execution matrix. He concluded his presentation with a list of research products that will be provided to NRR from this effort. This list includes: 1) a simplified method to assess risk associated with consequential tube rupture and a summary report, 2) draft regulatory guidance on risk-informed decision making with respect to C-SGTR, 3) a draft Risk Assessment of Operational Events (RASP) handbook section on assessment of C-SGTR suitable to support revisions to the inspection manual chapter, and 4) a summary report compiling key research results.

Phenomenological Aspects of C-SGTR

This topic was presented by Dr. Richard Lee. A steam generator tube rupture (SGTR) is a design basis event, but the events considered in this study are more severe, such as an induced SGTR. The severe accident conditions, created by the overheated core, are transported to the RCS loops through natural circulation. Severe accidents are characterized by core damage, high temperatures, and radionuclide releases. Dr. Lee explained a diagram showing severe accident natural circulation flow progression. He also discussed a diagram showing system code modeling considerations required for this effort. Some of the factors consider in the modeling include: the impact of decay heat, heat loss in piping, and the impact of the assumed heat transfer coefficient. An example CFD calculation was presented showing surge line flow and the predicted mixing. He also presented a slide showing the impact of inlet plenum mixing for two SG designs. The analysis performed by CFD indicates the conditions that the steam generator tube could encounter. Dr. Lee also explained a sample map of containment bypass potential that considered primary and secondary side leakage rates. He indicated that Dr. Christopher Boyd, the lead for most of this work, was unavailable to attend this meeting.

Technical Approach

Dr. Lee discussed the thermal hydraulic analysis performed using CFD and MELCOR. Key objectives of the proposed thermal hydraulic analysis are to: 1) update existing CFD and system code models for CE plants, 2) provide un-failed thermal hydraulic behavior for selected accidents, 3) provide failed thermal hydraulic and volatile (Cs, I, Te) releases based upon provided failures, and 4) provide assessment impact of instrument tube failures for Westinghouse and CE plants. Dr. Lee indicated that the staff is now developing the CE Calvert Cliffs plant model that will include a CFD model of the CE hot leg and SG lower plenum. There was discussion on the list of thermal hydraulic uncertainties; in particular, the uncertainty relating to the turbine-driven auxiliary feedwater availability was noted. Member Abdel-Khalik asked several questions related to auxiliary feedwater, which are documented in the attached table.

Mr. Charles Harris provided information regarding the condition of tubes in the current SG fleet. In order to accurately represent the current fleet, he noted that data would be required for CE, Westinghouse and B&W steam generators. To accurately describe the flaws, Mr. Harris said the staff needs to know the number, size, type, and location to get a total leak area to complete probability calculations. Most of the data generated on flaw distribution in the early nineties was on alloy 600 material, but this material has been replaced by alloy 600TT (thermally treated) and alloy 690. Mr. Harris indicated that the staff plans to update NUREG/CR 6521 flaw distributions. He said that the original statistics are still valid, but the staff plans to adjust the study for newer materials and incorporate newer in-service inspection data such as number, size, type, and location of flaws.

Dr. Iyengar discussed RCS component failure predictions. He said that the main task is to identify, characterize, and model relevant RCS nozzles, as well as other potential weak areas in order to determine when failure would occur. This information is input into a software package (calculator) to make an assessment of the potential for containment bypass. He discussed some of the challenges associated with this task. He also discussed the approach for predicting failure of RCS components. ABAQUS is a general purpose finite element analysis software that will be used to predict failure time.

Dr. Selim Sancaktar, the lead PRA engineer, discussed the PRA related activities of the NRR user need. The PRA task involves the creation of a user-friendly methodology for assessing the risk associated with consequential tube rupture/leakage in DBA and severe accident events. The methodology will be used to support risk-informing the regulatory process. Dr. Sancaktar said that a relatively small effort is being done to create a consequential steam generator calculator for the specific task of estimating steam generator tube leakage probabilities under different conditions and for different designs. The Subcommittee members asked a number of questions about the calculator. Dr. Sancaktar provided additional details related to the operation of this calculator. A PRA report and the C-SGTR calculator are expected to be delivered within the next two years.

Dr. Rempe conducted a brief round table discussion with Subcommittee Members at the end of the briefing to solicit comments. The staff (Kevin Coyne) indicated that they would like to informally meet with the Subcommittee Chair in September to provide an update on this work.

SUBCOMMITTEE QUESTIONS/COMMENTS

See attached

BACKGROUND MATERIALS PROVIDED TO THE SUBCOMMITTEE PRIOR TO THIS MEETING:

1. 1. U.S. Nuclear Regulatory Commission, NUREG-1740, “Voltage-Based Alternative Repair Criteria,” March 2001 (ML010750315)
2. A Risk Assessment of Consequential Steam Generator Tube Ruptures Final Report, RES 03/20/2009 (ML083540412)
3. Letter Report, JCN Y6486, “Severe Accident Initiated Steam Generator Tube Ruptures Leading to Containment Bypass – Integrated Risk Assessment,” prepared for the Office of Nuclear Regulatory Research by Sandia National Laboratories and Science Applications International Corp., 02/2008 (ML080500084)

Below:
1. Meeting Agenda
2. Slides
3. Questions and Comments

ACRS Meeting of the Subcommittee on Materials, Metallurgy, & Reactor Fuels

Consequential Steam Generator Tube Rupture (C-SGTR) Subcommittee Briefing

Rockville, MD

Wednesday, April 6, 2011

Cognizant Staff Engineer: Christopher L. Brown (301)-415-7111, Christopher.Brown@nrc.gov

Item	Topic	P resenter(s)	Time
1	Opening Remarks and Objectives	**Dr. Joy Rempe, ACRS**	1:30 - 1:35 p.m.
2	Staff Opening Remarks	Dr. Raj Iyengar, RES	1:35 - 1:40 p.m.
3	Origin of the User Need User Need Details and Regulatory Implications	Mr. Antonios Zoulis, NRR	1:40 - 2:00 PM
4	RES Project Plan to Address NRR User Need	Dr. Raj Iyengar, RES	2:00 - 2:15 PM
5	Phenomenological Aspects of the C-SGTR	Dr. Michael Salay, RES	2:15 - 2:45 PM
	Break		**2:45 PM – 3:00 PM**
6	Technical Approach	Michael Salay, Selim. Sancaktar, Charles Harris, and Raj Iyengar, RES	3:00 - 4:15 PM
7	Committee Discussion	**Dr. Joy Rempe, ACRS**	4:15 - 4:30 PM
8	Adjourn	**Dr. Joy Rempe, ACRS**	4:30 PM

ACRS Notes:

- During the meeting, 301-415-7360 should be used to contact anyone in the ACRS Office.
- Presentation time should not exceed 50 percent of the total time allocated for a given item. The remaining 50 percent of the time is reserved for discussion.
- Thirty five (35) hard copies (2 B&W slides per page) of each presentation or handout should be provided to the Designated Federal Official 30 minutes before the meeting.
- 10 full page colored copies for the ACRS members and the court reporter.
- One (1) electronic copy of each presentation should be emailed to the Designated Federal Official 1 day before the meeting. If an electronic copy cannot be provided within this timeframe, presenters should provide the Designated Federal Official with a CD containing each presentation at least 30 minutes before the meeting.

