ENCLOSURE 2

MFN 13-096

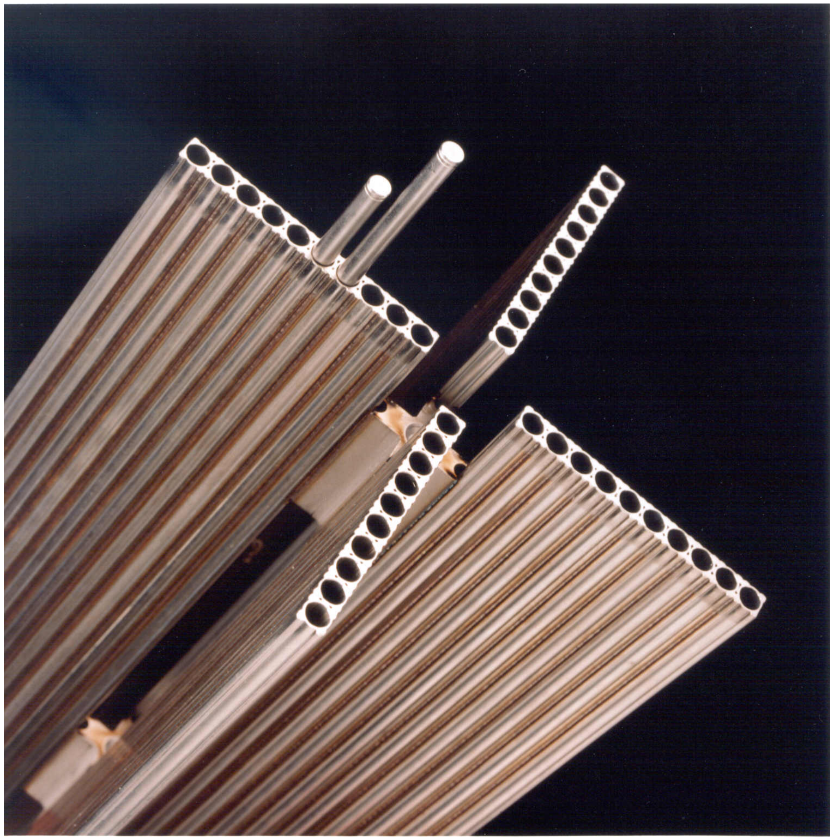
ACRS Subcommittee Presentations

Non-Proprietary Information – Class I (Public)

INFORMATION NOTICE

Enclosure 2 is a non-proprietary version of the ACRS Subcommittee Presentations from Enclosure 1, which has the proprietary information removed. Portions that have been removed are indicated by open and closed double brackets as shown here [[ ]].

Technology Update for the ACRS
November 2013



Control Rods

Scott Nelson

HITACHI

Ultra Control Rod Description

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- •Licensed by NEDE-33284P-A: June 2009
- •[[

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- •Licensed by NEDE-33284 Supplement 1P-A: March 2012
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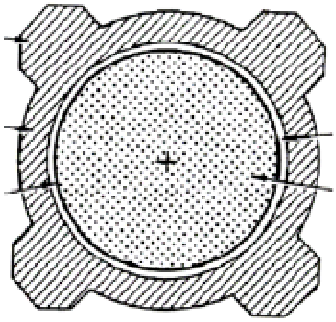
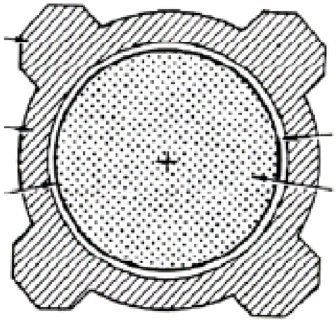
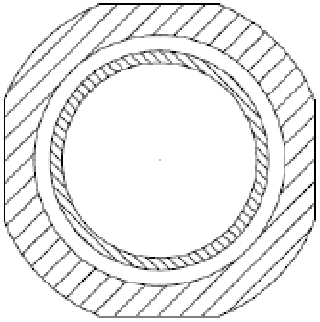
Marathon Design Comparison

Parameter	Marathon D/S	Marathon C	Ultra
Absorber Tube			
Local Boron-10 Depletion at Capsule Contact	[[		
Swelling Induced Strain at 100% Local Depletion			]]

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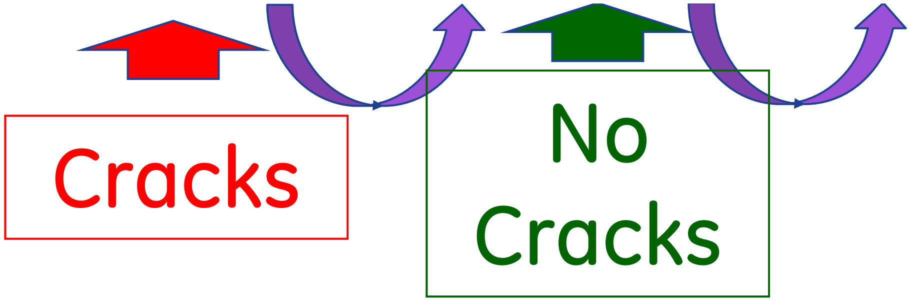

Cracks

No Cracks

Marathon Control Rod Inspections

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- The 2013 annual report is contained in MFN 13-034 (NEDC-33819P)

Marathon Control Rod Inspections

- In February 2011, GEH recommended reduced lifetime limits for D and S lattice Marathon control rods, based on the observed cracking.
  - The revised lifetime limits are contained in Safety Communication SC 11-01 (MFN 11-023).
  - [[

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Ultra Control Rod Inspections

- Both Ultra MD and Ultra HD control rods have been inspected.
- No crack indications have been observed.

Plant	Absorber Tube Type	Control Rod Type	Serial Number	Ship Year	Inspection Date	Thermal Fluence (snvt)	¼-Segment B-10 Depletion (%)	Peak Local B-10 Depletion (%)	Crack Indications ?
Plant M (US BWR/4)	D/S/N	Ultra MD	[[						
Plant N (Int'l BWR)	D/S/N	Utlra MD							
Plant R (Int'l BWR/4)	D/S/N	Ultra MD							
Plant N (Int'l BWR)	D/S/N	Ultra HD							
Plant M (US BWR/4)	D/S/N	Ultra MD							]]

Planned Inspections

Plant	Absorber Tube Type*	Control Rod Type	Planned Inspection Date	Number of CRBs to be Inspected	Thermal Fluence (snvt)	¼-Segment B-10 Depletion (%)	Peak Local B-10 Depletion (%)
Plant N (Int'l BWR)	D/S/N	Ultra MD	[[				
Plant M (US BWR/4)	D/S/N	Ultra MD					
Plant N (Int'l BWR)	D/S/N	Ultra HD					]]

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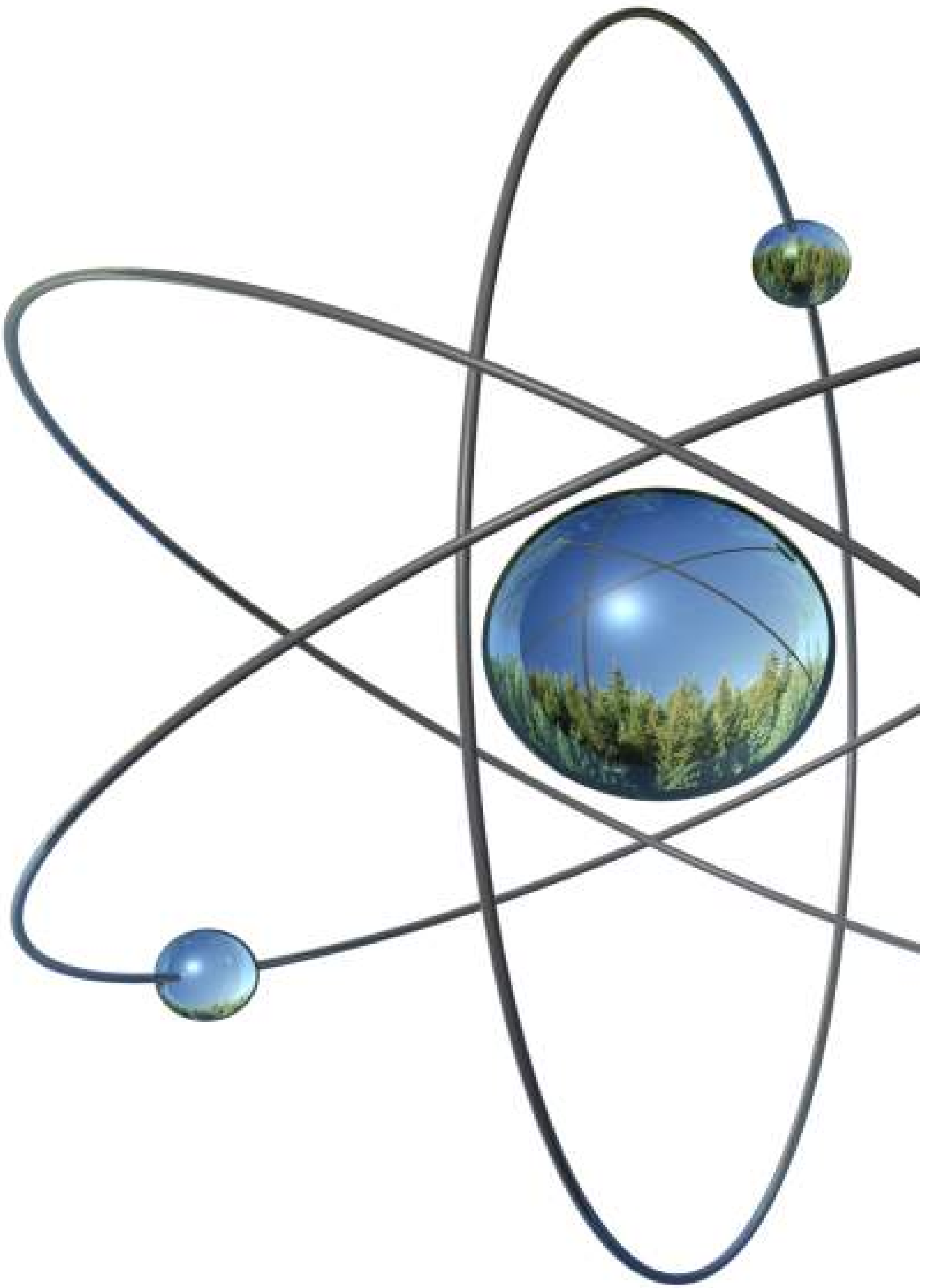
GE Hitachi Nuclear Energy

GEH Fuel Performance Update
Presentation to the ACRS
November 21, 2013

ATWS/I Methods

Jens Andersen, Ph. D.
Chief Consulting Engineer, Thermal Hydraulics

Charles Heck
Consulting Engineer



TRACG04 ATWS/I Methods

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Rewetting of hot surfaces

- Minimum film boiling temperature – T_{min}
- Quench front propagation
- Implementation
- Validation

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Minimum film boiling temperature – Tmin

- TRACG04 uses the Shumway correlation (EGG-RST-6781)

$$T_{min}=T_{sat}+3.7\frac{\rho_\ell+\rho_v}{\Delta\rho}\frac{h_{fg}}{C_{p\ell}Pr_\ell}\left(\frac{\rho_\ell k_\ell C_{p,\ell}}{\rho_w k_w C_{p,w}}\right)^{1/2}\left(1+\cancel{(1-\alpha)^2}\right)\left(1+1.5\times10^{-5}Re_\ell\right)^{0.15}\left(1-\frac{P}{P_{crit}}\right)^{0.1}$$

- Includes material property, void and mass flow dependency
  - 0.4 < P < 9 MPa
  - 0.1 < Re < 7x10^5
  - [[ ]] Stainless steel data
- Void dependency untested and predicts higher T_{min} at low void fractions
  - Void dependence disabled in TRACG04
- Flow and pressure dependence supported by data presented in EGG-RST-6781
- Material property dependence supported by additional experimental data

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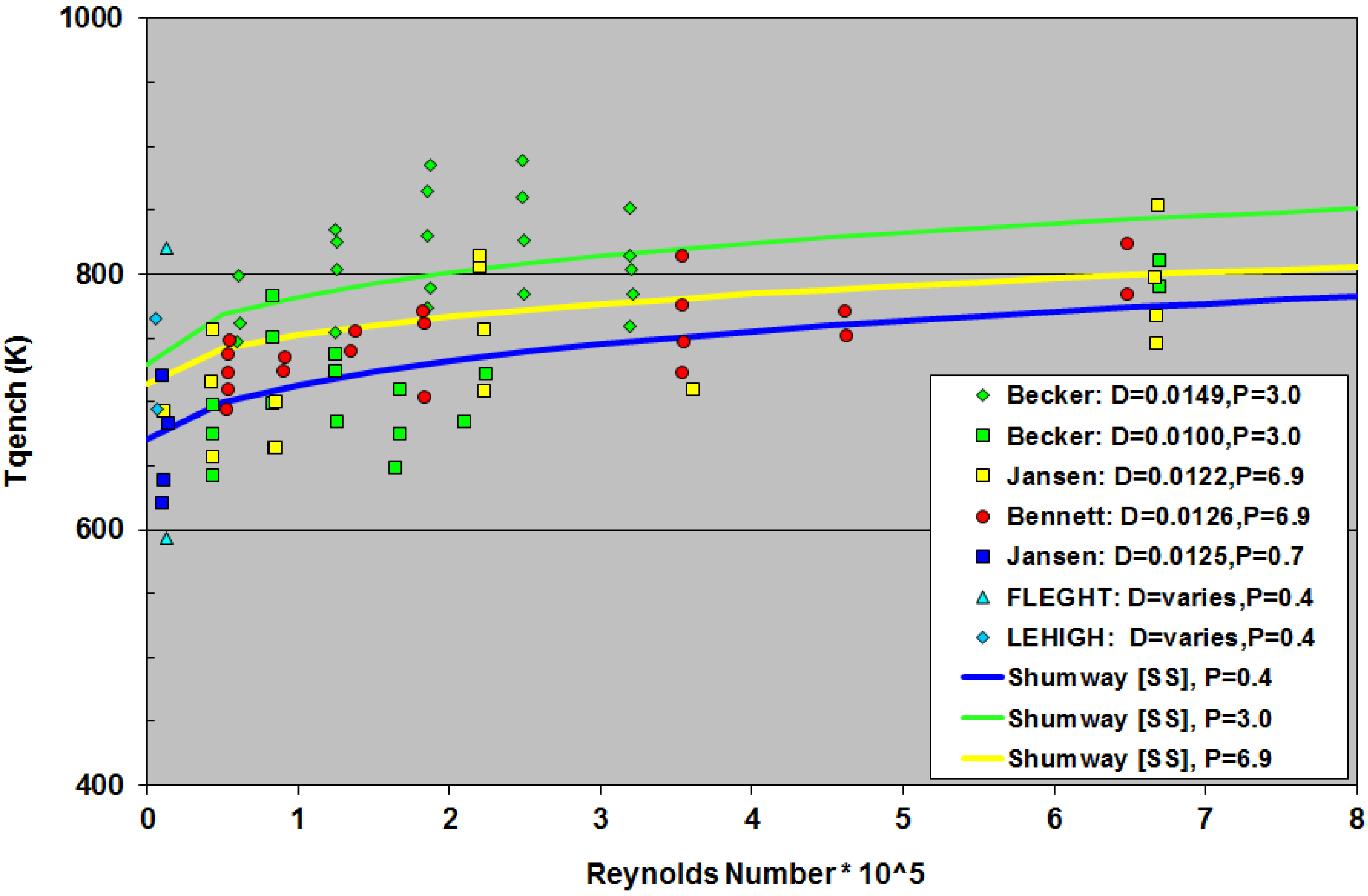
Shumway Correlation versus SS Data



