Canadian Nuclear Safety Commission
Commission canadienne de sûreté nucléaire

CMD 15-H8.33B

File / dossier : 6.01.07
Date: 2015-10-19
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Supplementary Information

Oral presentation

Revised presentation from Sunil Nijhawan

In the Matter of

Ontario Power Generation Inc.

Application to renew the Power Reactor Operating licence for the Darlington Nuclear Generating Station

Commission Public Hearing
Part 2

November 2-5, 2015

Renseignements supplémentaires

Exposé oral

Présentation révisée de Sunil Nijhawan

À l’égard de

Ontario Power Generation Inc.

Demande concernant le renouvellement du permis d’exploitation pour la centrale nucléaire de Darlington

Audience publique de la Commission
Partie 2

2-5 novembre 2015

DARLINGTON SEVERE ACCIDENT ISSUES

Sunil Nijhawan, Ph.D. P.Eng

Prepared 19 October 2015

A 13 year licence does not serve public interest

1. Periodically we must ask if the aging plant still meet the original licensing basis using the acceptance criteria employed by regulators last time the plant was licensed
2. Periodically we must analyze if new information changed the understanding of previously employed acceptance criteria within the original licensing basis
3. Periodically we must assess if compliance with original licensing basis mean that risk from the original licensing basis is acceptable today
4. Periodically public must inquire there been any private relaxation of original licensing basis along the way (e.g. reactor building not pressure tested last time it was due)
5. Periodically an independent, off-shore review of the licensing basis and its compliance would be undertaken
6. Periodically public can independently inquire if reactor licensable at that time in Canada and in other jurisdictions
7. Periodically public can reassess its evolving expectations of risk.
8. Plant operator can be periodically required to meet evolutionary and new / different public expectations
9. Periodically we can ask if risk from accidents previously not considered in licensing basis be evaluated and has it been properly evaluated and acceptable today
10. Regulatory regime may be seen as not independent, impartial, competent, effective or relevant; Current regulatory regime may be obsolete and irrelevant well before 13 years.

A 40 year old plant design not meant forever

- Reactor design obsolete and does not meet current design practices.
- Current refurbishment plans do not include necessary design upgrades.
- Risk assessment for various combinations of units under refurbishment not undertaken

Promised environmental assessment not undertaken honestly

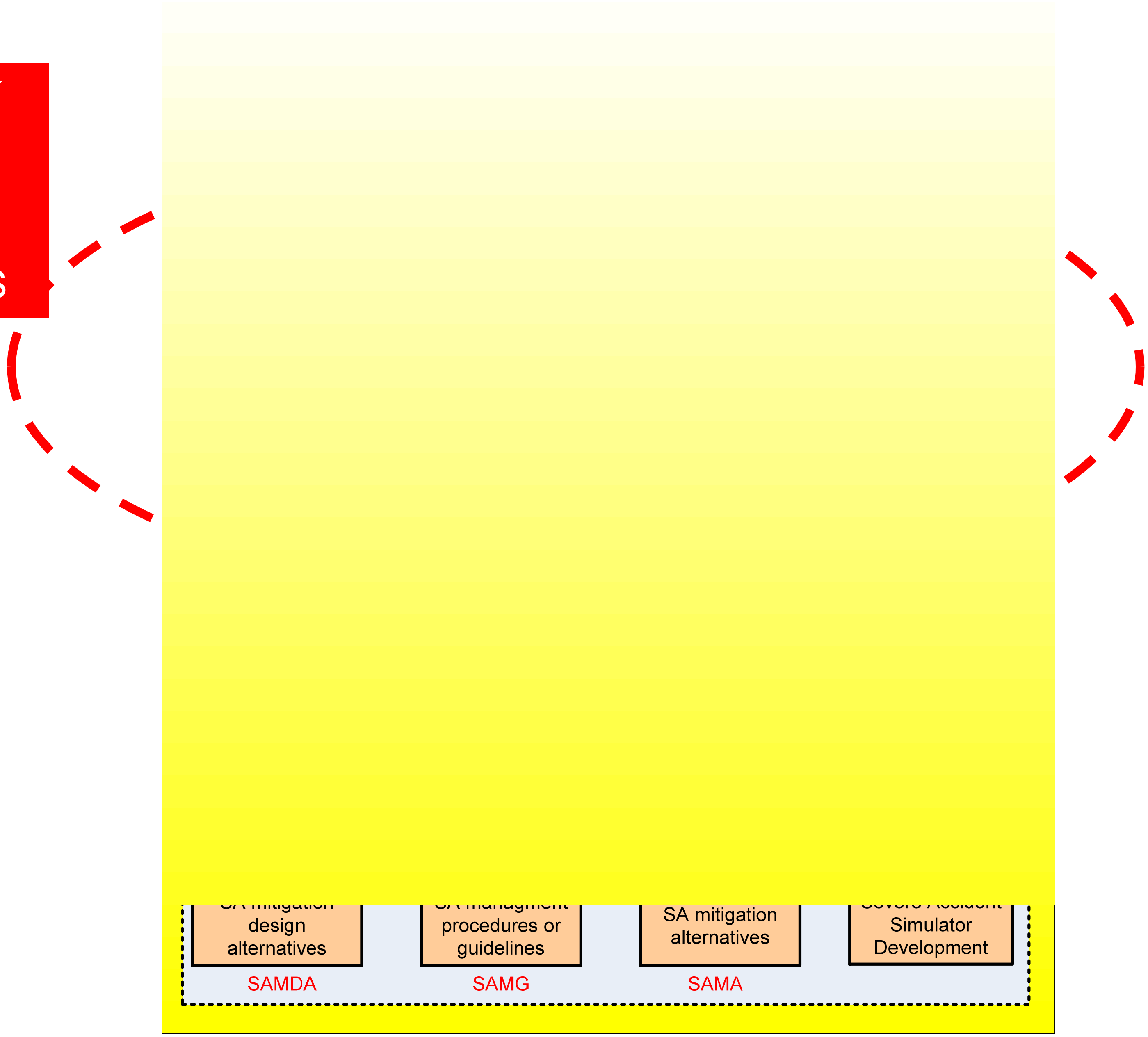
- CNSC study on consequences of a severe accident does not serve the intent and ill serves public interest

- *CNSC staff, because of public concern, agreed to provide an information document or equivalent assessing health and environmental consequences of more severe accident scenarios discussed by intervenors and intends on updating the Commission on this topic in fall 2013*

- *August 2015 CNSC study on consequences of a severe accident is irresponsible in its 'estimation' of source term and a mockery of the public trust. If it was a private corporation and fully accountable for its actions it would suffer greater humiliation than Volkswagen once facts are publically discussed.*

- *Darlington relicensing must be made subject to a comprehensive and acceptable risk assessment and environmental assessment review.*

POORLY DONE FOR MOST CANDUS



CMD 15-M10/15-M10.A
Transcript Public meeting – 26 march 2015

Quoting a staffer named Andrew McAllister - acting director of the Environmental Risk Assessment Division.

A "source term" is defined as the types and amounts of radioactive material released to the environment following an accident.

For this study, it was based on the magnitude of CNSC's large release safety goal of 1 X 10E14 becquerels of cesium-137, was comparable in magnitude to the 10E-7 type of severe accident scenario discussed by interveners during the Darlington refurbishment environmental assessment and was 4-5 orders of magnitude greater than the actual accident assessed as part of the aforementioned environmental assessment. The source term examined in this study is significantly larger than would be expected under any credible scenario.