Consequential Steam Generator Tube Rupture (C-SGTR)

Subcommittee Briefing

Advisory Committee On Reactor Safeguards

April 6, 2011

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Purpose

- Provide project status update on C-SGTR activities
- Outline the project plan that had been developed and discussed with NRR technical staff
- Early engagement with ACRS to gain insight and obtain feedback

Origin of User Need, User Need Details & Regulatory Implications

Antonios Zoulis, NRR

Outline

- Background
- User Need
- Summary

Background

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- As part of the closure of the NRC's Steam Generator Action Plan in 2009, items were identified that needed further work:
  - Further T-H analyses to address CE plants issues
  - Development of updated SG Flaw distributions and enhanced RCS structural analyses
  - Development of guidance and tools to support future risk assessments
  - Document summarizing key research and state-of-knowledge

Background (Cont.)

U.S.NRC
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- Staff decided to pursue further research items in a follow-on NRR user need to RES (ML092010380)
- This approach to closing out the SGAP was presented to, and endorsed by, the Advisory Committee on Reactor Safeguards in October 2009

User Need – NRR-2010-005

- Thermal-Hydraulic Analyses
  - Request updated CFD and system code models for CE plants
  - Report on impact of incore instrument tube failure on natural circulation for both Westinghouse and CE plants

User Need

- Materials and Structural Analyses
  - Update SG flaw distributions for current population of SGs
  - Structural analysis of both Westinghouse and CE RCS components to establish confidence in the prediction of RCS piping failure

User Need

U.S.NRC
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- Risk Assessment
  - Develop an efficient method for assessing the risk associated with C-SGTR/leakage in DBA and severe accident events
  - Reassess conditional SG tube failure probabilities based on updated flaw distributions and T-H analyses
  - Develop draft Regulatory Guidance on risk-informed decision making regarding C-SGTR
  - Develop Risk Assessment Standardization Project (RASP) Handbook guidance and update Inspection Manual Chapter (IMC) 0609 appendices to support risk assessments (SDP) for the Reactor Oversight Program

- Prepare a summary report compiling key insights and state-of-knowledge

Summary

- Develop and understand the C-SGTR phenomena and its implication to risk assessments

- Develop efficient tools to be used by SRAs and risk analysts to evaluate findings, risk-informed applications, and future issues involving SGs

- Document and develop guidance to capture knowledge

RES Project Plan to Address User Need

Raj Mohan Iyengar, RES

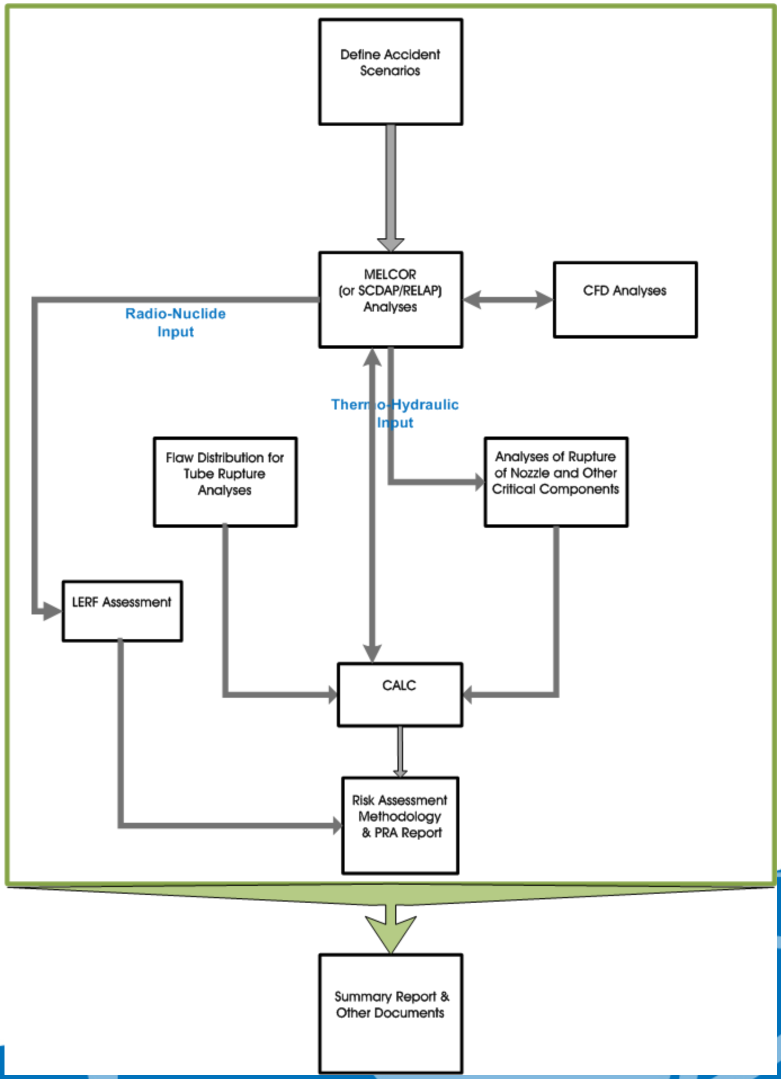
User Need Tasks

Item	Description	Priority	NRR Lead	RES Lead
1.1.A.i	Update existing computational fluid dynamics (CFD) and system code models (either the MELCOR or RELAP/SCDAP code) for a representative CE plant	High	Antonios Zoulis, DRA/APLA	Michael Salay DSA/FSTB
1.1.A.ii	Evaluate the expected T-H behavior and accident progression for selected risk-significant accidents from the associated PRA	High	Antonios Zoulis DRA/APLA	Michael Salay DSA/FSTB
1.1.B.i	A technical assessment of the impact of incore instrument tube failures on natural circulation for Westinghouse plants	Medium	Antonios Zoulis, DRA/APLA	Michael Salay DSA/FSTB
1.1.B.ii	A technical assessment of the impact of incore instrument tube failures on natural circulation for CE plants	Medium	Antonios Zoulis, DRA/APLA	Michael Salay DSA/FSTB
1.2.A	Updated SG flaw distributions representative of the current population of SGs	High	Emmett Murphy, DCI/CSGB	Charlie Harris DE/CMB
1.2.B.i	Structural analysis of Westinghouse RCS components to establish confidence in the prediction of RCS piping failure	High	Emmett Murphy, DCI/CSGB	Raj Iyengar DE/CIB
1.2.B.ii	Structural analysis of CE RCS components to establish confidence in the prediction of RCS piping failure	High	Emmett Murphy, DCI/CSGB	Raj Iyengar DE/CIB
1.3.A.i	Develop a simplified method for assessing the risk associated with consequential tube rupture/leakage in DBA and severe accident events	High	Antonios Zoulis, DRA/APLA	Selim Sancaktar DRA/PRAB
1.3.A.ii	Modify risk assessment tool to account for elevated axial tube loads due to thermal expansion between the SG shell and tubes during steam line break, loss of coolant accidents, and loss of main feedwater events (work to be sequenced with existing User Need NRR-2008-004 - ML082200693)	High	Antonios Zoulis, DRA/APLA	Selim Sancaktar DRA/PRAB and Charlie Harris DE/CMB
1.3.B	Reassess conditional SG tube failure probabilities based on updated flaw distributions and updated T-H analyses	High	Antonios Zoulis, DRA/APLA	Selim Sancaktar DRA/PRAB
1.3.C.i	Develop draft Regulatory Guidance on Risk-Informed Decision Making Regarding C-SGTR	High	Antonios Zoulis, DRA/APLA	Selim Sancaktar DRA/PRAB
1.3.C.ii	Develop draft RASP Handbook section on assessment of C-SGTR suitable to support revisions to the Inspection Manual Chapter (IMC) 0609 appendices supporting the SDP process	High	Antonios Zoulis, DRA/APLA	Selim Sancaktar DRA/PRAB
1.4	Prepare summary report compiling key research results	High	Antonios Zoulis, DRA/APLA	Raj Iyengar DE/CIB