Tmin Data indicates Max value at intermediate pressure

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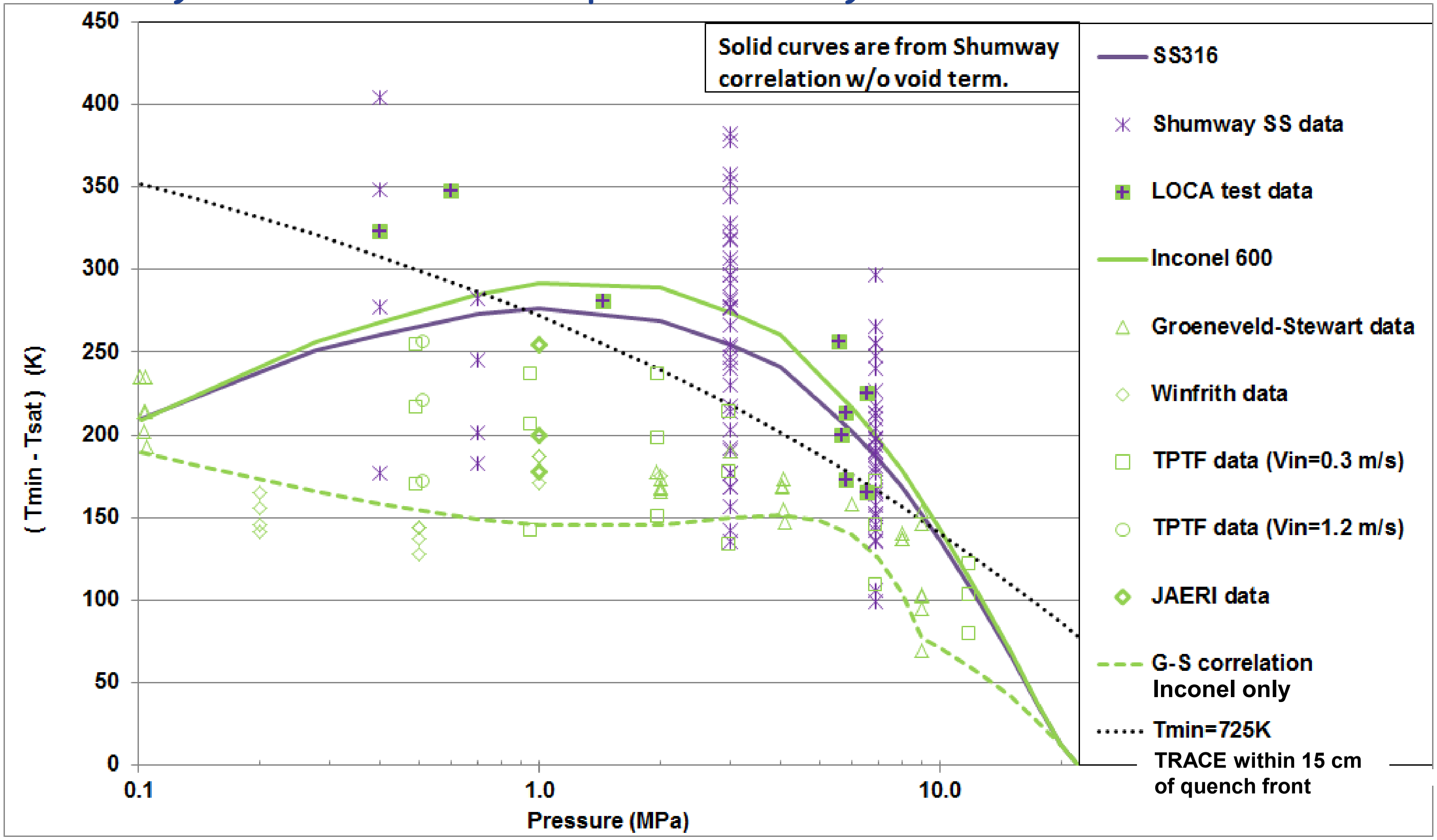
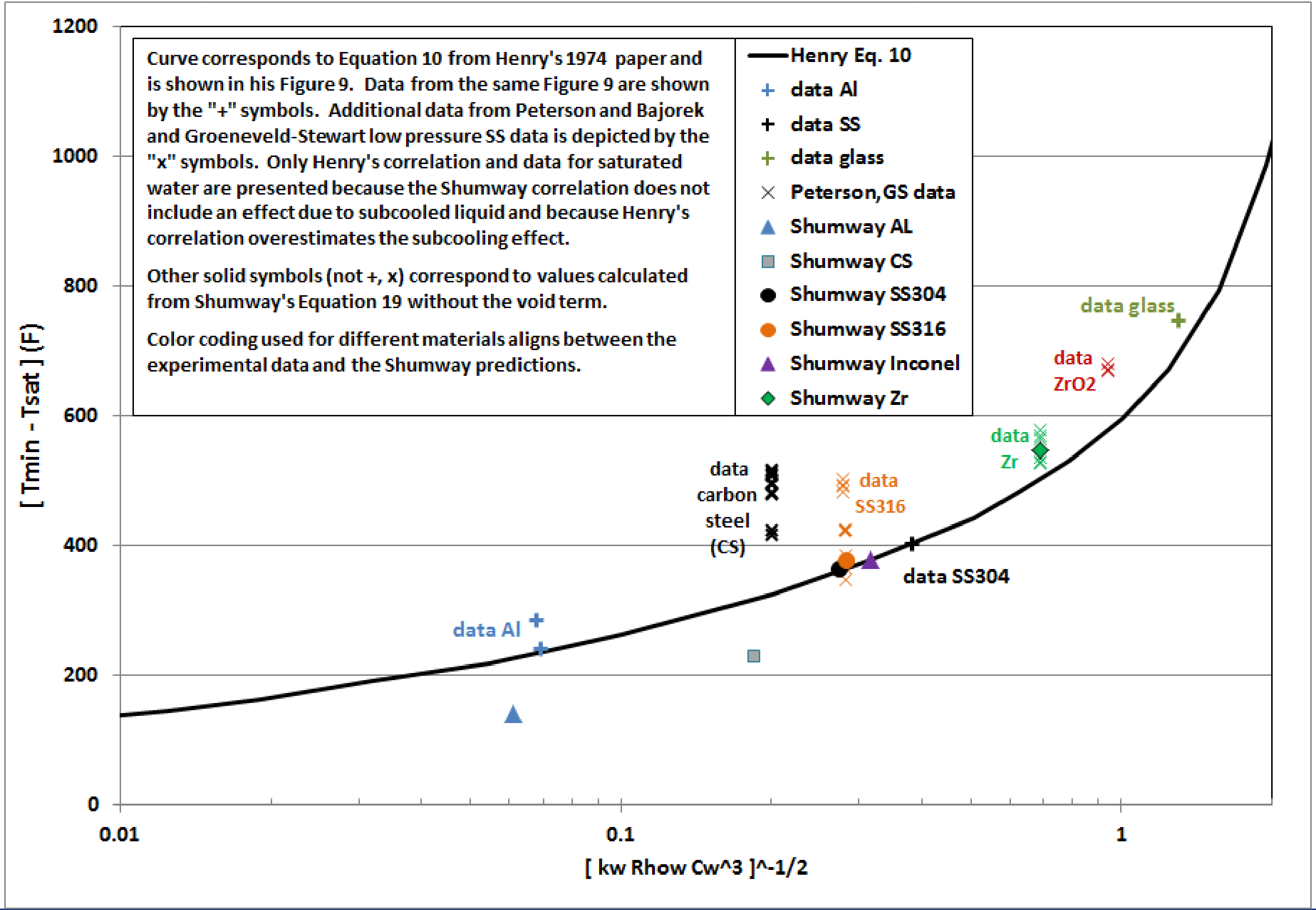
Shumway Pressure Dependency vs. SS & Inconel Data



Shumway Correlation matches Data Trend vs. Pressure

Shumway Tmin Dependency on Materials



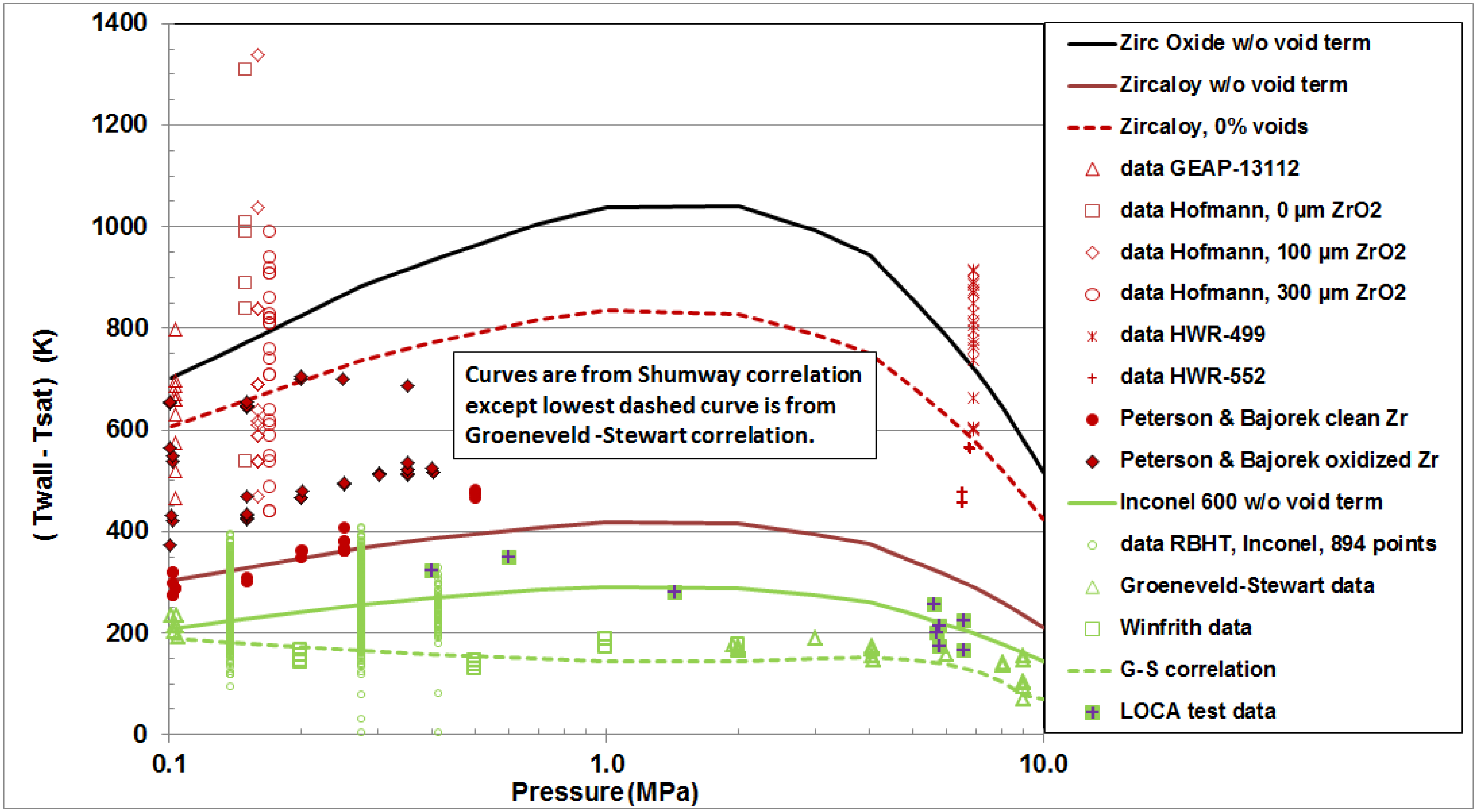
Curve corresponds to Equation 10 from Henry's 1974 paper and is shown in his Figure 9. Data from the same Figure 9 are shown by the "+" symbols. Additional data from Peterson and Bajorek and Groeneveld-Stewart low pressure SS data is depicted by the "x" symbols. Only Henry's correlation and data for saturated water are presented because the Shumway correlation does not include an effect due to subcooled liquid and because Henry's correlation overestimates the subcooling effect.

Other solid symbols (not +, x) correspond to values calculated from Shumway's Equation 19 without the void term.