SAFE NUCLEAR

FACTS ABOUT A DARLINGTON SEVERE ACCIDENT PROGRESSION

1. A SEVERE CORE DAMAGE ACCIDENT IS A PLAUSIBLE EVENT
2. FUEL GETS HOT IN A SEVERE ACCIDENT
3. RELAESE RATES OF FISSION PRODUCTS ARE HIGH AT HIGH TEMPERATURES
4. RELEASE RATE OF CS-137 COULD BE 1% PER MINUTE AT 1600^{o}C SO A LARGE FRACTION WILL RELEASE FROM FUEL AND DEBRIS INTO CALANDRIA IN A COUPLE OF HOURS
5. CALANDRIA RELEASES END UP IN THE CONTAINMENT IMMEDIATELY AND UNATTENEUATED
6. CONTAINMENT IS CLOSE TO ATMOSHERIC AT ONSET OF CORE DAMAGE AND WILL PRESSURIZE DUE TO FURTHER ENERGY RELEASE INTO IT
7. CONTAINMENT IS LEAKY; HAS AN UPTO 48% MASS PER DAY LEAKAGE RATE AT DESIGN PRESSURE.
8. CONTAINMENT WILL PRESSURIZE TO GREATER THAN DESIGN PRESSURE EASILY (0.5 Atm for VACUUM BUILDING; 0.9 Atm for RB)
9. DARLINGTON REACTOR VESSELS ATTACHED TO THE CONTAINMENT PRESSURE BOUNDARY

FACTS ABOUT A DARLINGTON SEVERE ACCIDENT PROGRESSION

11. MORE THAN LIKELY A LARGE FRACTION OF FISSION PRODUCTS WILL RELEASE FROM CONTAINMENT TO ENVIRONMENT
12. ‘HYDROGEN’ PRODUCTION FROM FEEDERS ADDS TO ‘HYDROGEN’ FROM FUEL CHANNEL AND IS HIGHER THAN PREVIOUSLY ANTICIPATED
13. COMBUSTIBLE ‘HYDROGEN’ WILL BE LIKELY TRAPPED IN REACTOR BUILDINGS
14. ASSERTION OF SMALLER RELEASES THAN 0.1% OVER THE TOTAL ACCIDENT DURATION IS UNFOUNDED AND IRRESPONSIBLE
15. ASSERTIONS OF 4-5 ORDERS OF MAGNITUDE LOWER RELEASE THAN REGULATORY LIMIT OF 100 TBQ ARE PATENTLY INCORRECT AND FRIGHTENING DISPLAY OF DISTORTED THINKING
16. CNSC SHOULD RECONSIDER THE EFFECT ON PUBLIC SAFETY SUCH ILL-ADVISED DECISIONS (TO ENDORSE SUCH STUDIES) CAN HAVE.
17. DARLINGTON REACTORS POSE MORE RISK THAN CLAIMED IN PSA STUDIES AND SHOULD BE UPGRADED WITH SUPPORT OF INTELLIGENT AND THOGHTFUL ANALYSES, NOT HAND WAVING

DARLINGTON NGS SEVERE ACCIDENT PROGRESSION & MITIGATION ISSUES

- Darlington reactors did not consider severe accidents in the design process. Unreasonable to expect easy severe accident mitigation.
- Severe accidents in all inter-connected units a nightmare scenario.
- Current Darlington NGS designs inherently forces a reactor damage even before an ECC loss leading to severe core damage.
- No provisions for manual depressurization after SBO. No super high pressure ECC or makeup intervention / injection.
- Onset of a severe core damage in Darlington reactors puts activity and combustible gases directly into the relatively weak containment. There is no holding of activity in a vessel like in a PWR pressure vessel.
- Significantly higher sources of hydrogen from large amounts of carbon steel and Zircaloy. Recombiners will cause explosions.
- Enhanced potential for energetic interactions with enveloping water
- Pressure relief in ALL relevant reactor systems in inadequate (PHTS, Calandria, Shield Tank, Containment)
- Darlington containment – a negative pressure concept – along with Bruce amongst the weakest in the world for pressurization; severe accidents will cause pressurization
- Containment bypass from reactivity device failure a likely outcome after a severe core damage

DARLINGTON NGS SEVERE ACCIDENT PROGRESSION & MITIGATION ISSUES

- Calandria vessel cannot contain debris and can fail catastrophically at welds.
- Current Shield Tank cannot contain pressure upon boiling and can fail. Restoration of cooling after water depletion problematic as flow outlet at the top of vessel.
- Inadequate instrumentation and control after a severe accident.
- Poor equipment survivability
- Currently planned PARS inadequate and potentially dangerous.
- No dedicated operator training / simulators for severe accidents.
- Severe accident simulation methods are outdated, crude and inadequate.
- No significant design changes implemented. Known problems ignored.
- Current SAMGs are inadequate. Many Emergency hookups not implemented
- High risk potential from external events
- Need to reconsider malevolent actions and sabotage.

MAJOR CONCLUSIONS

1. Steam generators remain an effective heat sink for 5 hours
2. Steam generator emergency cooling system can add another 8-10 hours of cooling.
3. A core collapse cools the core for extended period of time.
4. A minor containment breech (1 m^2) late in the game.
5. Only 0.2% of fission products are released into the atmosphere in 24 hours. Nothing released to atmoshere between 7 and 25 hours.
6. A totally unbelievable scenario. It just cannot happen. Operators will handle it.

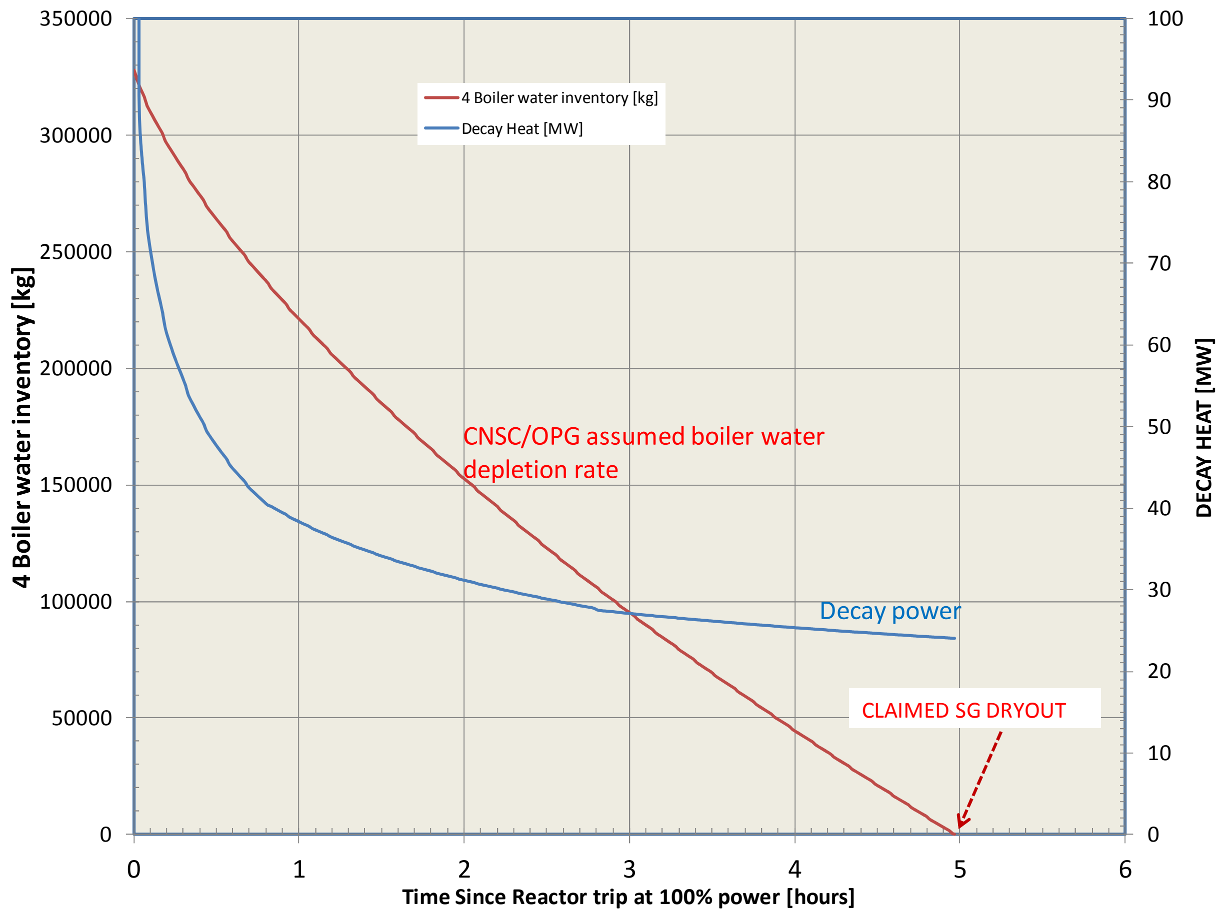
SOME OF THE KNOWN DEFICIENCIES IN MAAP-CANDU

- No consideration of heavy water, deuterium gas
- No momentum equation for HTS – no fluid flows
- Channel degradation before dry steam/D_2 heatup not modelled - Initial fuel temperatures at onset of heatup are arbitrary
- Channel hydraulics based on assumed header to header Δp and no overall T/H. No intra channel flows. No consideration of fluid discharge paths.
- A limited number of channels modelled. No explicit sheath modelling. No modelling of out of flux pressure tube lengths. No modelling of water retention in end fittings.
- No thermal modelling of feeders and end fittings