Simplified Project Flow Chart

Communication & Engagement

- RES Task Groups meet at least once a month
- Expect to provide status update and receive feedback from NRR on a quarterly basis
- Technical Engagement with ACRS



Research Products

- Simplified Method to Assess Risk Associated with Consequential Tube Rupture and a Summary Report
- Draft Regulatory Guidance on Risk-Informed Decision Making Regarding C-SGTR (Nature of this document will be determined later in the project)
- Draft RASP Handbook section on assessment of C-SGTR suitable to support revisions to the Inspection Manual Chapter (IMC) 0609 appendices supporting the Significance Determination Process (SDP)
- Summary report compiling key research results

Phenomenological Aspects of the C-SGTR

Richard Lee, RES

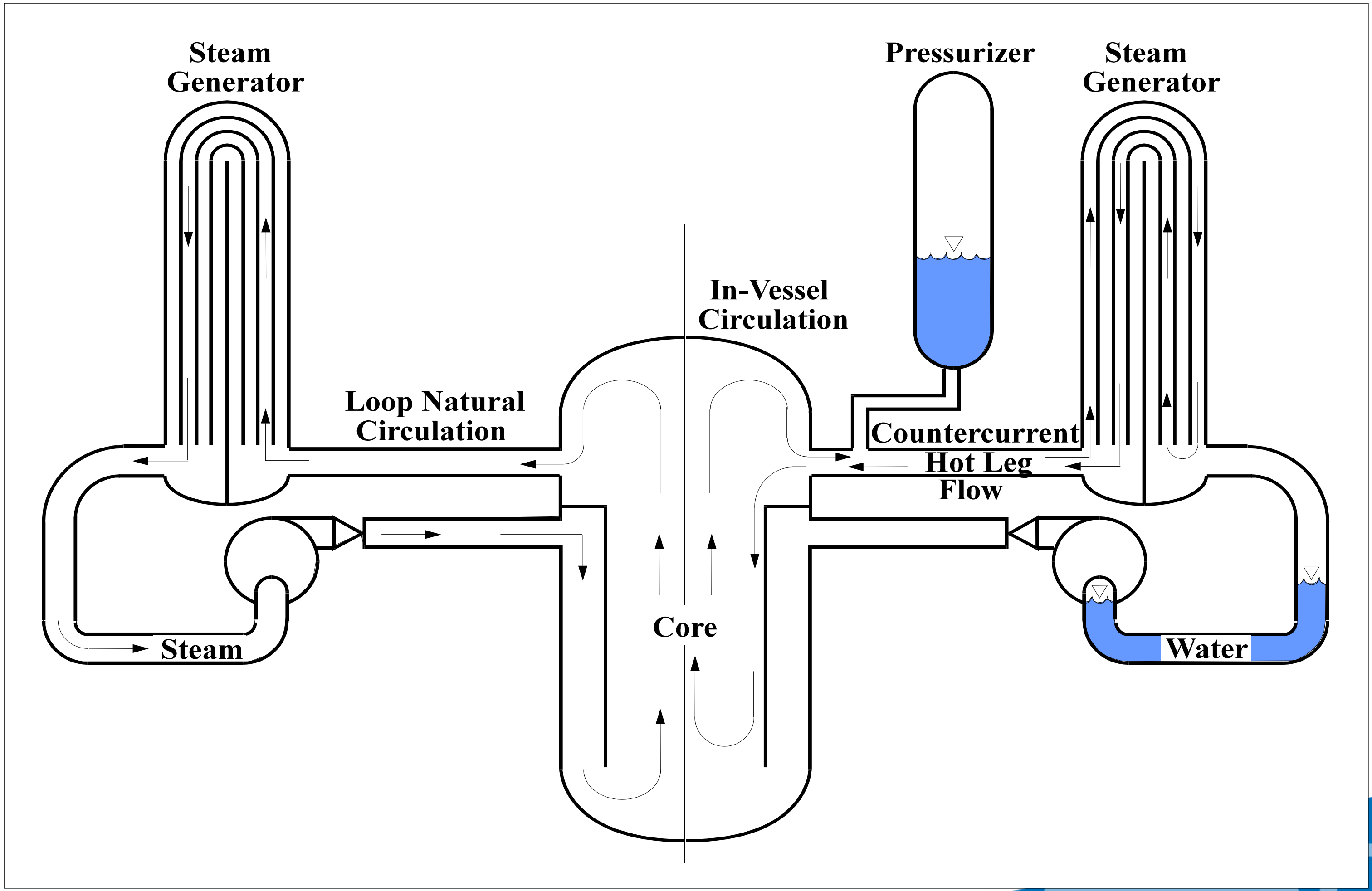
Steam Generator Tube Ruptures

- Steam generator tube ruptures are design basis events
  - Plants are designed to cope
  - Have for all events to date
- Progresses to Severe Accident only if something else happens
  - Failure to diagnose and respond can result in core melt
  - Multiple tube failure results in less time to react
- SGTRs (as initiating events) have been considered in risk analyses
  - Low probability to progress to SA but large consequences
    - Containment Bypass
    - Risk-dominant accident in PWRs at the time of NUREG-1150
- Recently risk analyses consider *consequential* SGTR

Severe Accident Induced Failure

- A primary system break induced by the high temperatures (and pressures) associated with severe accident conditions.
  - water level below the top of the fuel
  - superheated steam above core
- The severe accident conditions, created by the overheated core, are carried out into the RCS loops through natural circulation.
  - severe accidents are associated with core damage, high temperatures, and radionuclide releases
    - core temperatures over 2500 K
  - temperatures in the RCS challenge the structural integrity of the system
    - testing shows that a new steam generator tube will creep rupture at system pressure if exposed to temperatures above (approximately) 1170 -1200 K
- Significant induced failure points include the lower head, hot leg, pressurizer surge line, and SG tubing.