Color coding used for different materials aligns between the experimental data and the Shumway predictions.

Shumway Correlation follows Trend Observed by Henry

Shumway T_{min} vs. Zirc T_{quench} and Inconel & SS Data



Curves are from Shumway correlation except lowest dashed curve is from Groeneveld -Stewart correlation.

Minimum film boiling temperature – Tmin

- Shumway correlation (EGG-RST-6781)

$$T_{min} = T_{sat} + 3.7 \frac{\rho_\ell + \rho_v}{\Delta\rho} \frac{h_{fg}}{C_{p\ell} Pr_\ell} \left(\frac{\rho_\ell k_\ell C_{p,\ell}}{\rho_w k_w C_{p,w}} \right)^{1/2} \left(1 + \cancel{(1-\alpha)^2}\right)\left(1 + 1.5\times10^{-5} Re_\ell\right)^{0.15} \left(1 - \frac{P}{P_{crit}}\right)^{0.1}$$

- Flow and pressure dependence supported by data presented in EGG-RST-6781
- Material property dependence supported by additional experimental data
- Void dependency untested and predicts higher T_{min} at low void fractions
  - Void dependence disabled in TRACG04
- Shumway correlation conservatively does not credit liquid subcooling

Shumway correlation is conservative compared to Zr data

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Quench front propagation

- TRACG04 uses an empirical correlation that matches the one- and two-dimensional conduction solutions (NEDE-32176P, Rev. 4)

$$Q_{quench} = Sk_w (T_w^+ - T_{sat}) \left(\overline{Bi} (1 + 0.4 \overline{Bi}) \right)^{0.5}$$

- Where $\overline{Bi} = \frac{Bi}{\overline{T}^2}$ $\quad Bi = \frac{h_q d_w}{k_w}$ $\quad \overline{T} = \frac{\sqrt{\Theta}}{1-\Theta}$ $\quad \Theta = \frac{T_w^+ - T_o}{T_w^+ - T_{sat}}$

- Quench front heat transfer implemented as additional heat transfer from node containing quench front.

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TRACG04 Implementation

Energy balance for nodes

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Quench Front Heat Transfer Coefficient

$$\mathrm{Bi} = \frac{h_q d_w}{k_w}$$

- Bottom flooding quench heat transfer based on Yu, Farmer and Coney and implemented as described in NUREG/CR-2178 and encoded in the original version of TRAC that formed the basis for the BWR versions

$$h_q = \left(\frac{F_q}{\Delta T_q} \right)^2$$

$\Delta T_q = T_o - T_\ell$ Is the difference between the Leidenfrost temperature and the liquid temperature

$$F_q = \alpha F_s$$

$$F_s = 4.24 \cdot 10^4 v_\ell^{0.15}$$

$$\alpha = \begin{cases} \left(1 + v_\ell \Delta T_\ell^2\right)^{0.13} & \text{for } (1 + v_\ell \Delta T_\ell^2) \le 40 \\ 0.4839\left(1 + v_\ell \Delta T_\ell^2\right)^{0.346} & \text{for } (1 + v_\ell \Delta T_\ell^2) > 40 \end{cases}$$

The liquid subcooling $\Delta T_\ell = T_{sat} - T_\ell$ is used in this equation for α. ΔT_q was incorrectly used

- This error has been corrected in TRACG04 and its impact evaluated per 10 CFR Part 21

Validation

- Halden
- THTF
- TLTA
- ROSA III

- All calculations made with corrected TRACG04 code
  - Shumway T_{min} – void dependence disabled
  - Quench model heat transfer coefficient corrected.
  - Heat transfer for node containing quench front calculated separately for quenched and dry part.

Halden

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Test 3

Test 4

- Flow updated based on electronic data received from Halden

Halden

[[

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Test 4 – Sensitivity to Quench model and T_{min}

Halden

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Test 11

Test 12

- Flow updated based on electronic data received from Halden

Halden

[[

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Test 12 – Sensitivity to Quench model and T_{min}

THTF

[[

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ROSA-III

[[

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Summary

- TRACG04 methods status
  - Shumway T_{min}
    - Void dependence disabled
    - Material property dependence justified
    - Zr data conservatively predicted
  - Quench front correlation
    - Error in quench front heat transfer coefficient corrected
    - Heat transfer for node containing quench front calculated separately for quenched and dry part.
- Validation expanded
  - Halden, THTF, TLTA, ROSA-III
  - Quench model essential for good comparison to all tests
  - TRACG04 PCT prediction in good agreement with data
- References:

– Tmin	ML13253A128
– Quench front	ML13289A211
– TRACE V5 Theory Manual	ML120060218

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GE Hitachi Nuclear Energy

GEH Fuel Performance Update
Presentation to the ACRS
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TRACG Containment Application

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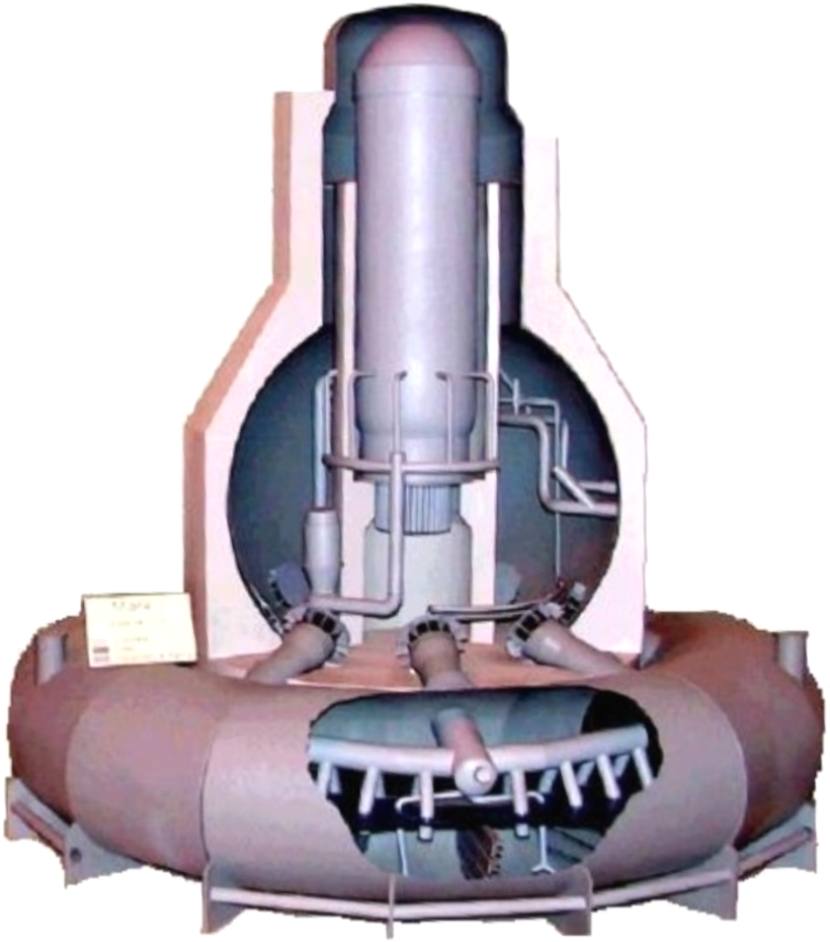
Consulting Engineer



Overview

- New Technology Introduction (NTI), New Product Introduction (NPI) multi-year program to develop detailed TRACG coupled Reactor Pressure Vessel (RPV), Reactor Coolant System (RCS), and containment models for operating BWRs
- Intended first to address extended Station Black Out (SBO) scenarios like the one at Fukushima-Daiichi (1F1, 1F2, 1F3)
  - Extend coping time, evaluate coping procedures, optimize equipment performance, evaluate new equipment options
- Protect public **AND** preserve asset by NOT
  - Damaging core or over pressurizing containment

Detailed Model Design Elements

(implemented since Conceptual Design Review)

Better initialization and distribution of temperatures in the DW due to:

- Spatial modeling of *penthouse* and *dog house*
- Fan coolers modeled with ability to set relative humidity
- Model for mirror insulation on RPV and piping
- Added heat structures and heat transfer to/from piping within containment

Integrated RCIC to assess changes with time of coupling between RCS and containment

- Detailed RCIC subsystem modeling that simulates RCIC performance as pressures and temperatures change

Summary: Nodalization Comparisons

Number of nodes determines spatial resolution

Model Area	TRACG	MELCOR	MAAP
core region including fuel chans & bypass	114 to 814	< 30	~ 25
RPV excluding core region	47*	7	7
drywell	87	4	4
pedestal cavity	16	1	1
wetwell vapor space	16	1	1
suppression pool	16	1 (8,16 ORNL)	1
DW/WW connection	5 cells in PIPE	1 (8 ORNL)	1
reactor building	173 available	12 A model ~25 B model	< 25

- Steam separators, jet pumps, and control blade drive cells occur within the vessel component and are not counted as part of the RPV model. Every physical component can be explicitly modeled although typically not required.

Different Tools for Different Uses/Needs

Code **Use**	**Core**	**RPV, RCS**	**Containment**
SHEX **SBO scoping**	none	boundary conditions	Simple lumped modeling
MAAP, MELCOR **Severe accident**	simple, few nodes	simple, few nodes	simple, few nodes
GOTHIC **Containment**	none	boundary conditions	detailed, many nodes
MAAP→GOTHIC **Severe accident**	MAAP: core during severe accident	MAAP: simple, few node	GOTHIC: detailed
TRACG→GOTHIC **Coupled system, extended SBO**	TRACG: detailed	TRACG: detailed	GOTHIC: detailed
TRACG integrated **Coupled system, extended SBO**	TRACG: detailed	TRACG: detailed	TRACG: detailed

TRACG Flexibility and Capability

Provides the flexibility to build detailed input models capable of modeling all the BWR geometric features and transient phenomena the RPV, RCS, containment, and reactor building (if needed).

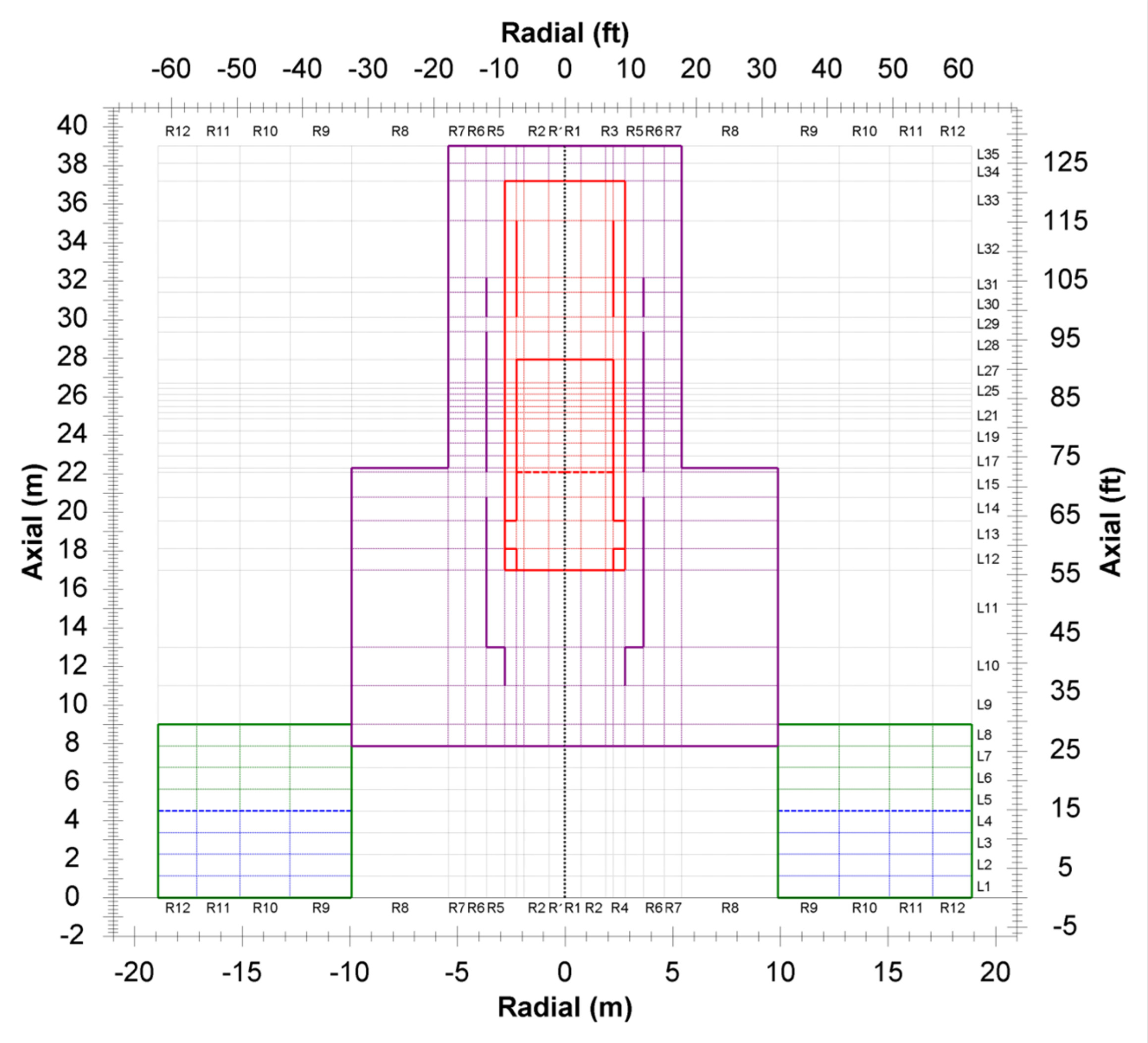
Provides spatial resolution for hydraulic parameters that cannot be matched by MAAP or MELCOR. TRACG better tool to model extended SBO, performance of FLEX equipment, and to calculate H_2 production and distribution prior to core relocation.

Provides control system interfaces so essentially any scenario can be simulated (up to significant core relocation).