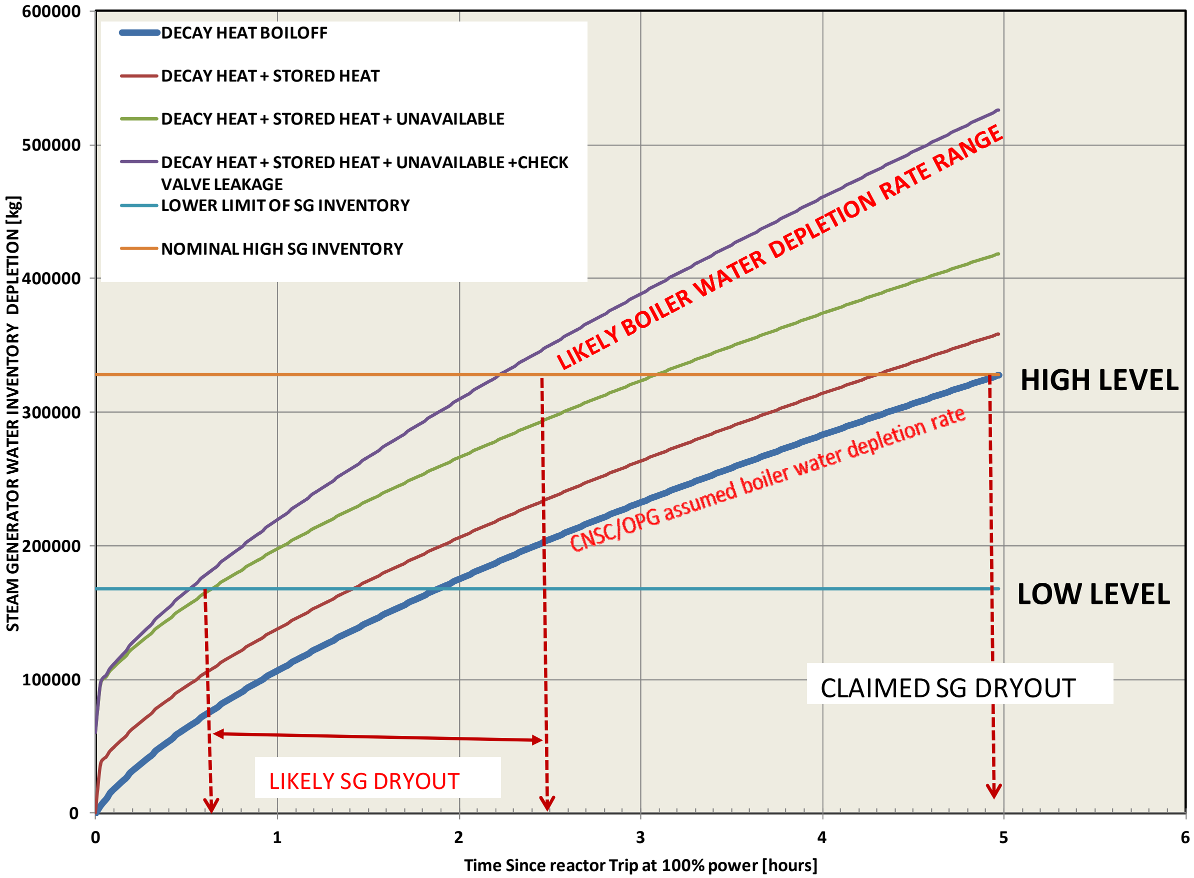
MORE KNOWN DEFICIENCIES IN MAAP-CANDU

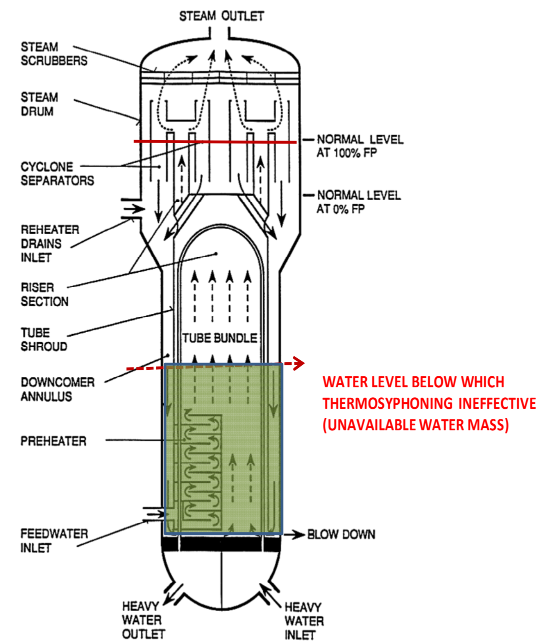
- No consideration of differences in burnup and power profiles between various channels
- No modelling of in-core devices.
- No modelling of piping into calandria vessel.
- Crude modelling of disassembly & solid debris
- Solid debris interactions with air not modelled
- Deuterium / Hydrogen generation by steel oxidation and Uranium-steam oxidation ignored.
- Fission product releases from debris crudely modelled.
- Fission products do not decay.
- As 'engineered' codes with specific accident progression pathways – many scenario development paths not considered.
- Difficult I/O; primitive post processing