Severe Accident Natural Circulation Flows



High – Dry – Low

- The challenge to the tubes under counter-current flow conditions is maximized when the plant is in a “high-dry-low” condition
  - High primary side pressure
    - RCS must remain intact with no significant leaks
  - Dry steam generator secondary side
    - auxiliary feedwater systems fail
  - Low pressure on the secondary side
    - leakage or valve failure must occur to depressurize the secondary side

System Code Modeling Considerations

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-shell heat loss
-SG depressurization

-pressurizer draining
-surge line orientation

-natural circulation
-core bypass flow

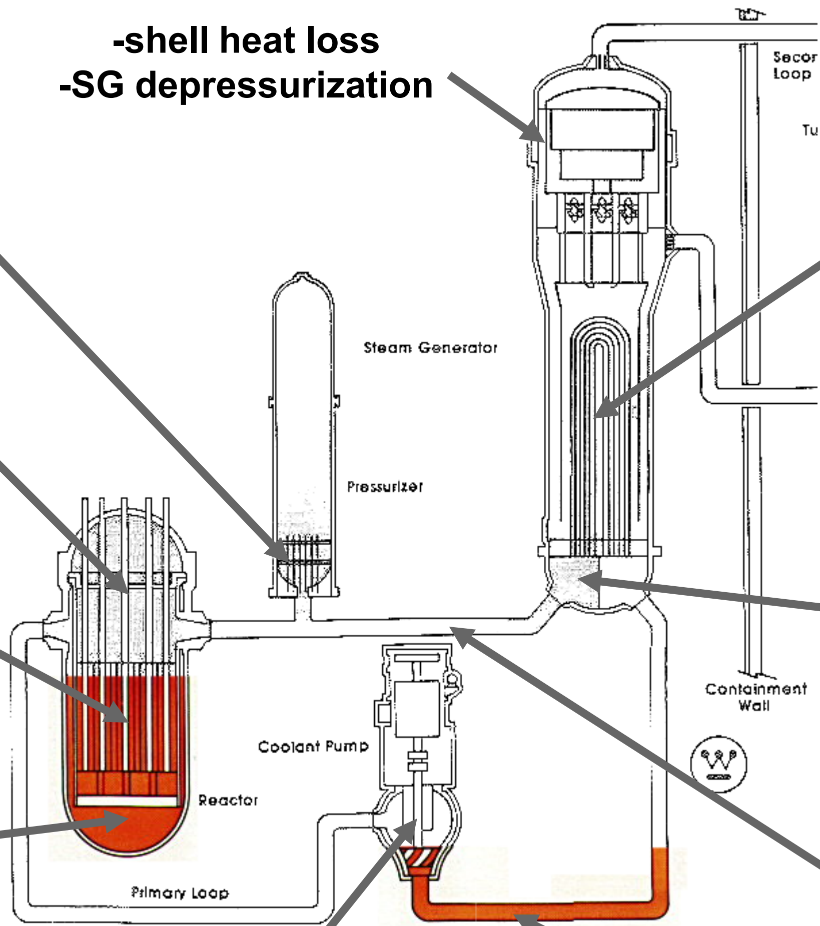
-oxidation rate
-core blockage
-nodalization
-natural circulation

-nodalization
-downcomer clearing

-tube heat transfer
-secondary flows
-mass flow
-hot tube fraction
-leakage
-plugging
-vertical node count

-inlet plenum mixing
-recirculation ratio
-plume T distribution

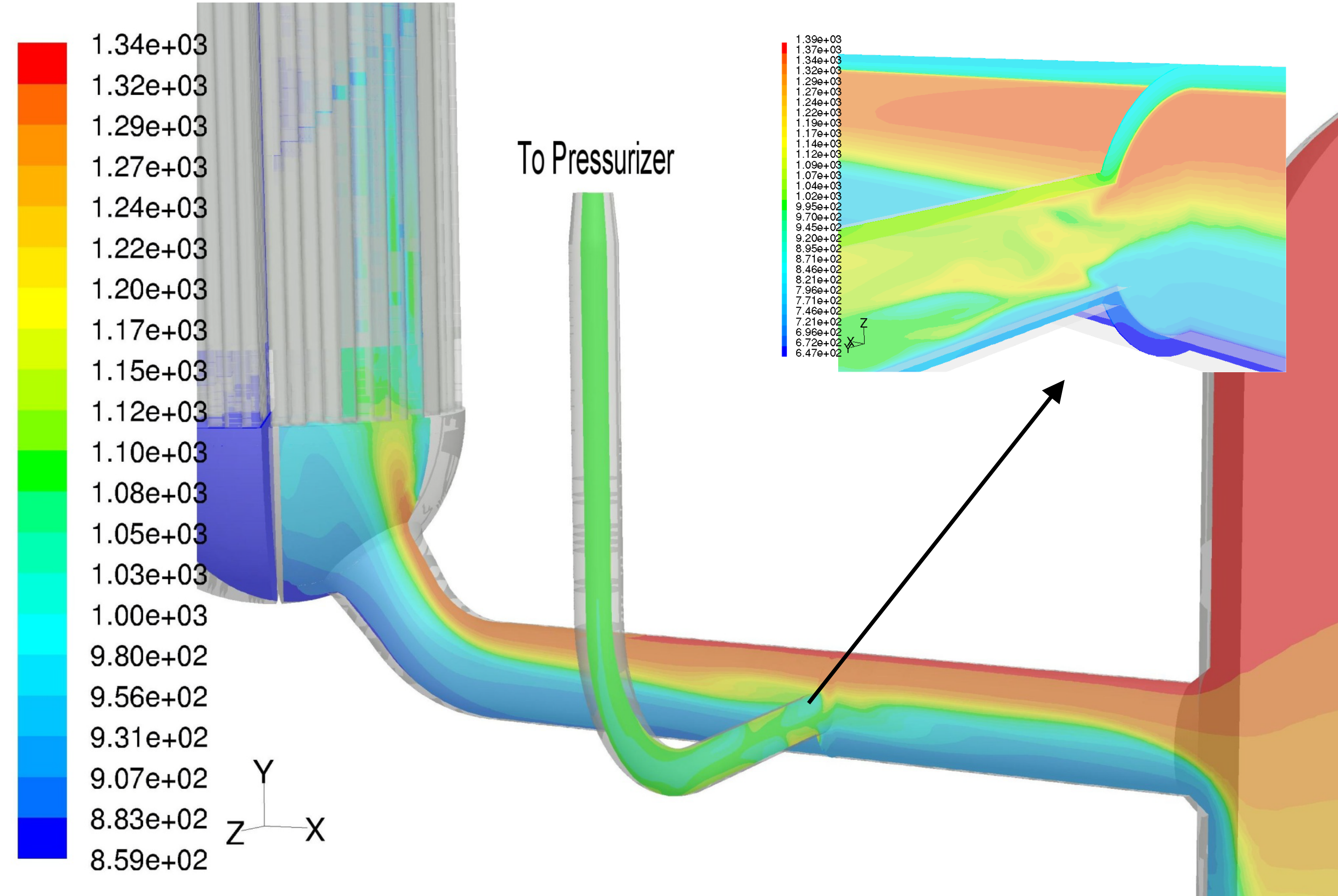
-HL flow rate
-entrainment
-radiation modeling
-entrance effects