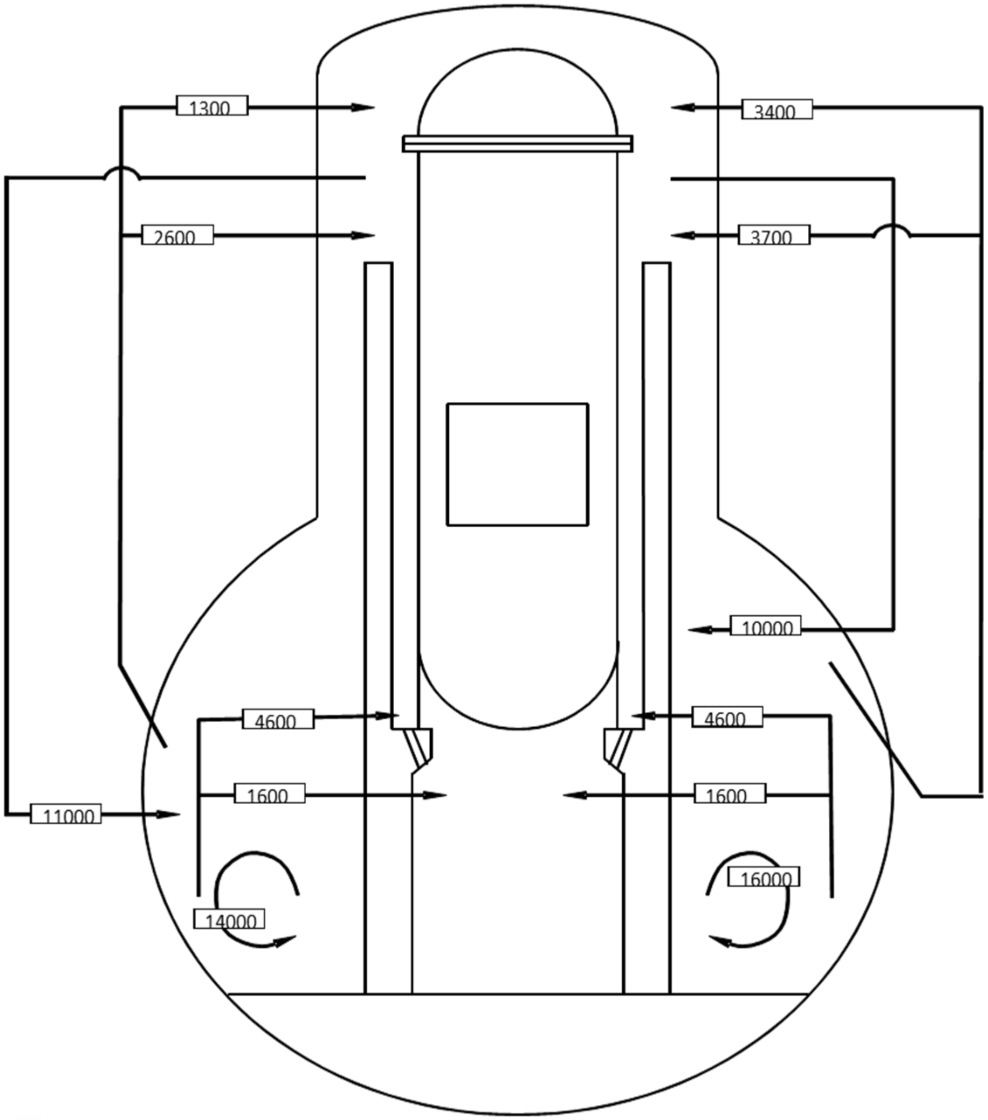
TRACG is best available tool for assessing extended SBO scenarios without significant core relocation. Suitable to optimize equipment and procedures to prevent severe accidents. Can also be used to establish better initial conditions for MAAP or MELCOR severe accident calculations.

TRACG and MAAP or MELCOR have different complementary roles.

TRACG Mark I Model

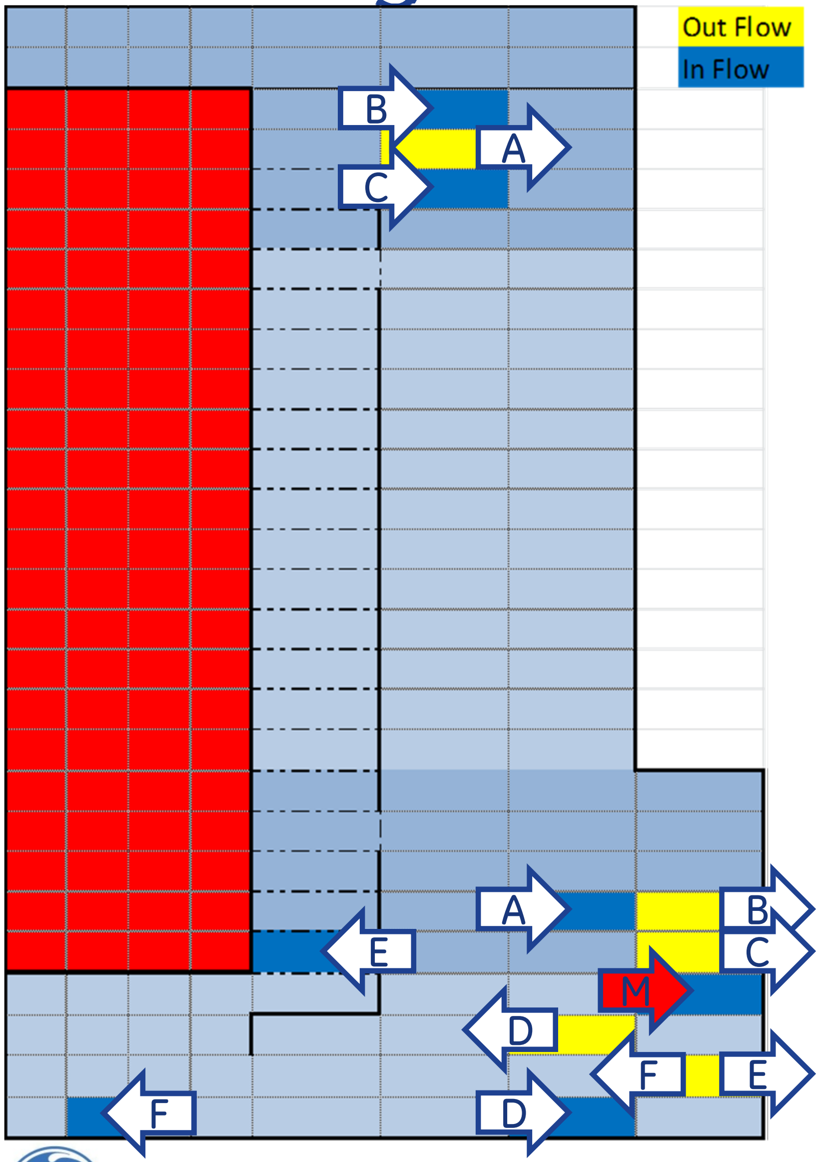


Representative Mark I DW Ventilation



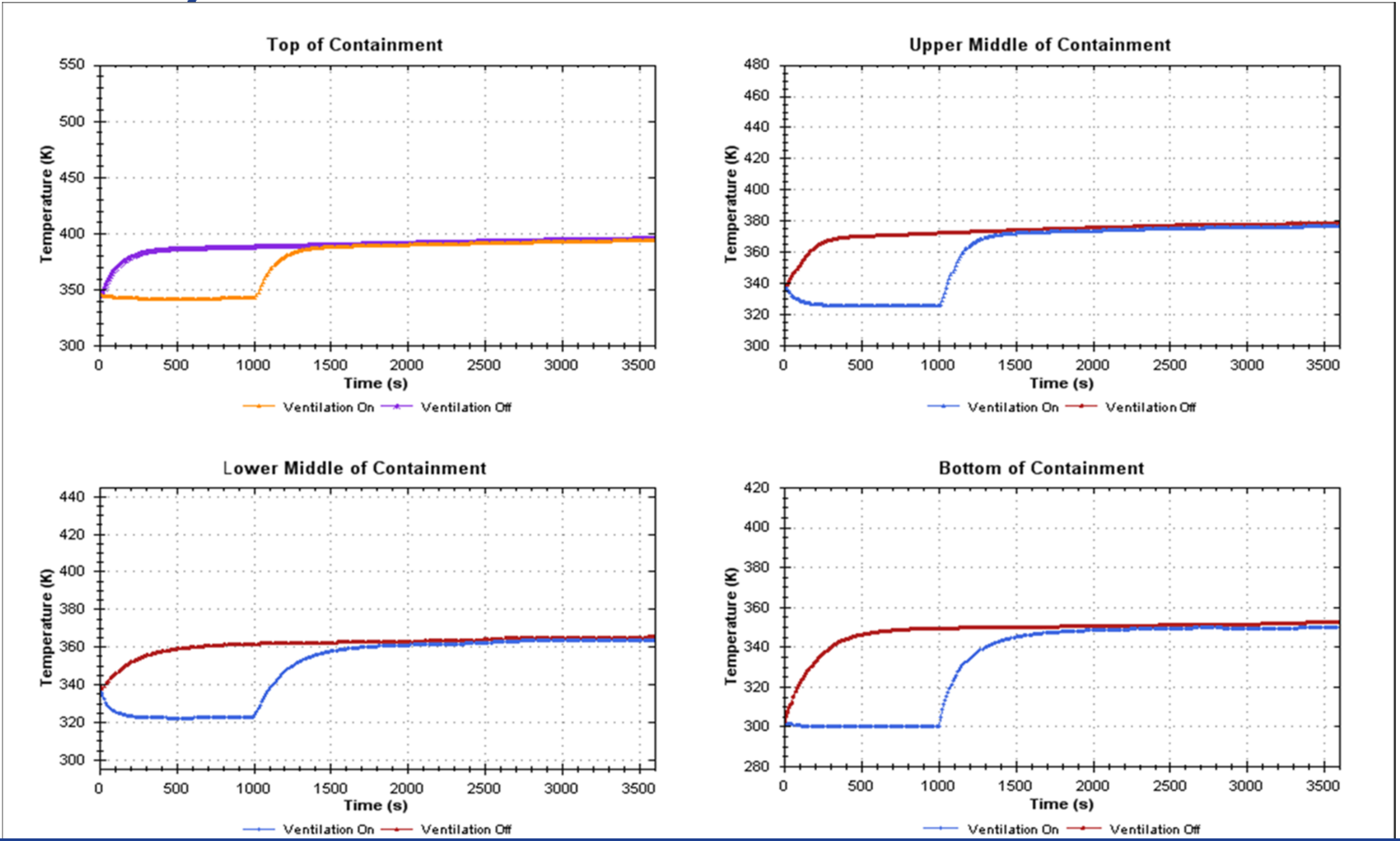
- Plant specific inlets/outlets must be modeled to be able to compare to measured temperatures.
- Volumetric flow rates (cfm) are known inputs.
- Known component cooling water (CCW) temperature determines discharge temperature of the coolers.
- Relative humidity is an input which indirectly sets the cooler efficiency.
- Model conserves NCG masses.
- Energy and H2O condensate are removed.

Modeling of Fan Coolers & Motor Heat



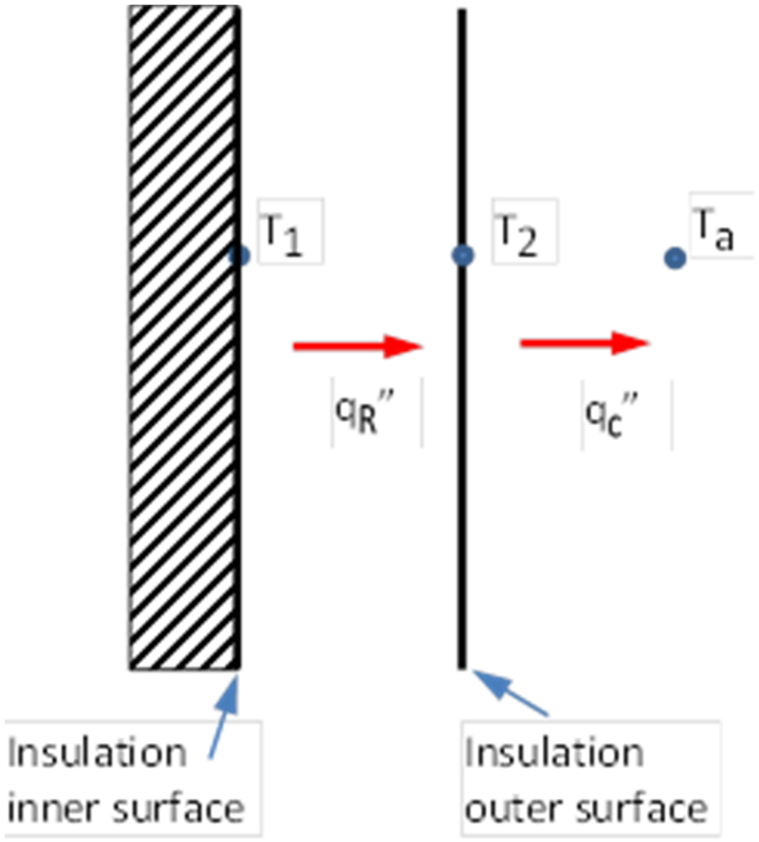
- Each cooler inlet/outlet pair is modeled with a pair of FILLs that are coupled via the TRACG CNTRL system.
- Volumetric flow rate is known and prescribed for each FILL pair based on plant specific values.
- Out-flow conditions are those at the local DW location.
- In-flow temperature is equal the input CCW temperature.
- Specified relative humidity determines H2O vapor remaining at the in flow (also condensing efficiency of the cooler).
- NCG mass flow rate in/out of a FILL pair is conserved although NCG is cooled in the process.
- Energy is removed. Rate of energy removal matches heat loads during steady state.

Example of Fan Cooler Effectiveness



Operating coolers decrease rate of containment temperature and pressure increase and extend time before venting is required.