REVIEW OF CNSC REPORT ON UNMITIGATED SBO



A MORE REALISTIC PREDICTION OF BOILER WORTH





SURRY PWR BOILER WORTH

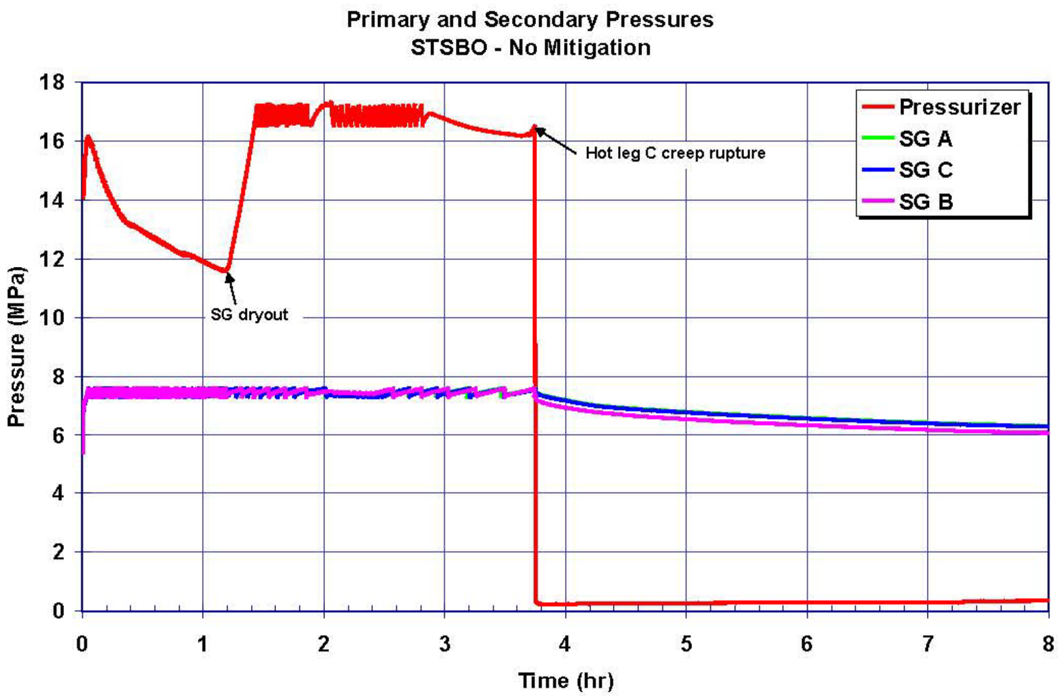
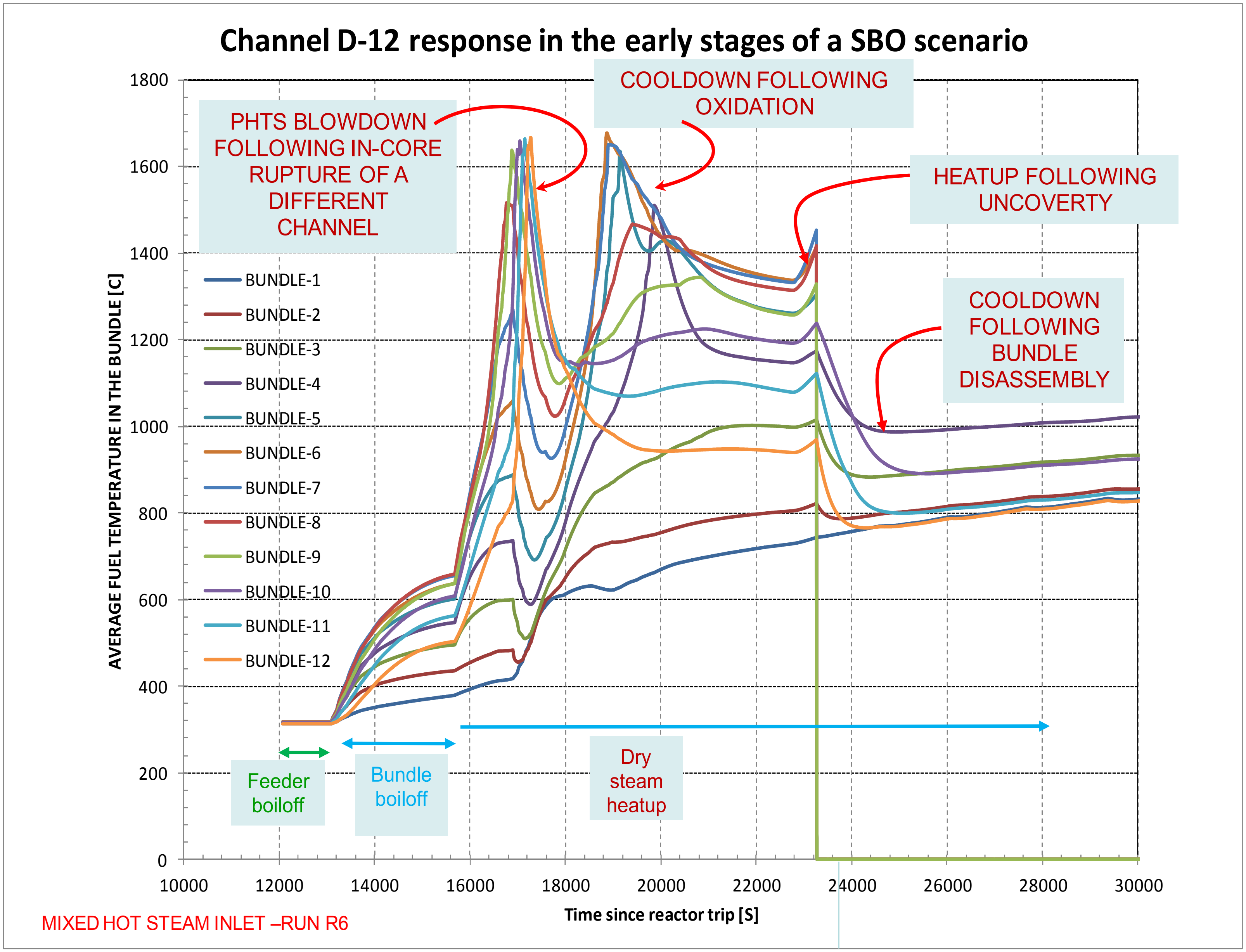
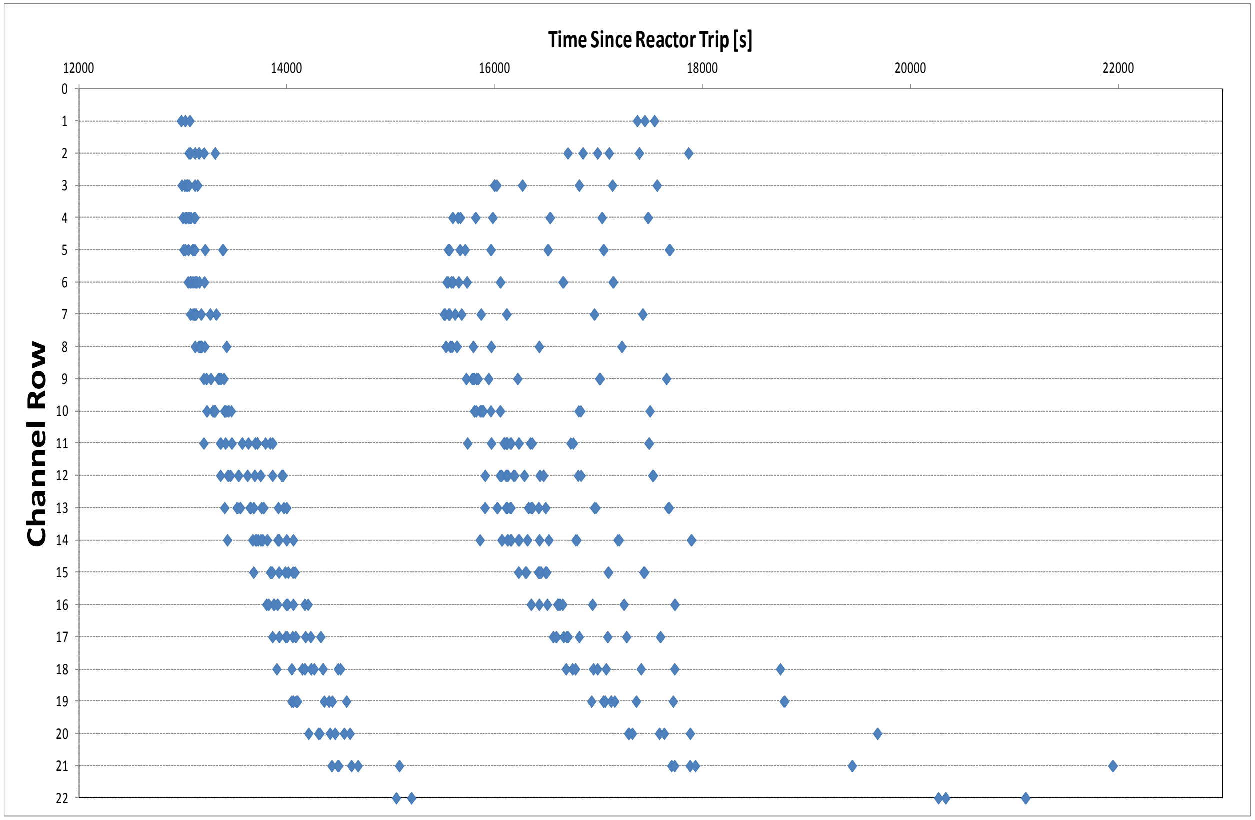