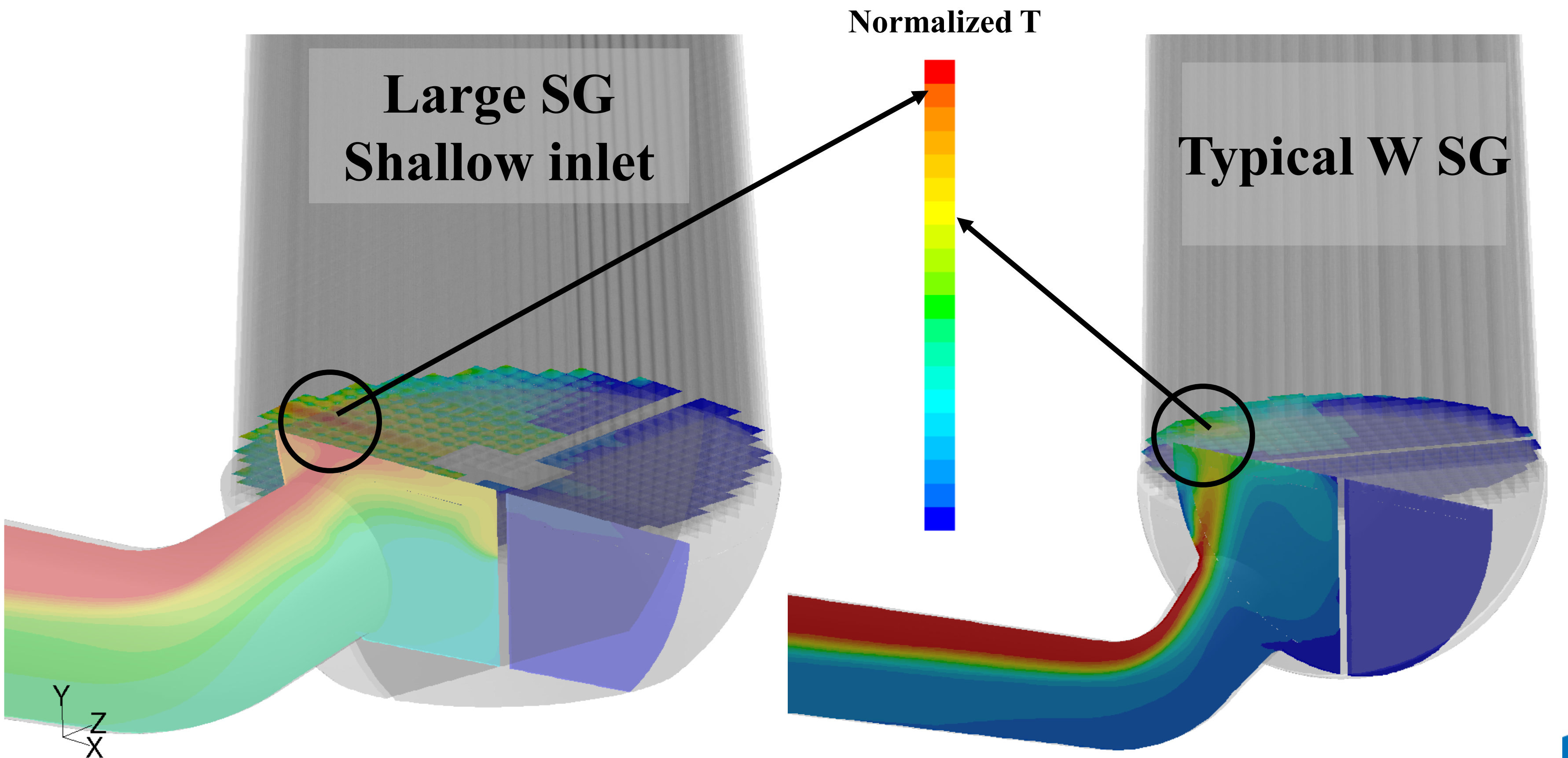
-pump seal leakage
-suction height

-loop seal clearing
-flashing; depth

Surge line Flows and Mixing Predicted



Impact of Inlet Plenum Mixing
CFD Predictions for two SG designs

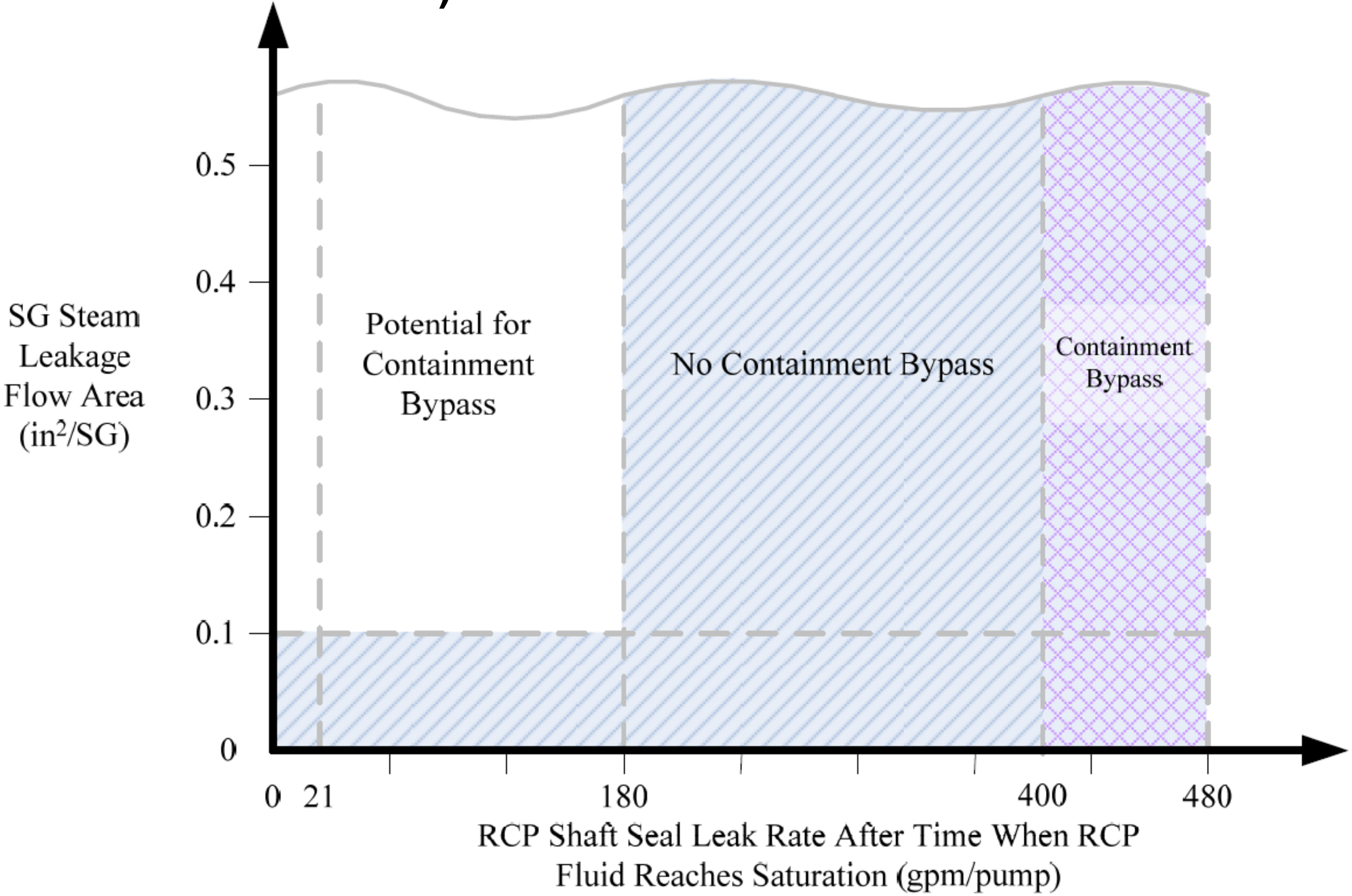


inlet plenum geometry affects mixing

(temperature contours shown)