Mirror Insulation Modeling



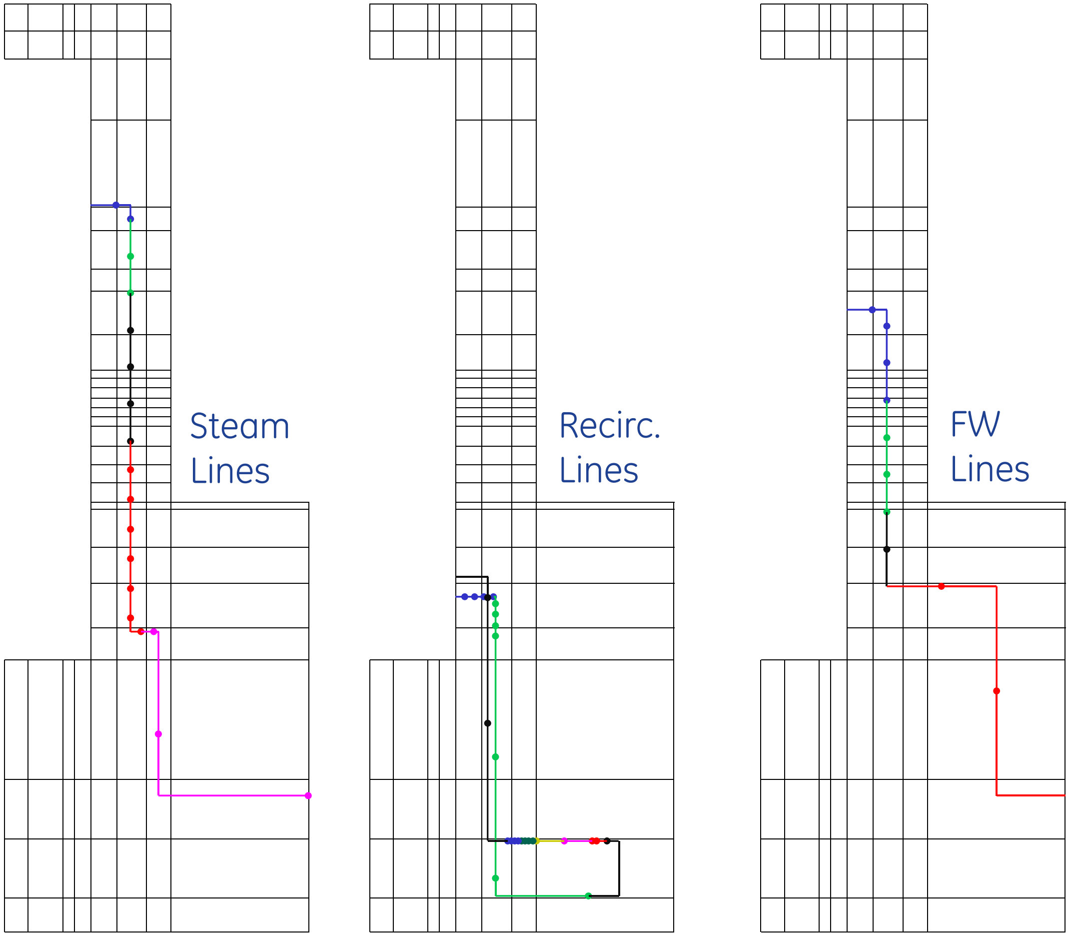
Model is necessary so that DW heat loads are produced at the correct location in the correct amounts.

- Overall heat transfer is limited by thermal radiation between the reflective layers

$$q_R'' = \frac{\sigma}{\frac{2}{\varepsilon}-1}\left(T_1^4 - T_2^4\right) = U_{ref}\left(T_{1,ref} - T_{2,ref}\right)$$

- Convective heat transfer from the outer surface is modeled using existing TRACG models. $q_c'' = h_c\left(T_2 - T_a\right)$
- Quasi-static approximation and neglecting thermal capacity of the insulation yields $h_R\left(T_1 - T_2\right) = h_c\left(T_2 - T_a\right)$
- An overall heat transfer coefficient is thus defined as $h = \frac{q''}{T_1 - T_a} = \frac{1}{\frac{1}{h_R} + \frac{1}{h_c}}$
- Implemented by change in TRACG coding which can be selected via the IOPHTC user input for application to vessel slabs and the outside of piping in the DW.

Heat Transfer to/from Piping

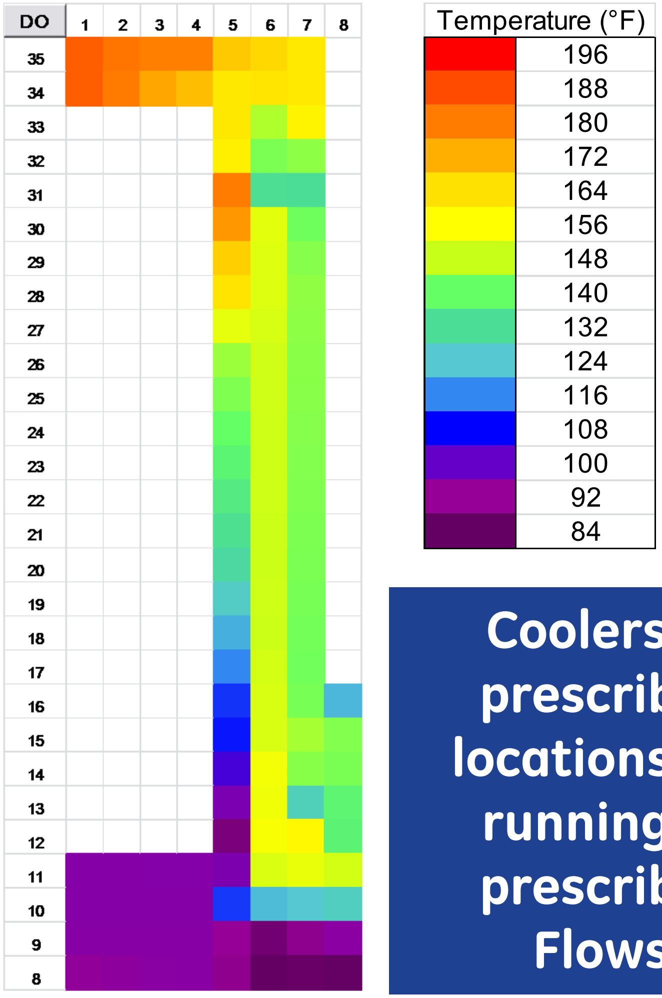


- Apply existing TRACG component-to-component heat transfer capability
- Route path of significant piping thru DW according to drawings (steam lines, recirculation lines, feedwater lines)
- Apply mirror insulation model where appropriate
- Account for motor heat from recirculation pumps (when running)

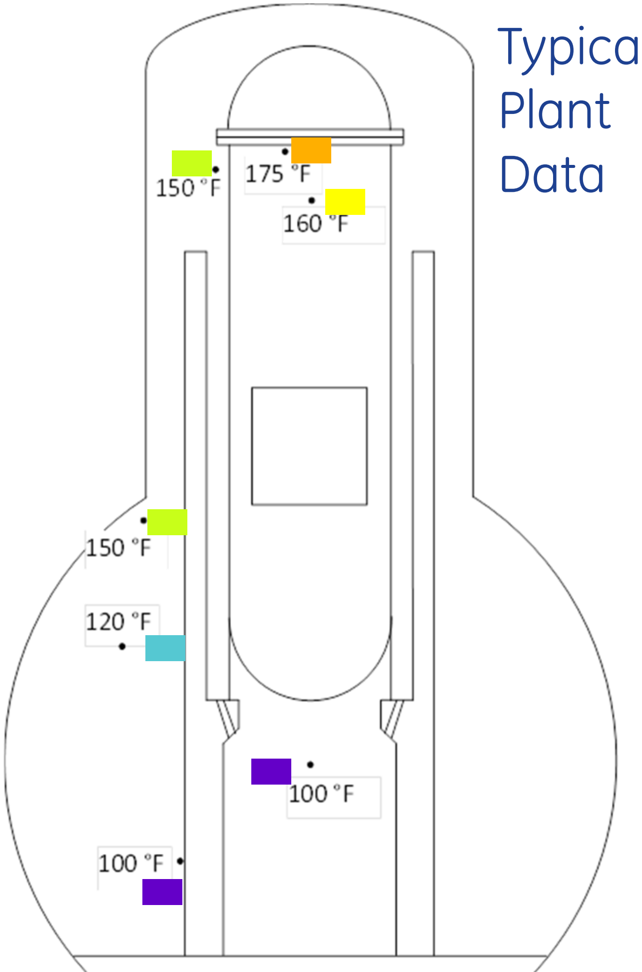
Modeling detail is necessary so that DW heat loads are produced at the correct location in the correct amounts. Steady state results are insensitive because of operating coolers; but, transient temperature distributions are impacted.

Initial DW Temperature Distribution

TRACG



Typical Plant Data

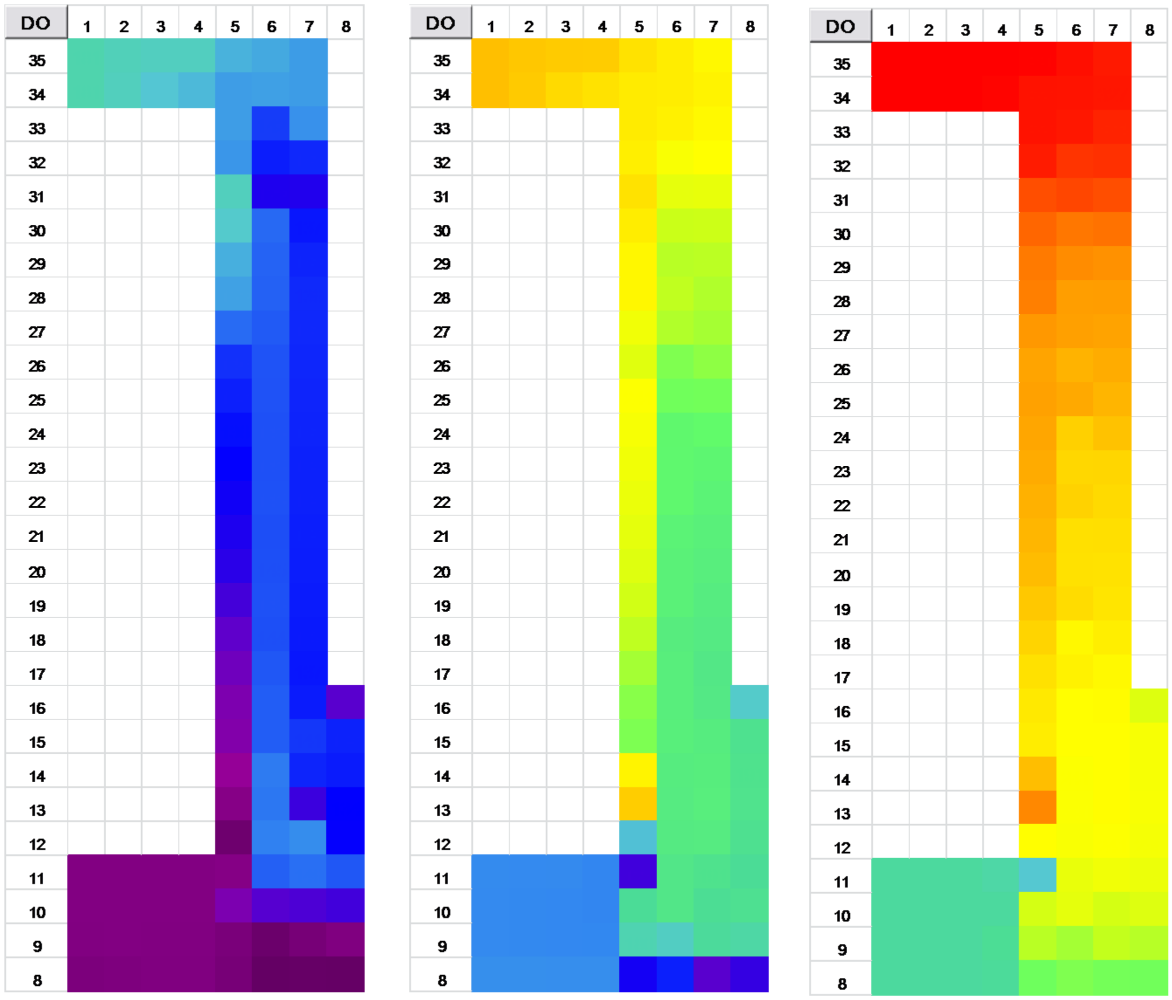


Coolers at prescribed locations are running at prescribed Flows.

TRACG calculated values reasonably match typical Mark I plant data.