Figure 5-28 Unmitigated STSBO primary and secondary pressures history

FUEL CHANNEL HEATUP TRANSIENT



STAGGER IN CHANNEL HEATUP



VARIABILITY IN CHANNEL POWER NECESSITATES DETAILED ANALYSES FOR RELIABLE SOURCE TERMS

FEEDER SIZES THAT DEPEND UPON POWER HAVE A LARGE IMPACT ON TIMING OF FUEL HEATUP IN INDIVIDUAL CHANNELS

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24
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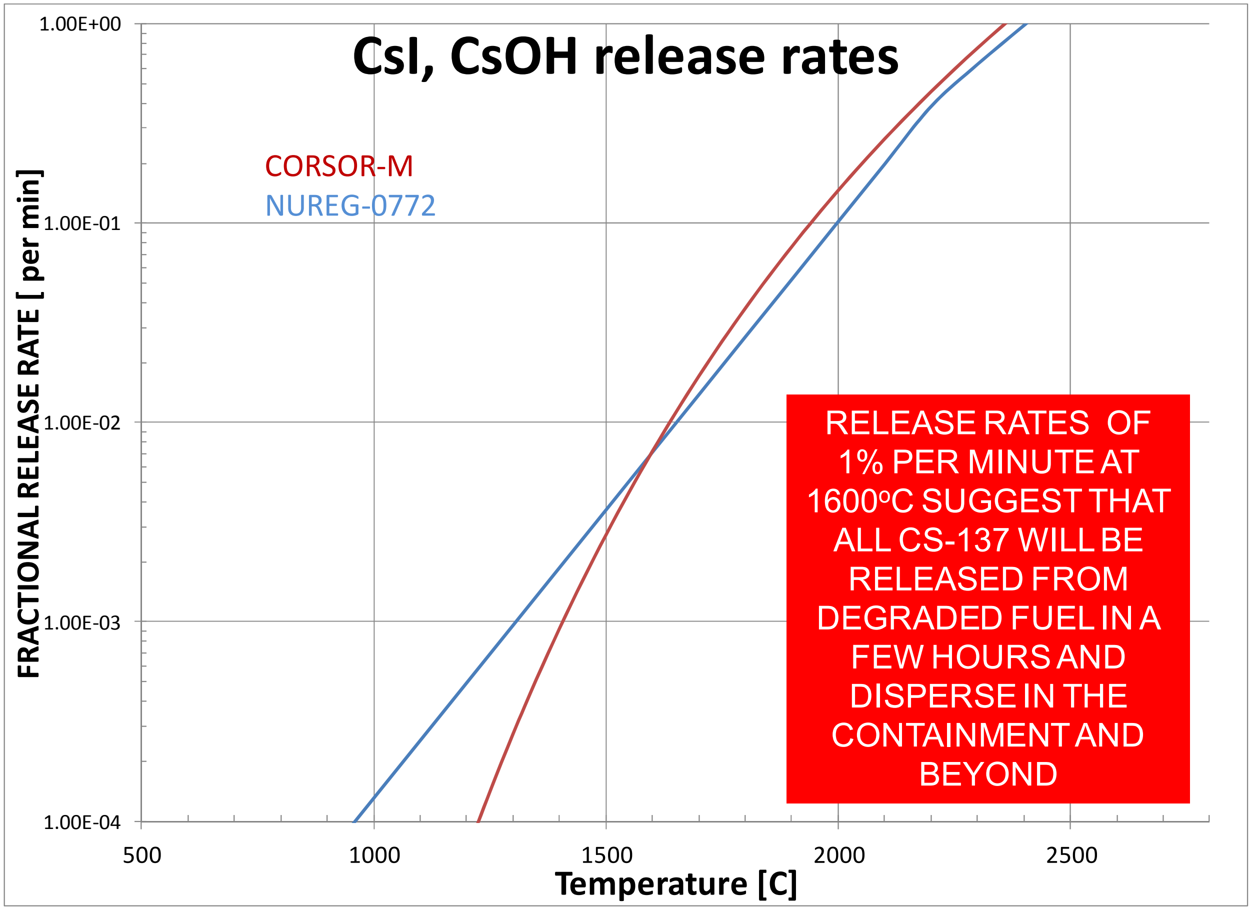
Power Groupings		
7500	6400	
6400	6300	
6300	6000	
6000	5700	
5700	5400	
5400	5000	
5000	4000	
4000	3000	

FISSION PRODUCTS COME OUT FAST ONCE THE FUEL IS HOT



SBO frequency as calculated by others

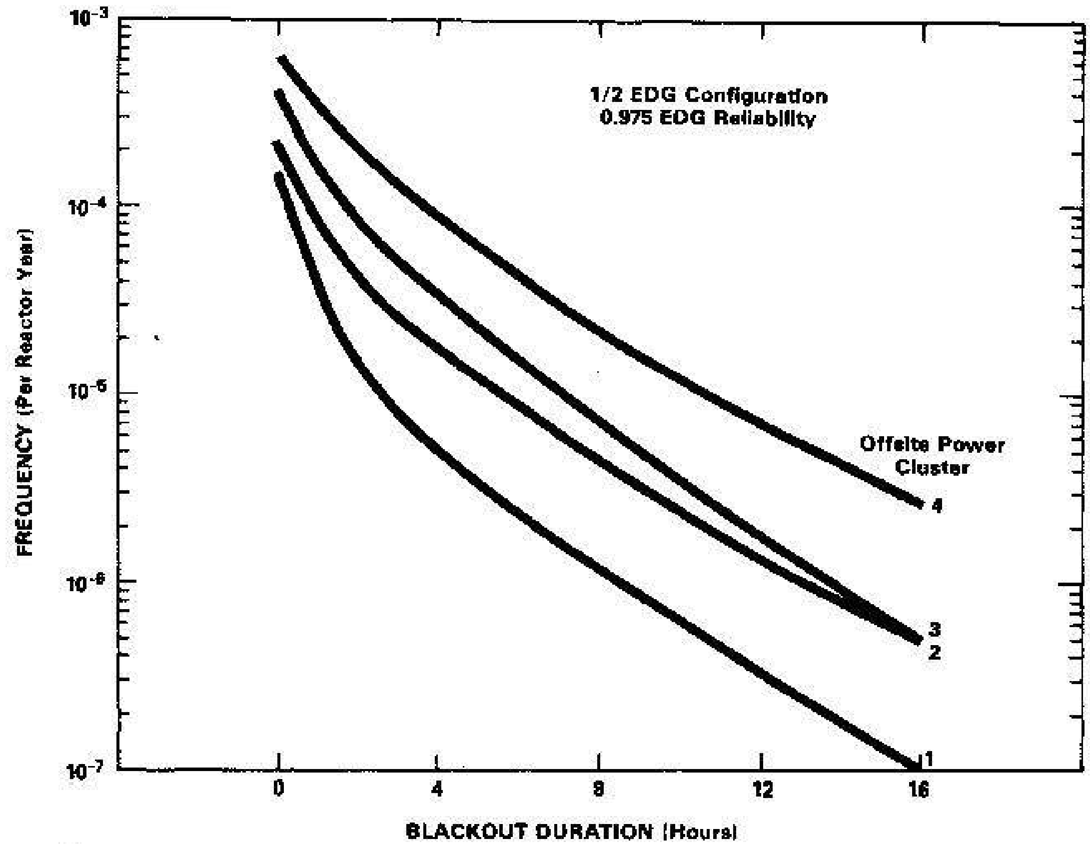
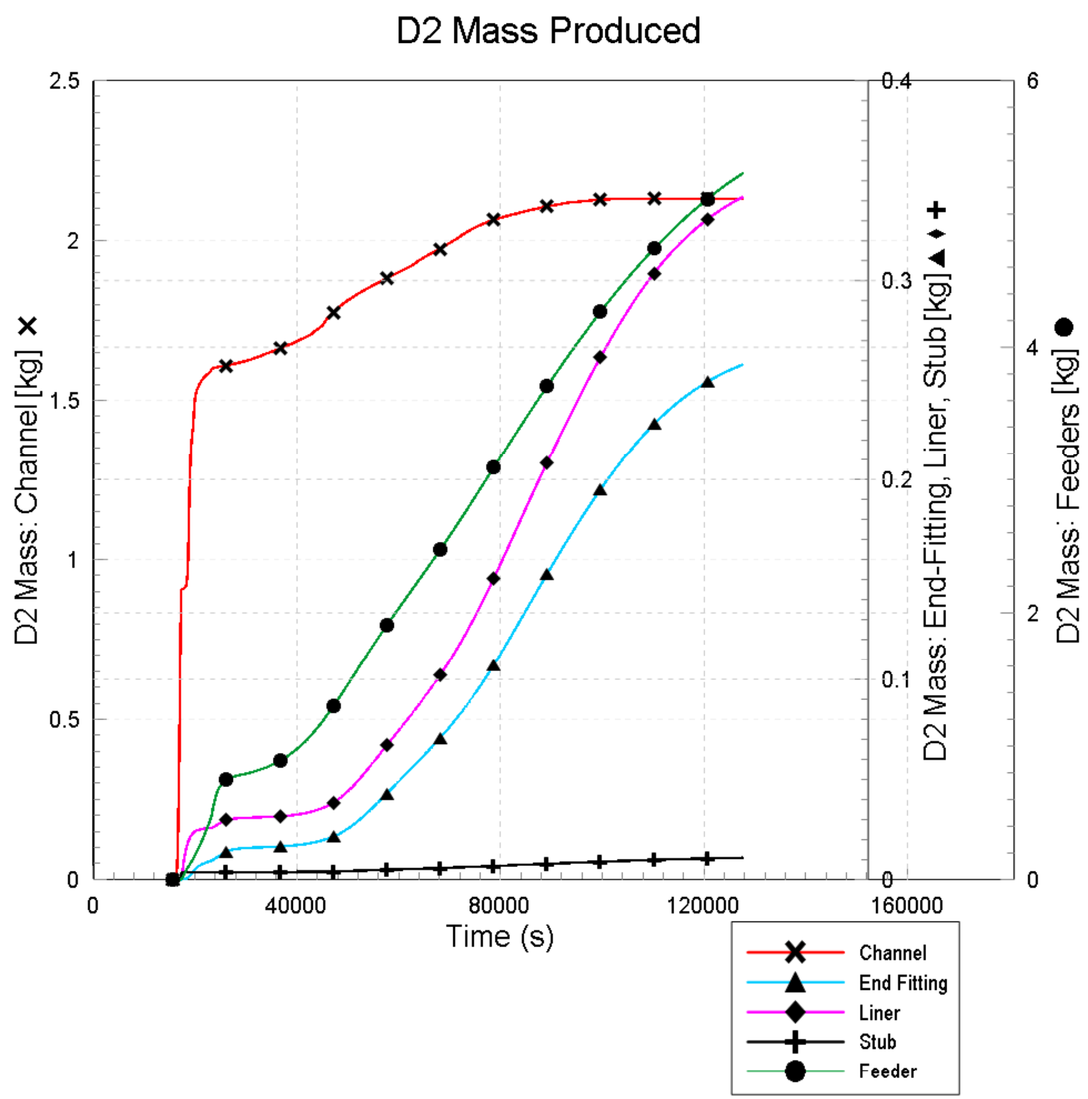