Sample Map of Containment Bypass Potential

Considering Primary and Secondary Side Leakage Rates (no operator intervention)



What happens to FPs that make it to SG? (1/2)

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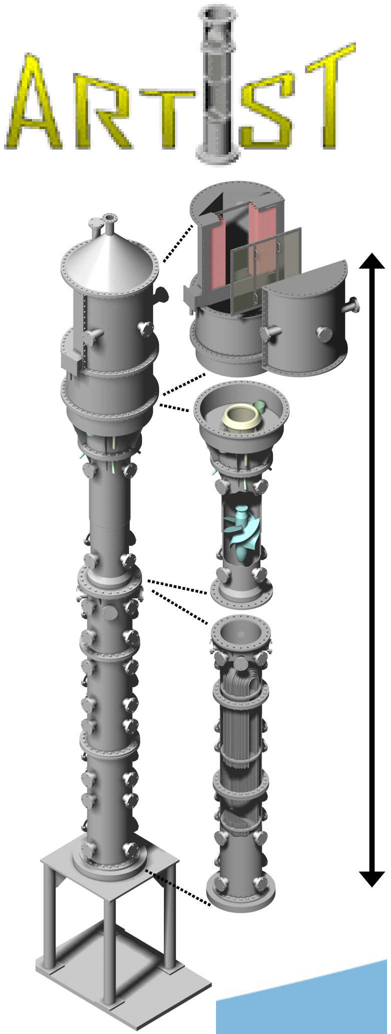
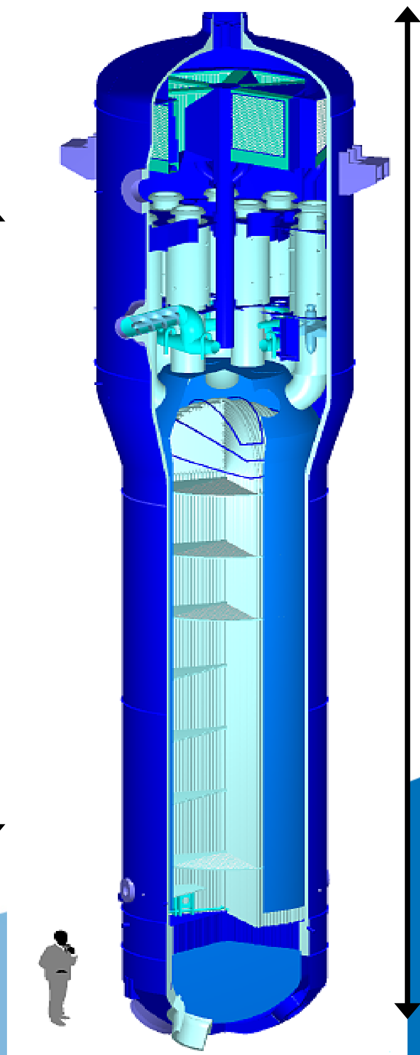
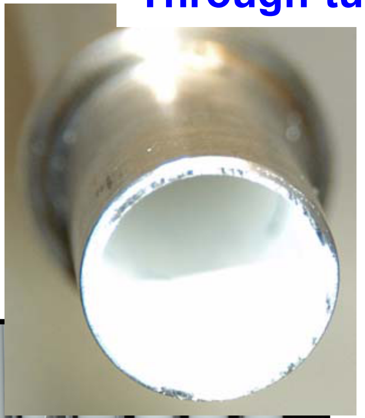
- Discrepancies in predictions of SG decontamination factors (DFs) = FP mass into tubes/FP mass out of SG
  - Predictions range from 5 to 10,000
    - Affects risk importance of this type of accident
  - To resolve this issue, NRC participated in the AeRosol Trapping In a STeam generator (ARTIST) project
    - Multinational project, conducted at PSI in Switzerland, involved Separate Effects tests and Integral tests of decontamination for both dry and wet conditions



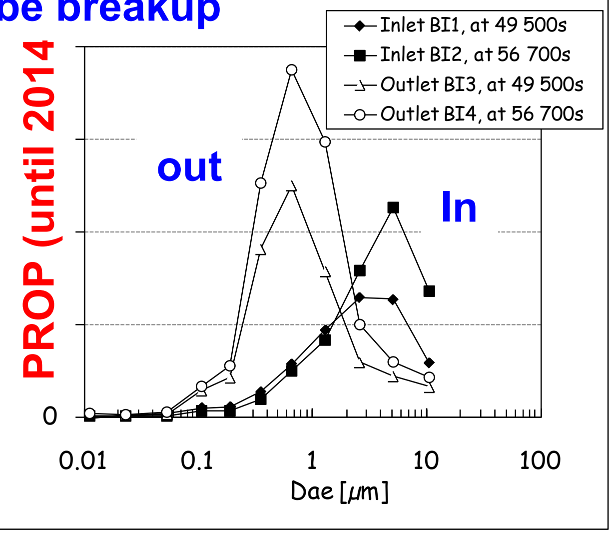


What happens to FPs that make it to SG? (2/2)

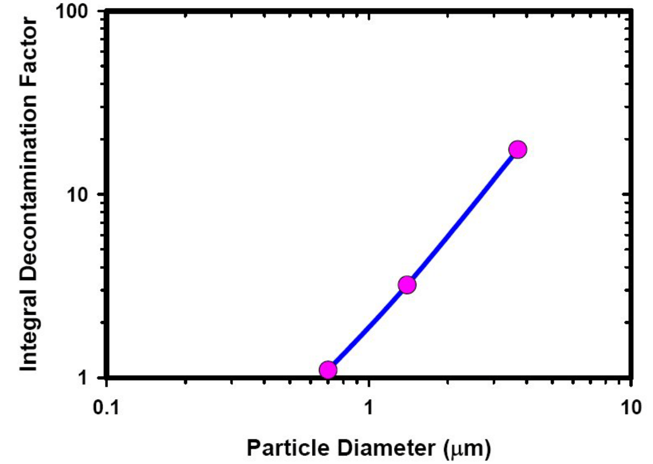
- What was found:
  - Agglomerates can break-up when going through tubes
    - FPs emanating from degrading core are multi-component agglomerates
  - Particles can bounce
  - Low decontamination observed on SG secondary side