DW Temperature Distributions

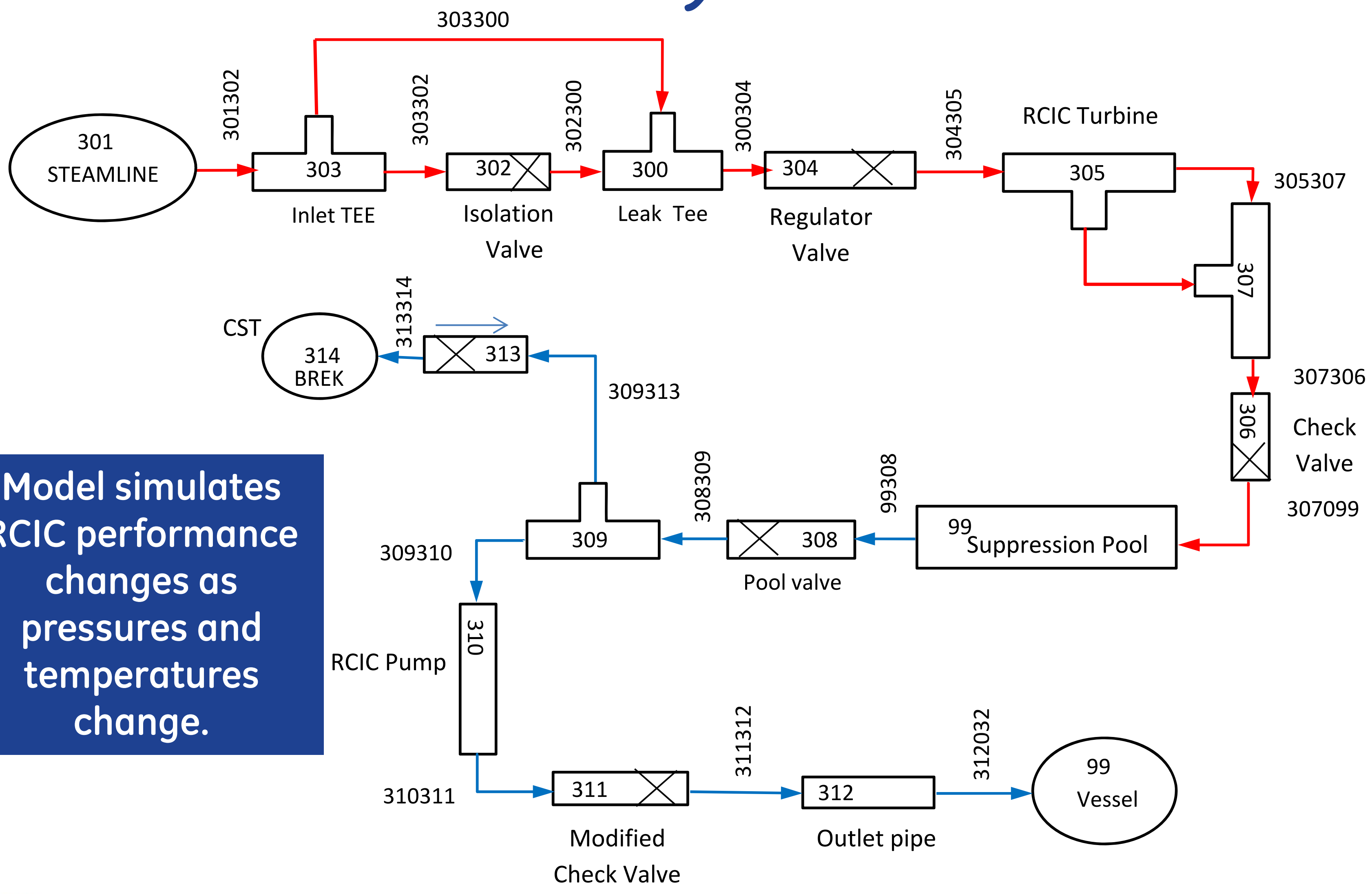
Initial 1 hour 10 hours ← Time after scram and SBO



Temperature (°F)	
🟥	341
🟧	323
🟧	305
🟧	287
🟨	269
🟨	251
🟩	233
🟩	215
🟩	197
🟦	179
🟦	161
🟦	143
🟪	125
🟪	107
🟪	89

Coolers are not powered during SBO. If they were, containment pressure and temperature increases would be much less.

Detailed RCIC Subsystem



Model simulates RCIC performance changes as pressures and temperatures change.

Completed Analyses Scenarios

48-hour baseline calculation with optimum RCIC performance

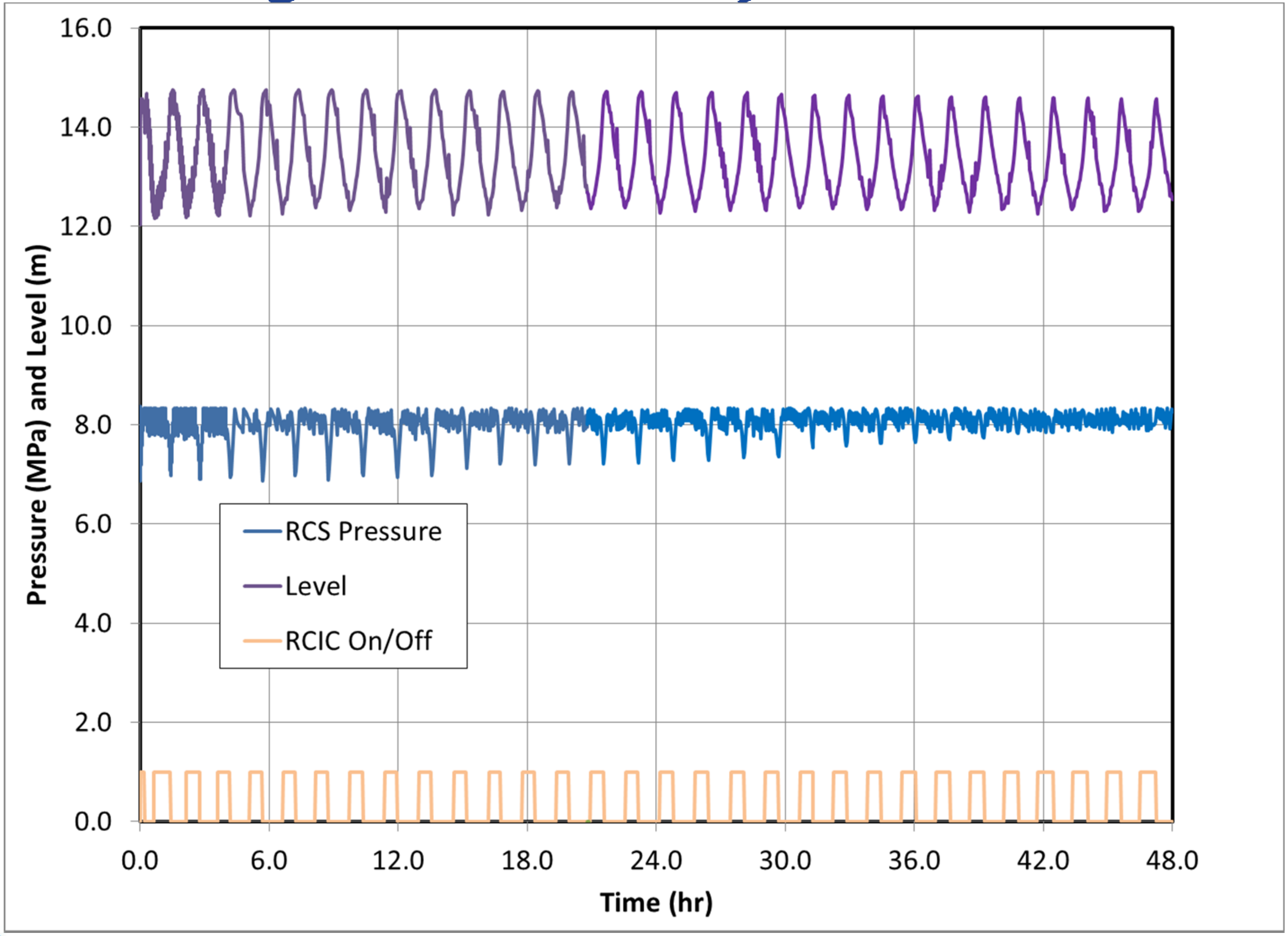
- RCIC constant speed mode, cycling on/off to control RPV level
- Fluid energy increasing, containment heating up and pressurizing

1F3-like scenario (see separate slide)

- Similar to scenario analyzed by Sandia using MELCOR
- RCIC operating same as baseline calculation for first ~21 hours
- RCIC stopped and HPCI started at 21.8 hours resulting in RPV depressurization (implies HPCI pump flow was being recirculated)
- HPCI stopped at 35.917 hours, RPV pressure increases
- At 42.350 hours the RPV is depressurized through SRVs
- Calculation continued without fluid makeup until core is destroyed

RCIC 48-hour Performance (constant speed)

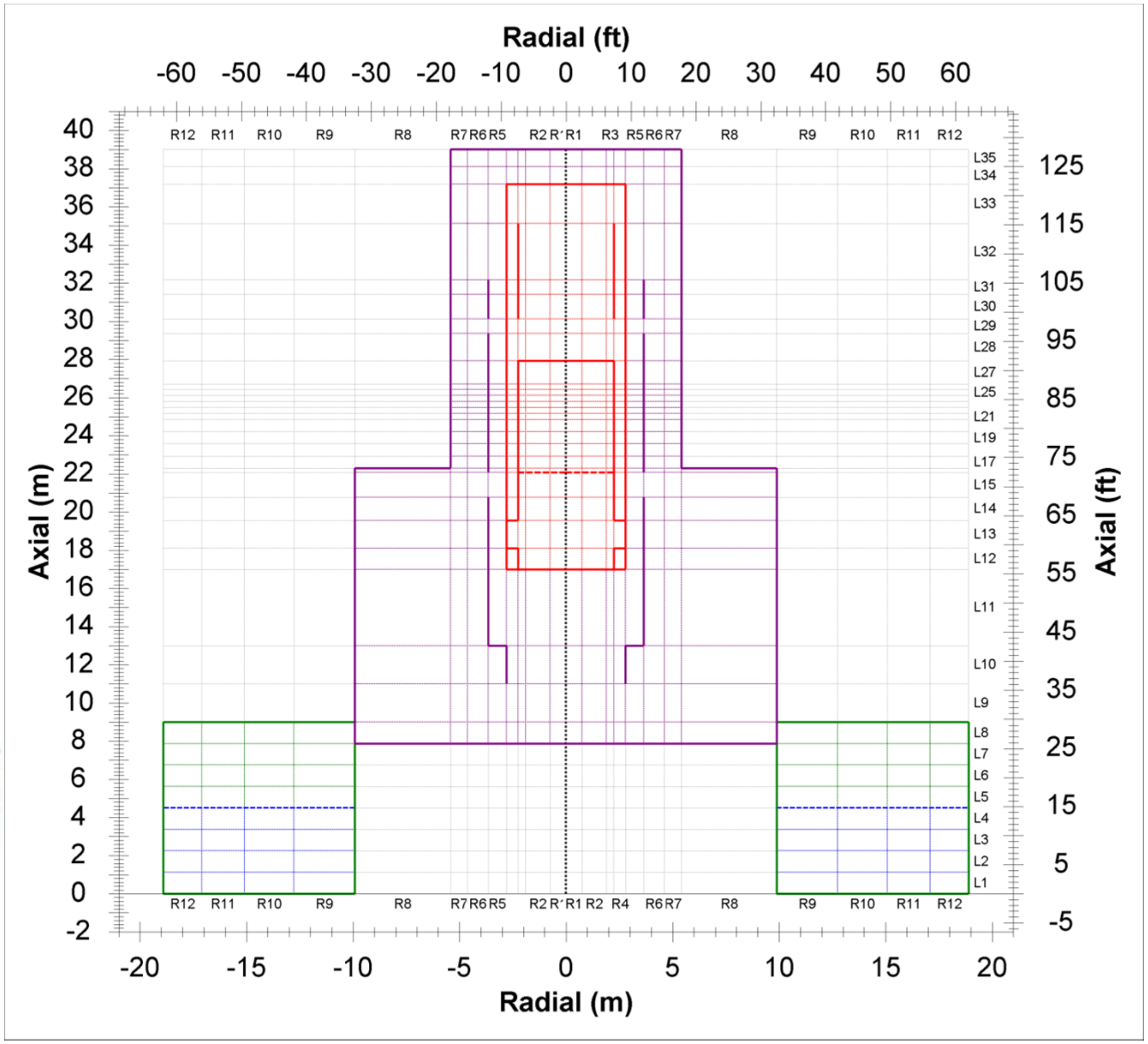
Maintaining level as decay heat decreases.



1F3 Sequence of Events used for 1F3-like Analysis Scenario

Date Hour:Minute	Time after Scram Hours	Description
3/11/2011 14:47	0.000	Reactor scram
3/11/2011 15:05	0.300	RCIC starts
3/11/2011 15:25	0.633	RCIC stops on level 8
3/11/2011 15:38	0.850	Tsunami, EDGA tripped
3/11/2011 16:03	1.267	RCIC started manually
3/12/2011 4:20	13.550	RCIC suction swapped to torus
3/12/2011 11:36	20.817	RCIC shuts down
3/12/2011 12:06	21.317	Torus spray started (fire pump)
3/12/2011 12:35	21.800	HPCI started on Level 2
3/13/2011 2:42	35.917	HPCI was secured, switched Fire Pump injection
3/13/2011 5:08	38.350	Torus spray started
3/13/2011 5:10	38.383	Loss of all injection, water level starts to fall
3/13/2011 7:39	40.867	Drywell spray started
3/13/2011 8:41	41.900	Containment vent completed
3/13/2011 9:08	42.350	RPV depressurization using SRVs (relief valve function)
3/13/2011 9:25	42.633	Fresh water injection
3/13/2011 13:12	46.417	Sea water injection
3/14/2011 11:01	68.233	Hydrogen explosion. Fire pump and hoses were damaged
3/14/2011 15:30	72.717	Sea water injection resumes with new pump and hoses.