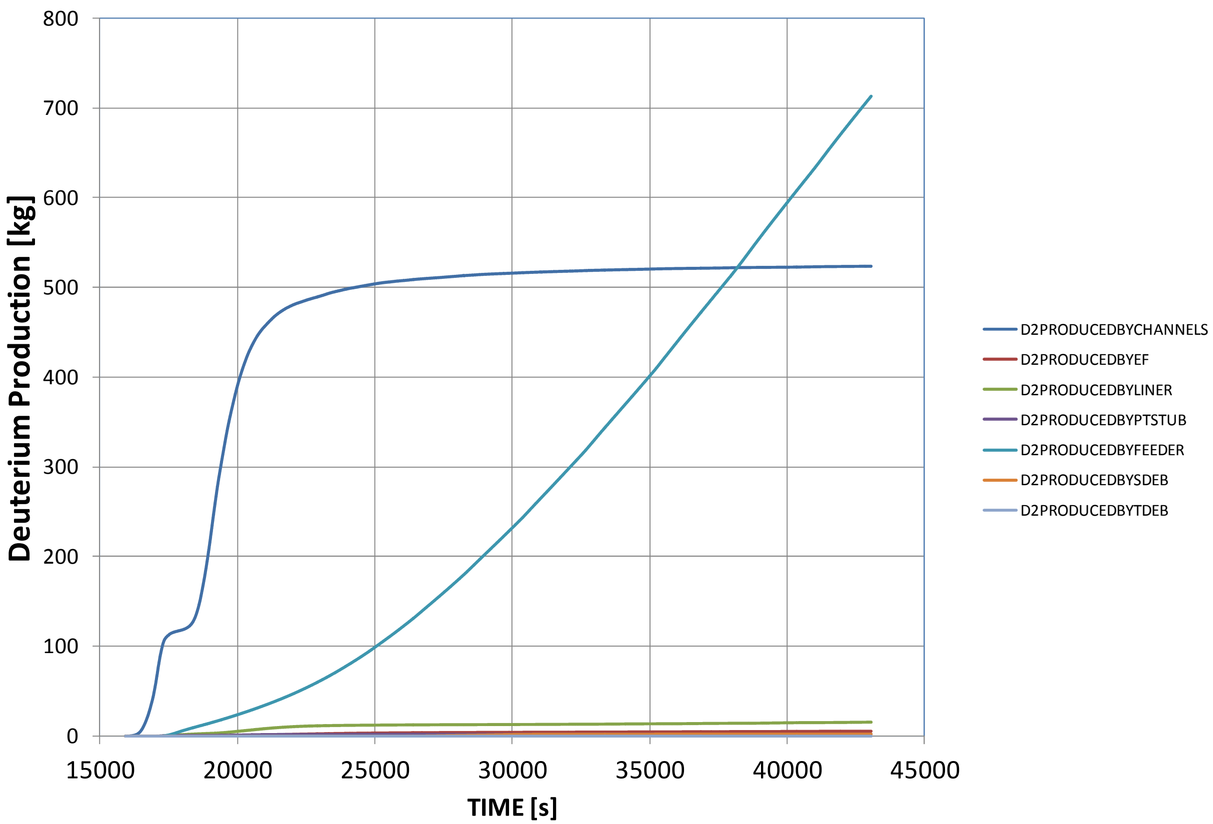
Figure 5.1 Estimated frequency of station blackout exceeding specified durations for several representative offsite power clusters

NUREG-1032 5-2

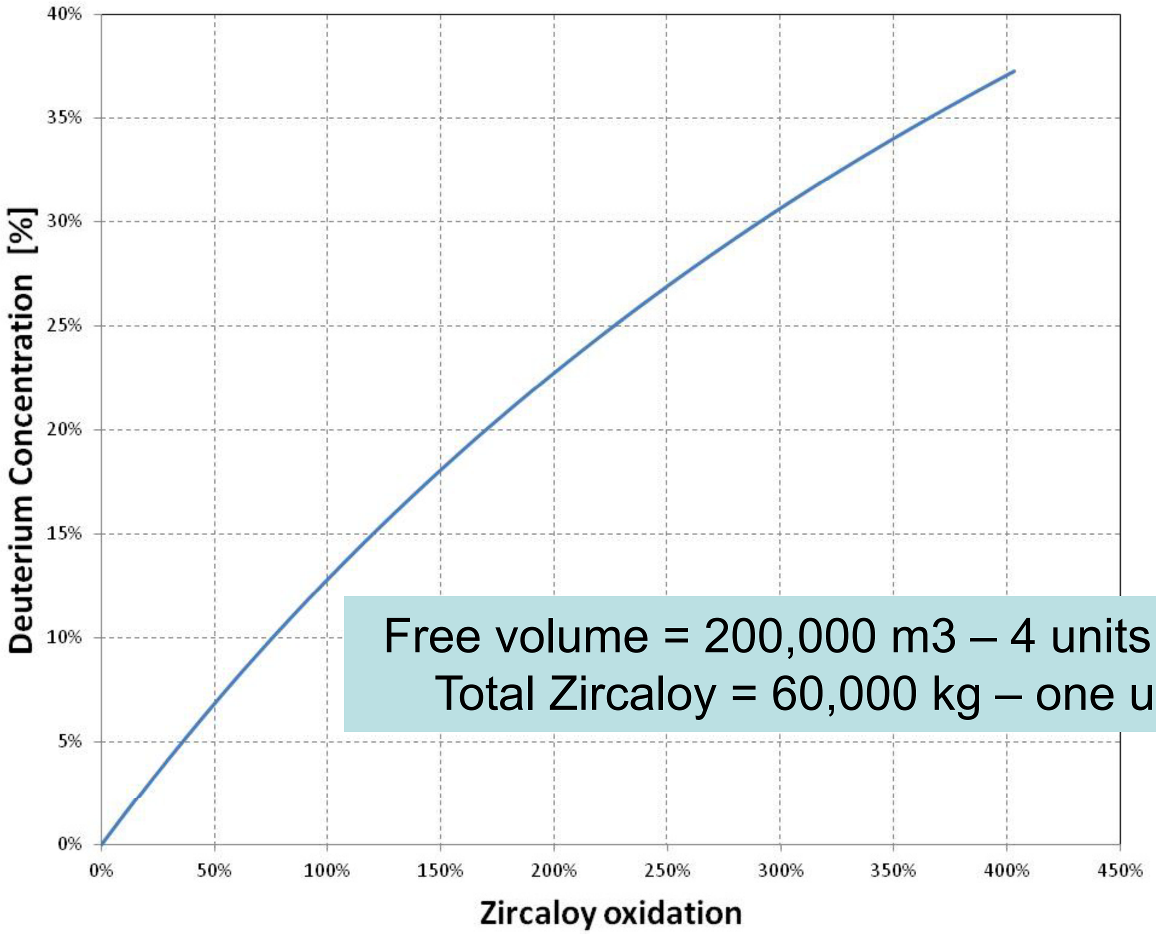
Deuterium gas production from a single channel