Through-tube breakup



Integral Decontamination



Technical Approach

Richard Lee, Charles Harris,
Raj Iyengar, and Selim Sancaktar
RES

TH Analyses

- Update existing CFD and system code models for a CE plant
- Provide un-failed thermal hydraulic behavior for selected accidents (Item1.1.A.ii)
  - Boundary conditions for failure calculations (T, P)
    - spatially variant tube T
  - TH uncertainty estimate
  - Component failure time estimates
  - Run needed sensitivities (complementary to prior analyses)
- Provide failed thermal hydraulic and volatile (Cs, I, Te) releases based upon provided failures
  - Potential iterative process with failure models needed to obtain releases.
    - Preliminary calculations indicate that temperatures in CE SG will be hot enough for unflawed tubes to fail prior to other RCS components.
      - Likely sufficient to depressurize system preventing failure of other RCS components
- Provide assessment impact of instrument tube failures for Westinghouse and CE plants

TH status

- Update existing CFD and system code models for a CE plant (Calvert Cliffs)
  - Generate CFD model of CE hot leg and SG lower plenum
    - Obtained plant info, drawings.
    - Preliminary CFD model developed. Running initial calculations.
  - Generate MELCOR CE deck
    - Obtained some plant info, drawings, R5 deck.
    - Obtained previous MELCOR and SCDAP/RELAP MELCOR decks
    - Deck generation in progress - building upon pre-existing CE (MELCOR and SCDAP/RELAP) decks.
    - Taking into account lessons learned from the previous C-SGTR analysis
  - Communication between MELCOR and FLUENT deck developers
    - ensure consistency between decks
    - Provide mixing parameters
- TH analyses will be conducted with these models
- Will use results of pre-existing analyses for Westinghouse plants if needed
- Instrument tube failure impact
  - Review of existing analysis – due to lower priority and later deadline, will focus on subsequent to TH calculations

TH Uncertainties

- Base failure timing calculation (tubes & RCS components)
  - Relative failure timing (tubes vs RCS)
- Major TH uncertainties identified in previous analyses – considering:
  - Loop seal clearing – limiting calculations, don't expect a definitive answer
  - Pump shaft seal leakage sensitivity
  - Secondary leakage sensitivity
  - TDAFW availability sensitivity
  - Battery availability sensitivity
  - Stress multiplier sensitivity

Flaw Distribution in SGs

Condition of SG Tubes

- Represent current fleet
  - Describe flaws in CE, W, B&W
    - Number, size
    - Type, location
    - Total leak area
  - New Materials
    - Alloy 600TT, alloy 690

Condition of SG Tubes

- Update NUREG on flaw distributions
  - NUREG/CR-6521 (1998)
    - Original statistics still valid
    - 1998 - applied to Alloy 600MA

    - Adjust for new materials
    - Incorporate newer ISI data
      - number, size, type, location

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Failure of RCS Components

Failure Prediction of RCS Components

Tasks

- Identify, characterize, and model relevant RCS nozzles to assess their potential for failure during a severe accident for both Westinghouse and CE plants

- Develop finite-element models, addressing variables such as nozzle geometries/configurations, boundary conditions, loading conditions, fabrication effects, primary water stress corrosion cracking mitigations, and degraded conditions

Challenges

- Develop failure model for critical RCS components based on numerical experiments – for consistency with the tube rupture assessment

- Resulting methodology will be more conducive to the procedure adopted in the C-SGTR risk assessment method to be developed as part of the Task 1.3.A

Failure Prediction of RCS Components

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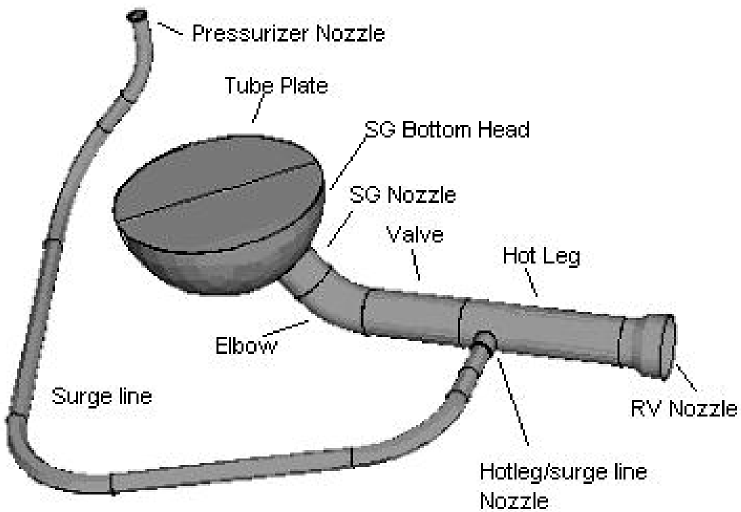
Approach

Validate three-dimensional sub-model of Hot-leg nozzle with shell model of the hot-leg to surge line. This would allow for the development of failure envelope of generic hot-leg nozzle for different thickness of pipe and overlay welds.



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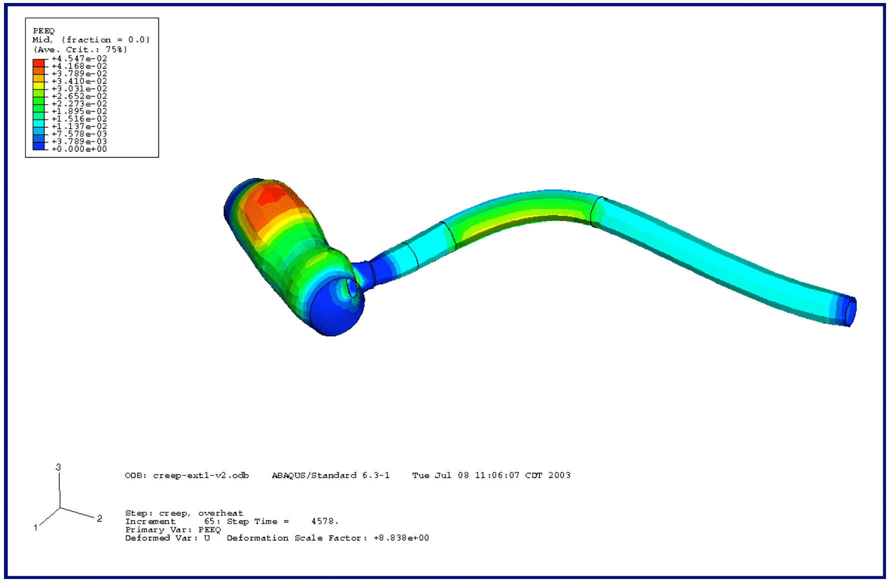

Software Tool

ABAQUS - general purpose finite element analysis software will be used to predict failure time of hot-leg nozzle. Weakest link - the hot-leg nozzle(previous ANL study)

Uncertainties

- Material Properties – Data available
- Geometry - Geometric dimensions, Defect, Weld Overlay
- Failure Models – Creep Rupture, Tensile Properties
- Thermal Properties – Conductivity, Thermal expansion coefficient

PRA-Related Activities

User Need Summary

PRA-related activities are captured in tasks 3 and 4 of the user need.

3.A A user-friendly methodology for assessing the risk associated with consequential tube rupture/leakage in DBA and severe accident events.

3.B A reassessment of the conditional probabilities of C-SGTR based on updated flaw distributions and updated T-H analyses.

3.C Regulatory guidance on risk-informed decision-making regarding C-SGTR.