Videos of Mark I Model Performance



Video: Baseline 48-hr RCIC operation

Video: 1F3-like scenario with HPCI at 21.8 hours

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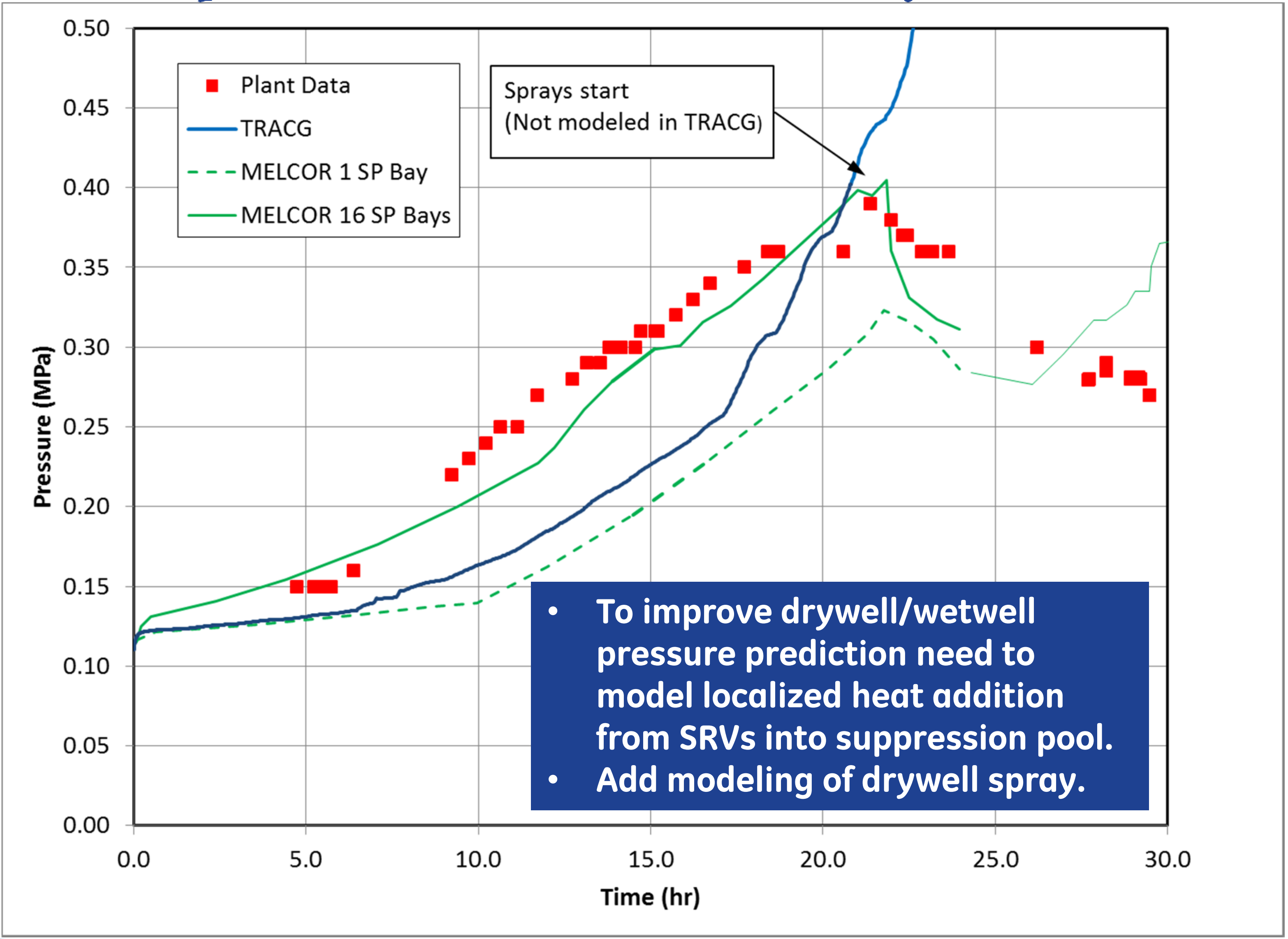
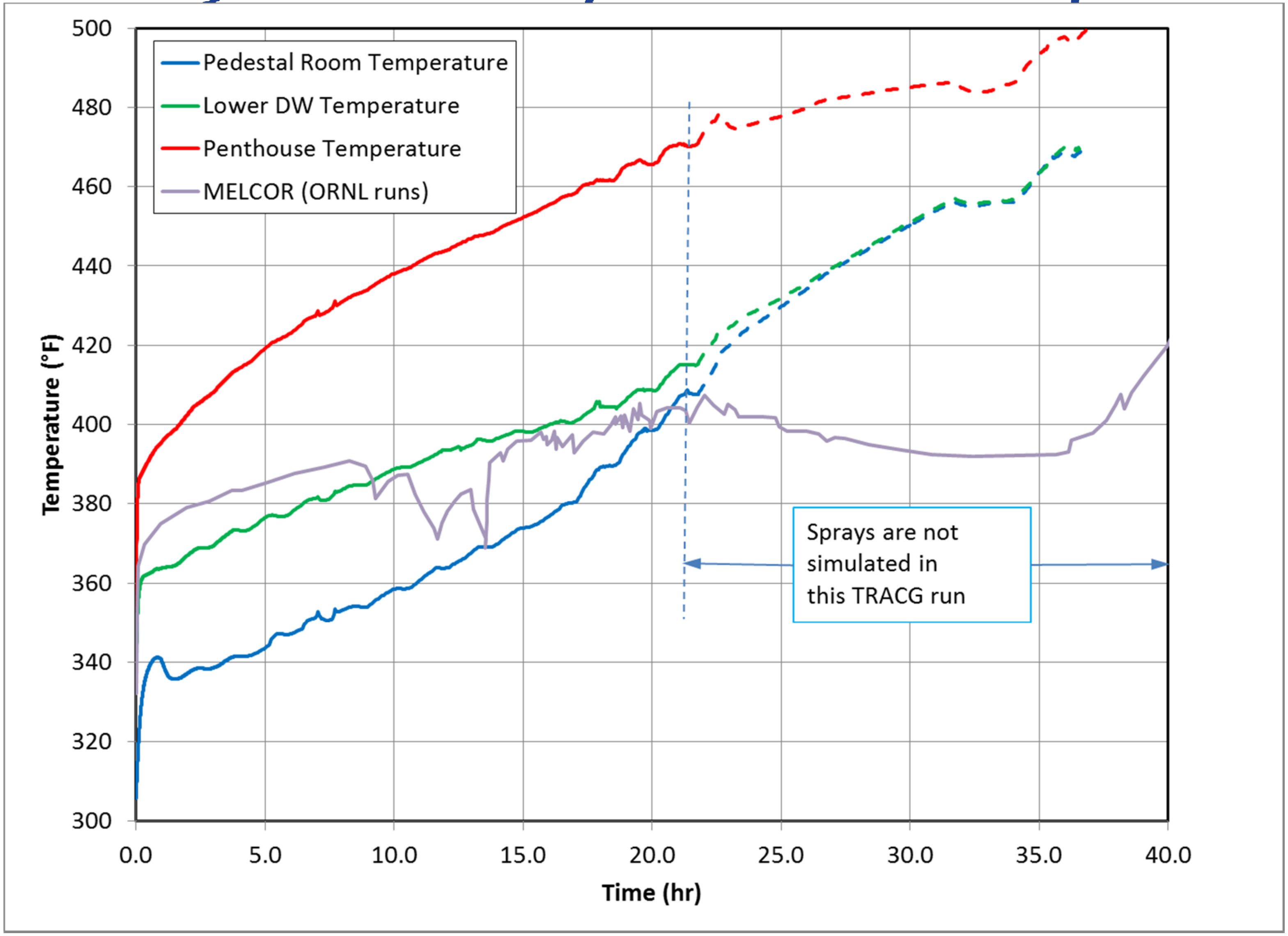
1F3 Drywell Pressure Response



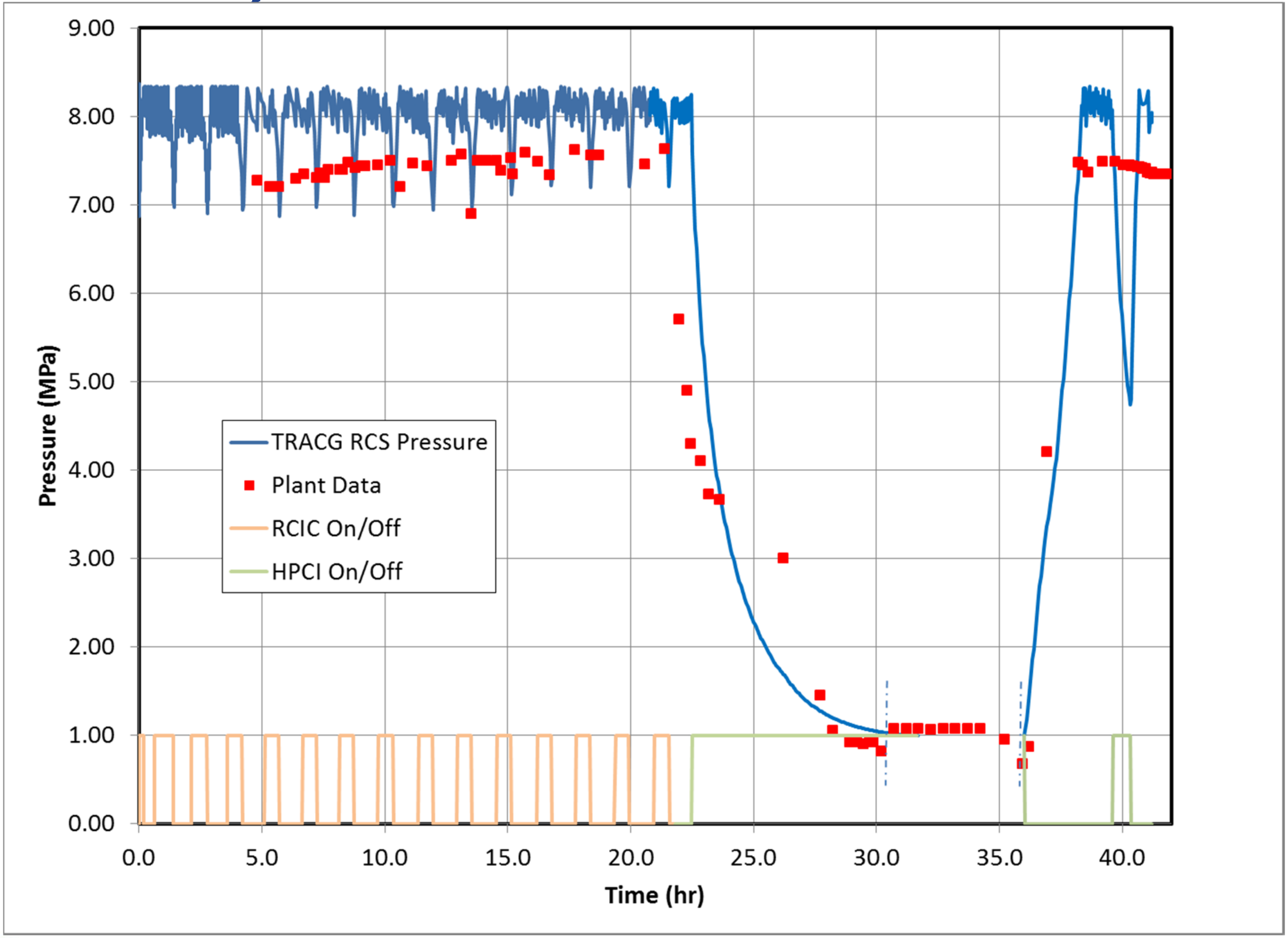
- To improve drywell/wetwell pressure prediction need to model localized heat addition from SRVs into suppression pool.
- Add modeling of drywell spray.

1F3 Drywell Temperature Response



Sprays are not simulated in this TRACG run

1F3 Responses to HPCI at 21.8 hours



Concluding Remarks and Path Forward

- All required detailed design elements have been integrated and successfully demonstrated to produce good agreement with data.
  - All technical development risks have been mitigated.
  - Beginning to look at requested BWROG and utility scenarios.
- Complete integration work for Mark II to same level as for Mark I.
  - Integrate: detailed RCIC, fan coolers, pipe heat losses, insulation
- Complete **Detailed Design Review**
  - Documentation / verification of detailed design elements and models nearly complete.
- Execute the qualification plan for **Design Validation Review**.
- Obtain transient temperature data for Mark I and/or Mark II essential for the **Design Validation Review**.
- Complete parallel testing of TRACG code mods to support Level 2 production Engineering Computer Code (ECP).

References

[1] Fukushima Daiichi Accident Study (Status as of April 2012), SAND2012-6173, Unlimited Release, printed July 2012.

[2] Institute of Nuclear Power Operators (INPO), —Special Report on the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station,‖ Revision 0, INPO 11-005, November 2011. http://www.nei.org/filefolder/11_005_Special_Report_on_Fukushima_Daiichi_MASTER_11_08_11_1.pdf

ACRS Materials, Metallurgy and Reactor Fuels Subcommittee

GEH Fuel Performance Update

ECCS EM/LOCA Topics

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November 21, 2013

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Outline

- SAFER and SAFER/CORCL EM Changes/Errors
  - ΔPCT impact estimates and *50.46* reporting for 2013
- Realistic LOCA EM development
  - Status of TRACG LOCA RAIs
  - Technical advances: reduction of noise-driven uncertainty
- 10CFR50.46 Rulemaking-related activities
  - Update on 50.46c
  - Recent evaluation of fuel dispersal potential during LOCA

SAFER EM Changes/Errors

SAFER EM Changes/Errors

- In June, new SAFER production version addresses current software problem corrections and improvements.
- Most of the changes either have no impact or insignificant (ΔPCT = ~0°F) impact on LOCA results except for one where a minimum core ΔP at SAFER to CORCL interface causes a continuous steam flow and unjustified cooling for some conditions for external pump plant analyses when the core spray is delayed.
- Evaluations with the corrected code and associated *50.46* reports to the licensees are underway. Target is *1/31/14* for completing impact reports to licensees.
- No 30-day reporting is expected from the recent changes.

GEH/GNF follows the reporting process that meets 10CFR50.46 requirements.

TRACG LOCA Licensing Review

TRACG LOCA Licensing Review

- First set of RAIs, having 66 questions, are formally received on 10/22/12.
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TRACG LOCA Application Methodology licensing review continues – GEH is working on formal RAI responses.

Elimination/Reduction of Excess Unc’y.

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Comparisons of Core Configurations

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Reduction of Noise

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Comparisons of Statistical Results for BWR/2 Discharge DEGB

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Rulemaking Activities

Rulemaking Activities

- BWROG-TP-11-010 Rev. 1 (June 2011) provided an evaluation of BWR LOCA analyses and margins against high burnup fuel research findings. Substantial margin for jet pump design was demonstrated.
- Compliance with the new rule without a revised analysis (and without a new methodology) can be demonstrated for majority of U.S. BWRs.
- A '*fast track*' approach was proposed by GEH and the proposal gained acceptance/support from the industry, as well as the staff. Next step is to 'codify' the details.
- AREVA and Westinghouse also support the approach and will participate. Efforts to develop a common communication to NRC continues. It will be available by the end of year, on consistent timing with the rulemaking schedule.