Short term D2 production in a CANDU6 Core



DARLINGTON 'HYDROGEN' CONCENTRATION



Free volume = 200,000 m3 – 4 units +VB
Total Zircaloy = 60,000 kg – one unit

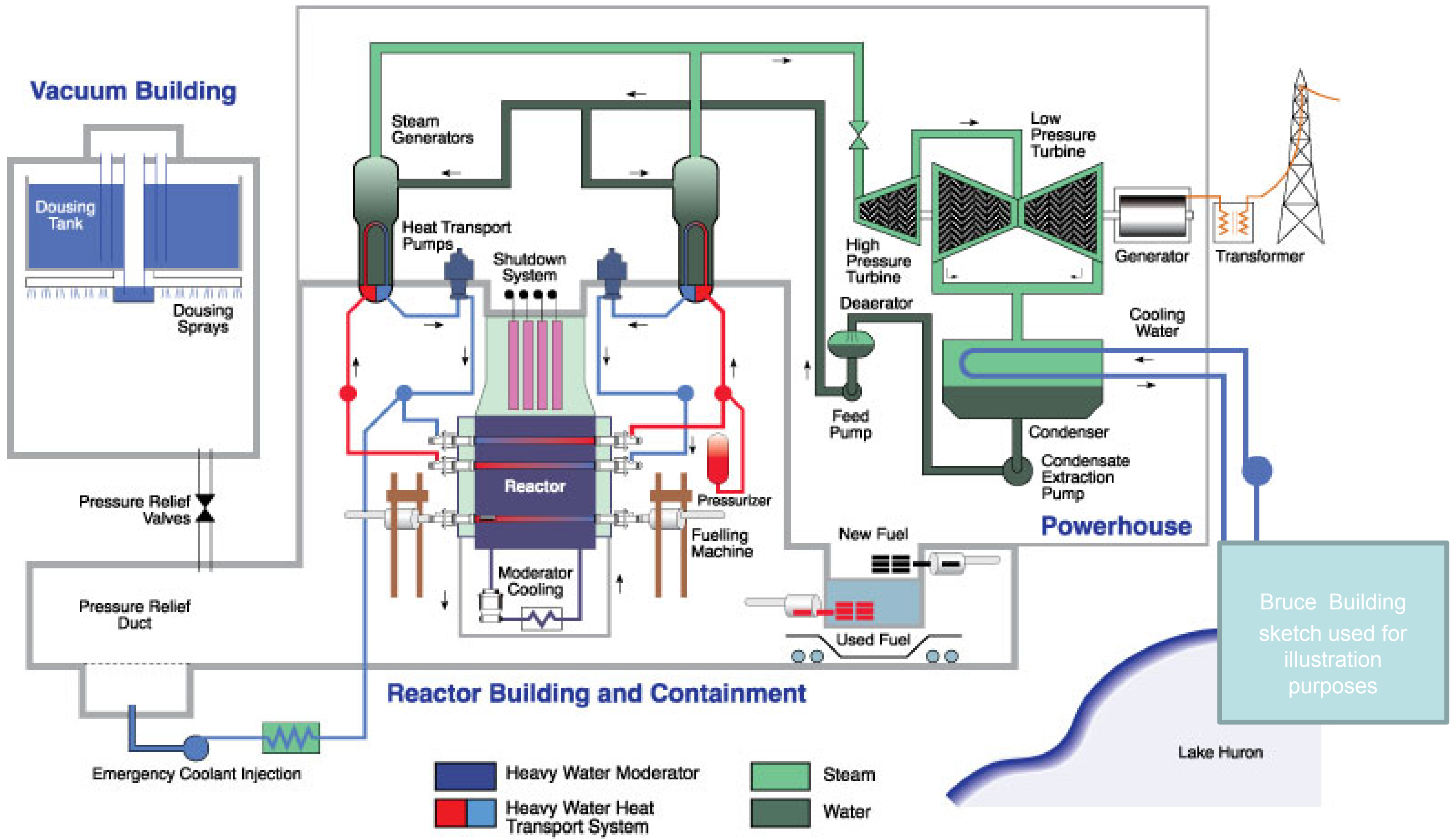
FUKUSHIMA REVIEW CONCLUSIONS

The TEPCO Fukushima Nuclear Power Plant accident was the result of collusion between the government, the regulators and TEPCO, and the lack of governance by said parties. They effectively betrayed the nation's right to be safe from nuclear accidents. Therefore, we conclude that the accident was clearly "manmade." We believe that the root causes were the organizational and regulatory systems that supported faulty rationales for decisions and actions, rather than issues relating to the competency of any specific individual.

The National Diet of Japan Nuclear Accident Independent Investigation Commission Report

IS CURRENT CANADIAN REGULATORY REGIME MUCH DIFFERENT ?