4. Report compiling and summarizing key research, building upon NUREG-1570, work performed as part of SGAP activities, and this user need.

Current Activities

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- Two PRA-related projects are underway:

1. A contract was recently placed for creation of a PRA report to address task 3.A
2. A second contract is underway to create a C-SGTR calculator to estimate SG tube leakage probabilities under different conditions and for different SG designs.

PRA Report

- Task 3.A requires that
  - a simplified method for assessing the risk associated with C-SGTR events is to be developed and its use is illustrated taking advantage of updated SG and T-H data.
  - the method should be based on standard PRA techniques and the reference documents supplied by the NRC and should be documented in a report acceptable to RES and the NRR.
  - The method should address design basis accident and severe accident events.
- The report will support risk-informing the regulatory process by assisting the NRC staff to make risk informed decisions concerning C-SGTR events.

- The method and the report will be used to facilitate the quantification of C-SGTR events in future NRC and/or licensee risk models, and the development of guidance for future risk assessments.

C-SGTR Calculator

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- A software package is developed to estimate SG tube leakage probabilities for given RCS and secondary side conditions (scenario parameters)

- The basis document for the software is being peer reviewed by expert(s) cognizant with the subject matter.

PRA Effort - Conclusion

- The PRA report and the C-SGTR calculator are expected to be ready within the next two years, after incorporating input from other disciplines (T&H analyses, behavior of other RCS components, additional SG tube failure data, etc.).

- Afterwards, the task of providing regulatory guidance on risk-informed decision-making regarding C-SGTR can be addressed.

CONCLUSION

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- A multi-year project involving interdisciplinary technical work by several RES divisions
- A comprehensive project plan developed
- Ongoing continuous engagement and coordination with various divisions

ACRS C-SGTR Subcommittee Items

No.	SUBCOMMITTEE MEETING DATE	SUBCOMMITTEE QUESTIONS/COMMENTS	CONTEXT	AREA	LEAD(s)	ACTION / DISPOSITION
1	4/6/11	Several members questioned the benefit of this study if it were completed. Clearly indicate the staff's views with respect to the benefits of this study (and how much uncertainty will be reduced when it is completed). Possible answers include: - Incorporates CFD analysis - Includes new data - Simpler guidance (consider proposed actions with respect to DBE versus severe accident space). - Document state-of –the –art -Insights about activities to mitigate cracking, tube repair criteria . - Guidance on additional experiments In addition, provide a project/execution/decision diagram that illustrates what will be done with information obtained from this study.				
2	4/6/11	Several members commented that the staff should update documentation to: - Note that EPR SGs will have a similar issue as SGs in the CE plant (note relationship) -Note that AP1000 uses a CE SG (so it may also be applicable); -Note that W replacement SGs may actually be CE designs.				

ACRS C-SGTR Subcommittee Items

No.	SUBCOMMITTEE MEETING DATE	SUBCOMMITTEE QUESTIONS/COMMENTS	CONTEXT	AREA	LEAD(s)	ACTION / DISPOSITION
3	4/6/11	Several members expressed concern about existing data and questioned if additional data were needed. Please provide staff views related to: • What additional experiments are needed to 'believe' CFD results? Should 1/7th scale data be redone? The 1/7th scale had geometrical connection distortions (mixing parameters may not be appropriate). Can you benchmark against it and then redo the CFD calculation for the appropriate geometry? • Data for updating tube failure criteria? Quality of updated information for flaw distributions. Adequacy of experimental data?				•

ACRS C-SGTR Subcommittee Items

No.	SUBCOMMITTEE MEETING DATE	SUBCOMMITTEE QUESTIONS/COMMENTS	CONTEXT	AREA	LEAD(s)	ACTION / DISPOSITION
4	4/6/11	Members questioned the impact from running auxiliary feedwater flow directly into the steam generators during surveillance testing rather than using a recirculation line. Some plants evidently do this just prior to startup or shutdown and a member postulated that the introduction of cold water from the condensate storage tank might represent a degradation mechanism for the steam generators.			Murphy/ Mirela	*RES* expects that this would be detected along with all other flaws during ISIs. Since they are going to generate the flaw distribution inputs using ISI data, flaws induced by cold water injection would be accounted for in the analyses. If there are other transients within the design basis that may lead to consequential tube leaks or ruptures, these should be included within the scope.
5	4/6/11	Should flaw shapes be considered in this study? Members indicated that the staff needed to look at realistic issues in the plant.			Csontos	RES is trying to decide on the level of detail in the study.
6	4/6/11	What is industry doing in the area? Does industry believe that existing SAMGs are adequate? If not, can updating the SAMGs be incorporated into the NRC effort?			Csontos	EPRI has developed a methodology and Westinghouse has addressed the possibility of SGTR.

ACRS C-SGTR Subcommittee Items

No.	SUBCOMMITTEE MEETING DATE	SUBCOMMITTEE QUESTIONS/COMMENTS	CONTEXT	AREA	LEAD(s)	ACTION / DISPOSITION
7	4/6/11	ACRS needs clarification on the User Need task. User Need states the following: NRR has requested that RES modify the C-SGTR assessment method to include consideration of tube failure in once-through steam generators (OTSGs). The axial loads on OTSG tubes during certain design-basis accidents (e.g., loss of coolant, steam line break, and loss of main feedwater), because of thermal expansion differences between the SG shell and the SG tubes, could create high axial stresses on the tubes, potentially resulting in tube rupture			Iyengar	*No answer from RES yet*
8	4/6/11	ACRS Subcommittee Members are concerned that RES may not be identifying all scenarios. RES needs to get a better handle on how this study will be used by NRR. J. Stetkar	Quantifying Scenarios		Sancaktar	*No answer from RES yet*
9	4//6/11	How good is the industry flaw distribution update? How good is the data and the quality.	Flaw distribution		C. Harris	There is no industry update on flaw distribution. The information from the industry is contained in the in-service inspection reports. NRC will audit the information, but specifically from EPRI or any of the utilities there's no direct input on flaw distributions.

ACRS C-SGTR Subcommittee Items

No.	SUBCOMMITTEE MEETING DATE	SUBCOMMITTEE QUESTIONS/COMMENTS	CONTEXT	AREA	LEAD(s)	ACTION / DISPOSITION
10	4/6/11	How is the staff currently accomplishing their task since this work is not complete? Is this work better than what the staff is doing now? Corradini	Risk assessment tools		Zoulis	Staff has not dealt with any issues to date relating to C-SGTR. Staff believes that if an accident occurred that they could address the issue. Staff believes that the work in this area will give them a better understanding of the phenomenon of C-SGTR and its implication to risk. Ultimately a simpler tool for detailed assessments.
11	4/6/11	Some members questioned if the staff had considered other, simpler, options that could reduce plant risk from these events, e.g., alternate SAMGs, hardware, etc.				
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15						
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17						

Other Comments

NRC received a NUREG report, "Elevated Thermal Loads for Severe Accident and Design Basis Accidents in OTSG Designs." It is complete, but not yet published. Results indicate no appreciable added effects from elevated temperatures. Also, because of the once-through design, the problems of back-flow steam, actually re-entering the tubes in the reverse direction, is found to be insignificant.