Background

- In RIL-0801 (“Technical Basis for Revision of Embrittlement Criteria in 10 CFR 50.46”, May 2008):
  The current NRC burnup limit of 62 MWD/kgU for peak rod average was considered to be low enough to prevent significant fuel loss during LOCA.
- In NUREG-2121 (“Fuel Fragmentation, Relocation, and Dispersal During the Loss-of-Coolant Accident”, March 2012):
  NRC found that the previous conclusion related to axial fuel relocation remains founded, meaning that the conservatism in Appendix K methods is adequate and no immediate regulatory action is necessary.

The recent study evaluating the fuel dispersal potential complements the NUREG-2121 conclusions.

Fragmentation/Dispersal Evaluation

- OECD Halden Reactor Project, a series of LOCA tests have been conducted to examine fuel performance under high burnup conditions. Some of these tests showed that the highly fragmented fuel can be dispersed from the burst opening under certain conditions.

NEA/CSNI/R(2010)5

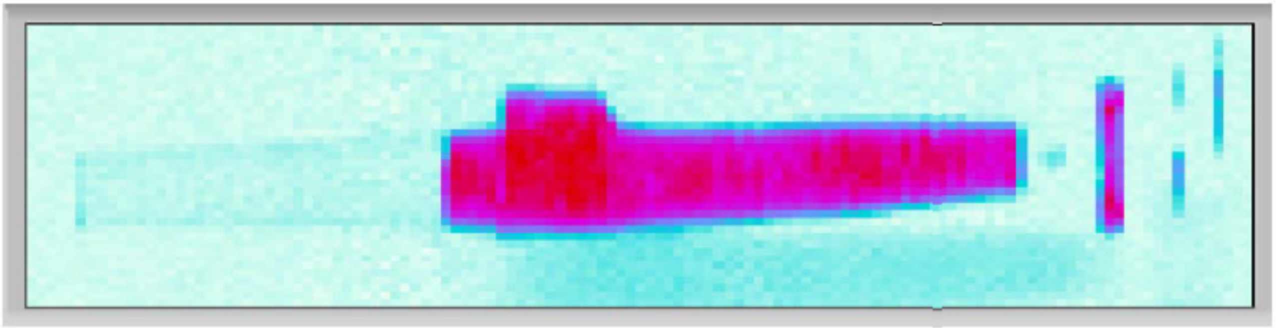


Figure 4 – Gamma scanning of LOCA fuel rod from IFA-650.4. Fuel is missing at the top (left), ballooning at half height. Some fuel has fallen to the bottom of the flask (right).

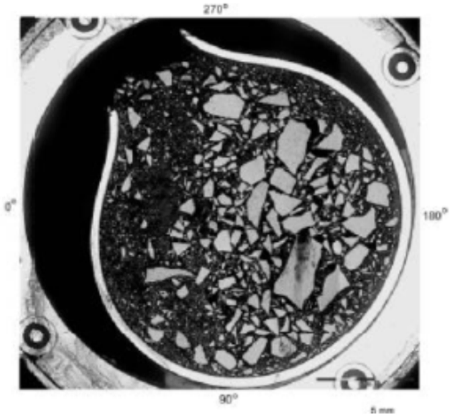
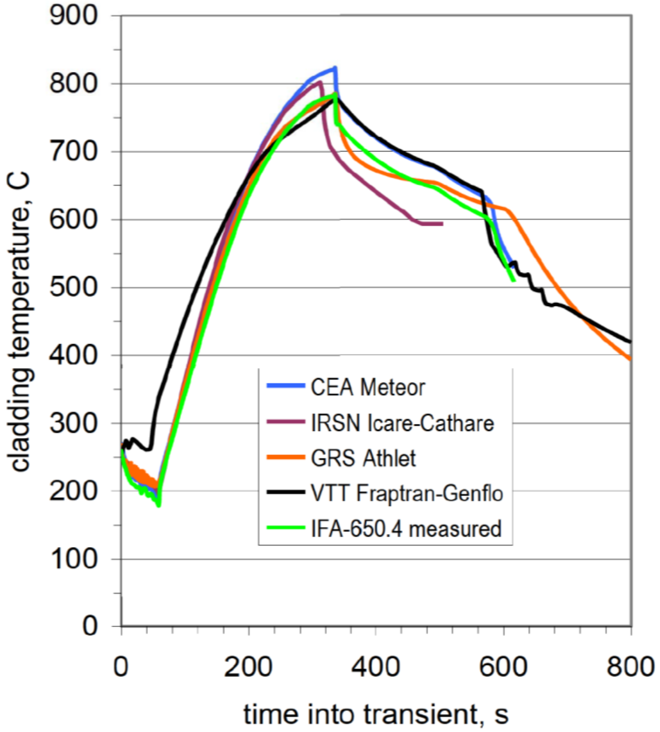


Figure 5 – Cross-section showing fuel relocation. Filling ratio 38%

Dispersal - Background

Test Number IFA.650.X	Fuel Type	Cladding	Burnup (MWD/kgU)	Hydrogen (ppm)	Oxide Thickness (μm)	Pressure (bar)	Target PCT (°C)
1,2	Commissioning tests with fresh fuel.						
3	PWR	Zry-4	82	250	18-27	40	800
4	PWR	Zry-4	92	50	10	40	800
5	PWR	Zry-4	83	650	70-80	40	1100
6	VVER	E110	56	100	5	30	850
7	BWR	LK3/L	44	44	10	6	1100
8	System check-out test with fresh fuel.						
9	PWR	Zry-4	90	30	7-8	40	1100
10	PWR	Zry-4	60	40	20-30	40	850
11	VVER	E110	56	100	5	30	1000
12	BWR	LK3/L	72	300	40	20	850

Experiment versus BWR



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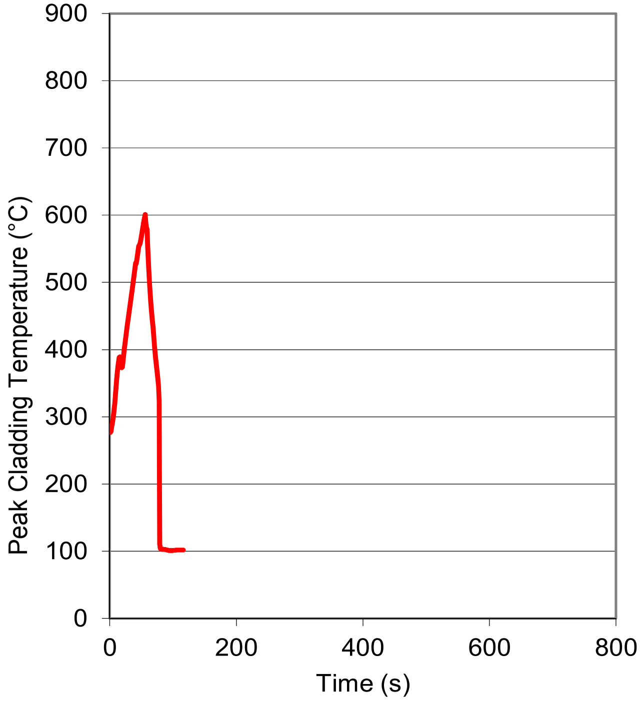

App. K calc. at T/M limit

The temperature profile used in the experiments is not only unrealistic but also more conservative than App. K-based analytical predictions. Therefore, the tests don’t represent conditions applicable to plant calculations.

Evaluation Approach [[

- Evaluation performed using SAFER/CORCL (an approved ECCS EM).
- “Appendix K” input assumptions used for conservative assessment.
- Thermal-mechanical (T-M) design limits used. EM rod stress and perforation model does not underestimate the incidence of rupture based on applicable data including those data reported in NUREG-0630.
- When bundle power from core design/operation is ‘LOCA-limited’, ECCS analysis limits the peak LHGR to the maximum value that would yield the specific target PCT and ECR oxidation values below the acceptance criteria.

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Results

	Nodal Burnup (MWd/kgU)	0	1.10	5.51	11.02	16.53	22.05	27.56	38.58	49.60	60.63	71.65	82.67
AOR	PCT (°C)	1198	1197	1195	1197	1194	1188	1189	1170	1130	1092	1016	918
	# of burst rods	92	92	92	92	92	92	92	92	64	92	56	56
95%	PCT (°C)	1114	1094	1120	1110	1108	1115	1100	1079	1057	1033	984	876
	# of burst rods	92	64	56	56	56	56	56	56	56	56	56	8
90%	PCT (°C)	1059	1047	1036	1033	1042	1038	1038	1023	1005	984	938	840
	# of burst rods	56	56	56	56	56	56	56	56	56	56	56	2
85%	PCT (°C)	984	984	979	979	993	976	985	966	958	941	887	792
	# of burst rods	56	56	56	56	56	56	56	56	56	56	36	-
80%	PCT (°C)	938	932	934	933	944	939	935	932	913	888	851	755
	# of burst rods	42	42	42	28	43	28	28	28	18	8	8	-
75%	PCT (°C)	899	897	892	897	891	883	881	882	856	837	799	680
	# of burst rods	10	10	-	-	-	-	-	-	-	-	-	-
70%	PCT (°C)	872	851	849	850	844	840	838	811	798	778	739	666
	# of burst rods	-	-	-	-	-	-	-	-	-	-	-	-
65%	PCT (°C)	808	797	798	790	794	789	770	765	747	732	687	630
	# of burst rods	-	-	-	-	-	-	-	-	-	-	-	-
60%	PCT (°C)	759	751	746	729	725	723	725	719	683	686	657	602
	# of burst rods	-	-	-	-	-	-	-	-	-	-	-	-

The particular study presented here was performed to address the Swiss Regulator ENSI's questions.

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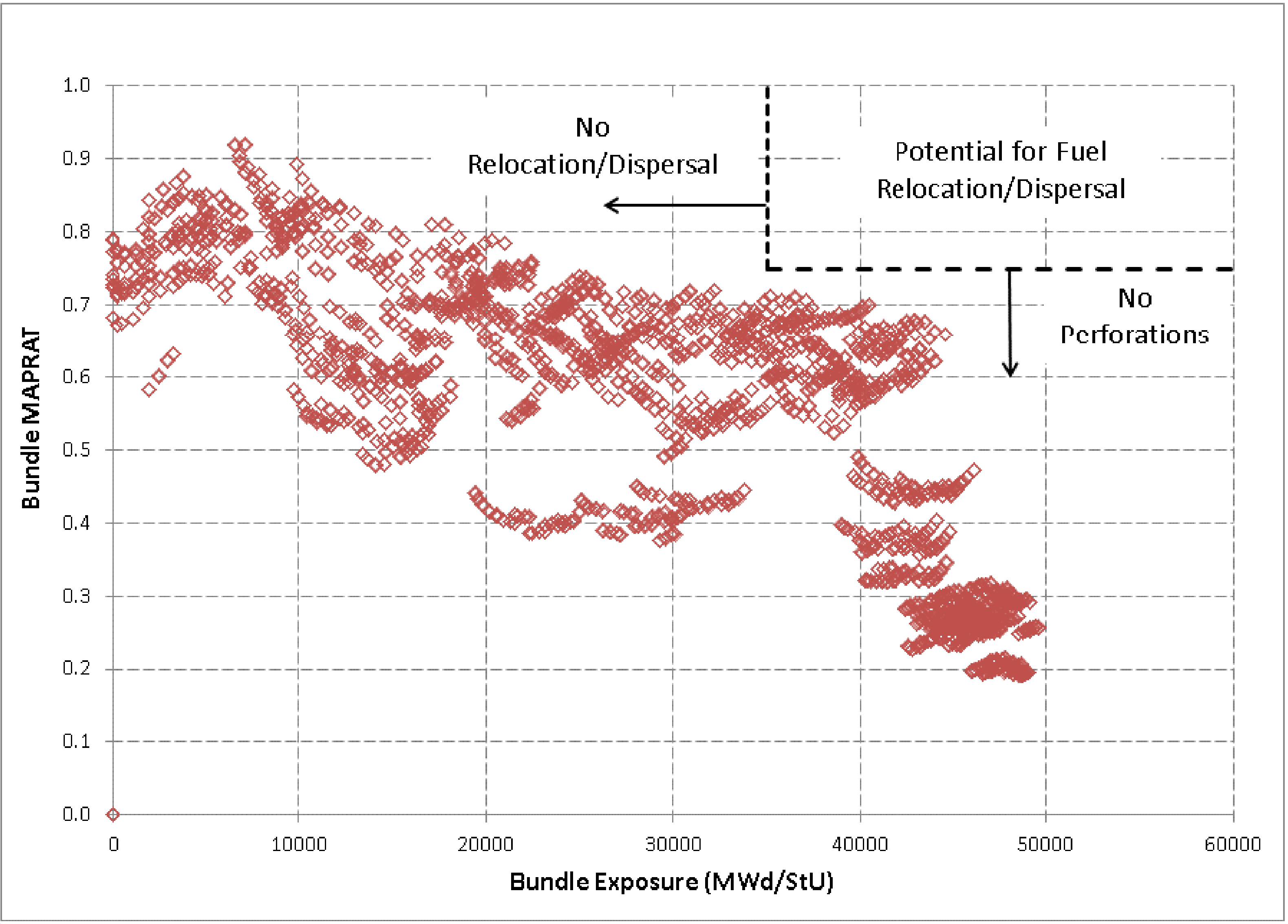
Results



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Conclusions

- The conditions at which the high burnup fuel specimens are tested for fragmentation and dispersal purposes are beyond the expected conditions during LOCA in a commercial BWR.
- T-M limits that are imposed on the fuel and the core loading and exposure plans prevent high burnup fuel from operating at elevated linear heat generation rates.
- High-burnup fuel does not attain temperatures above 700°C and do not rupture during LOCA.
- For the highest exposures, including the peak pellet exposure extension up to 80 MWD/kgU, there are no concerns for fuel dispersal since, when operating at the T/M limit, the rods would not have the heat to reach burst conditions.

No safety concern arising from the tests.

[redacted]