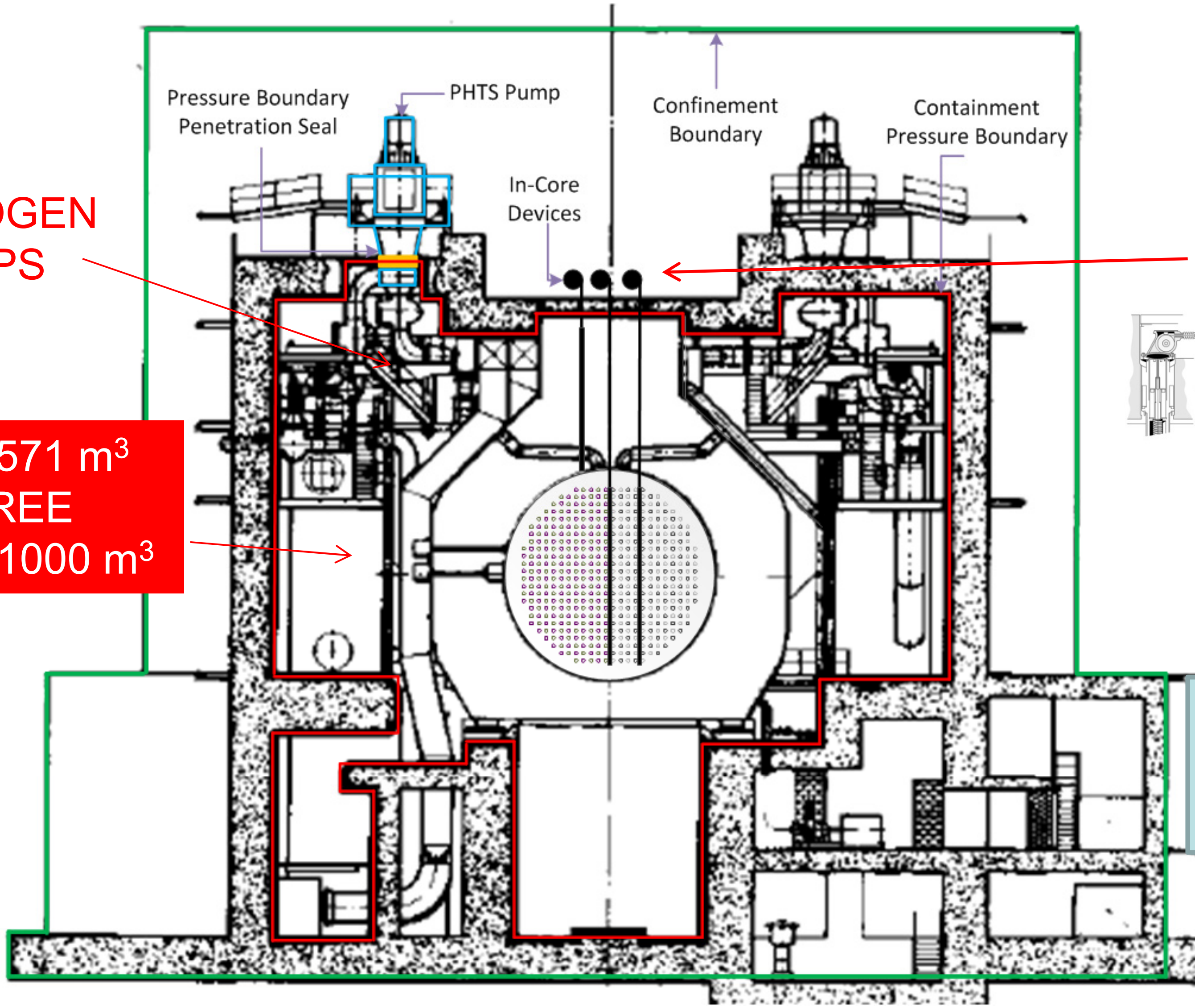
MANY COMPONENTS ARE OUTSIDE CONTAINMENT

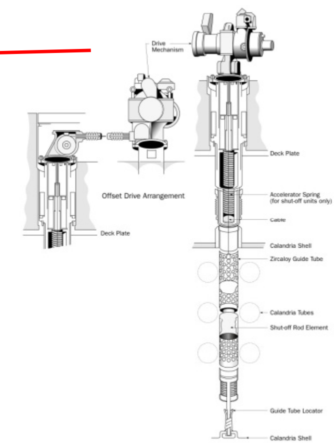


VULNERABLE LOW PRESSURE CONTAINMENT WITH SMALL REACTOR BUILDINGS, POOR D_2 MIXING

HYDROGEN TRAPS

VAULT 13,571 m^3
UNIT FREE VOLUME 21000 m^3



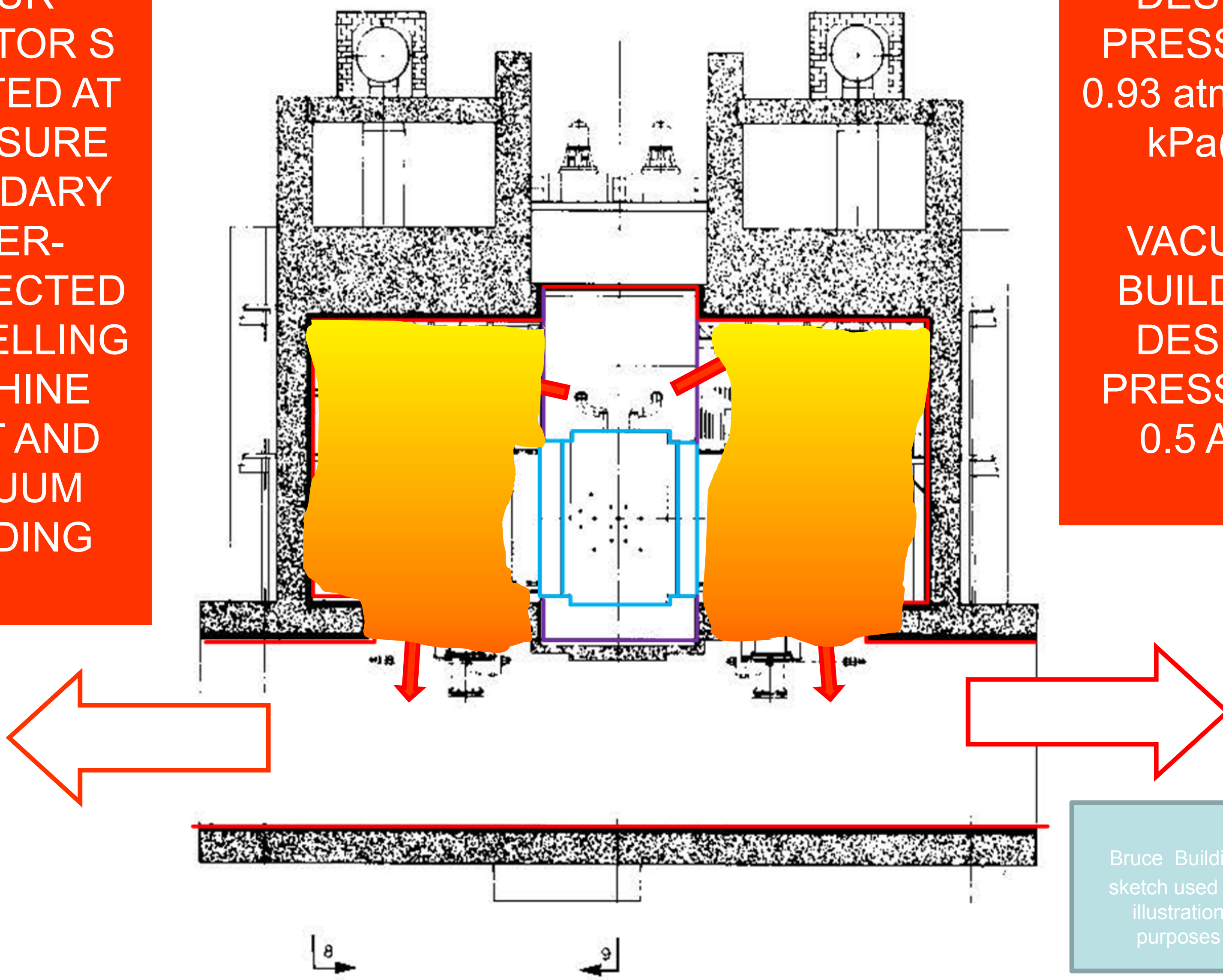


Bruce Building sketch used for illustration purposes

Total free volume 120,000 m^3 for 4 units, 95000 m^3 in vacuum building, max design pressure 48 kPa(g) at VB, 96.5 kPa(g) reactor vault

DARLINGTON reactor building

FOUR REACTOR S LOCATED AT PRESSURE BOUNDARY INTER-CONNECTED BY FUELLING MACHINE DUCT AND VACUUM BUILDING



DESIGN PRESSURE 0.93 atm (96.5 kPa(g))

VACUUM BUILDING DESIGN PRESSURE 0.5 ATM

Bruce Building sketch used for illustration purposes

DARLINGTON containment leakage characteristics unfavourable

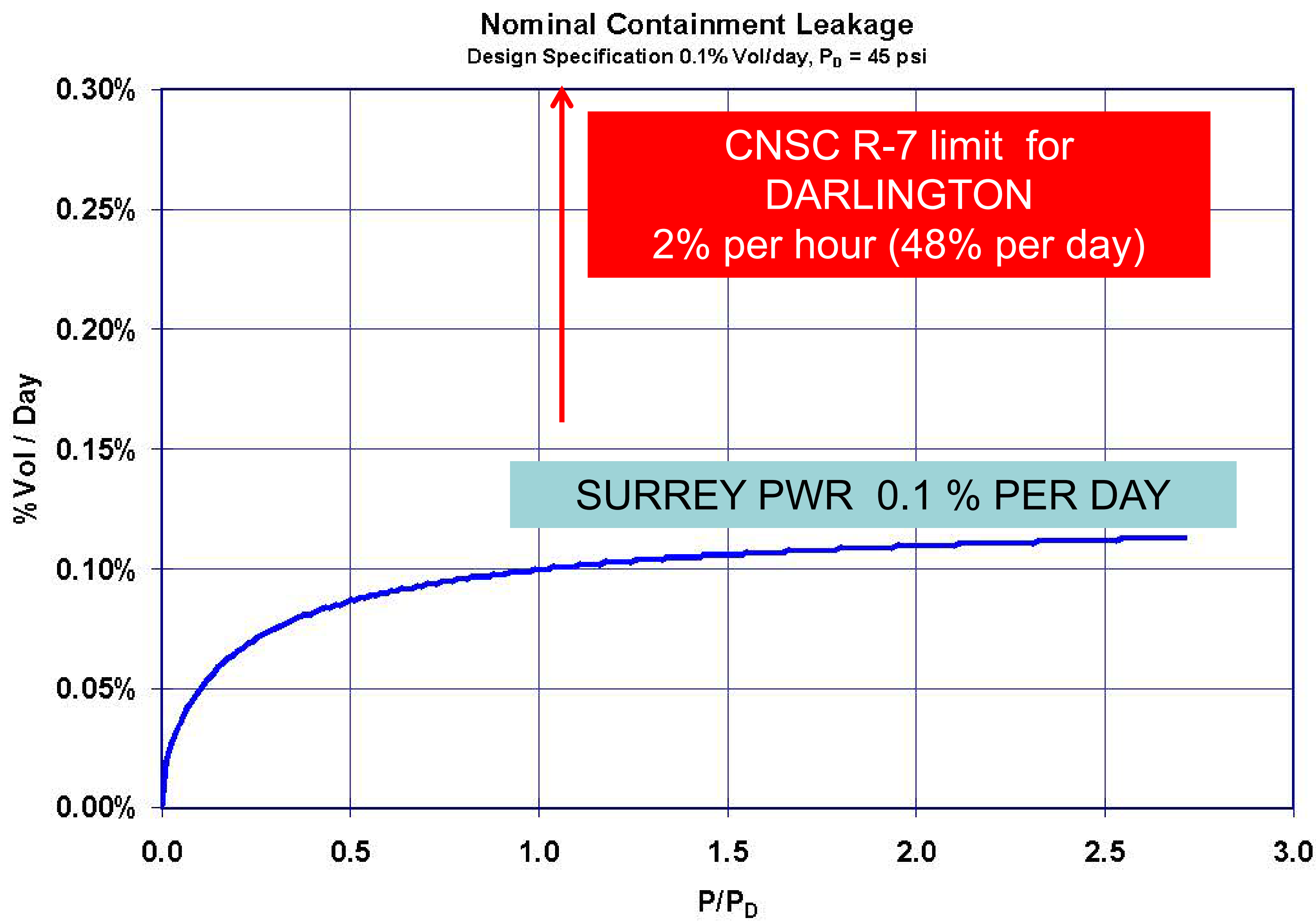
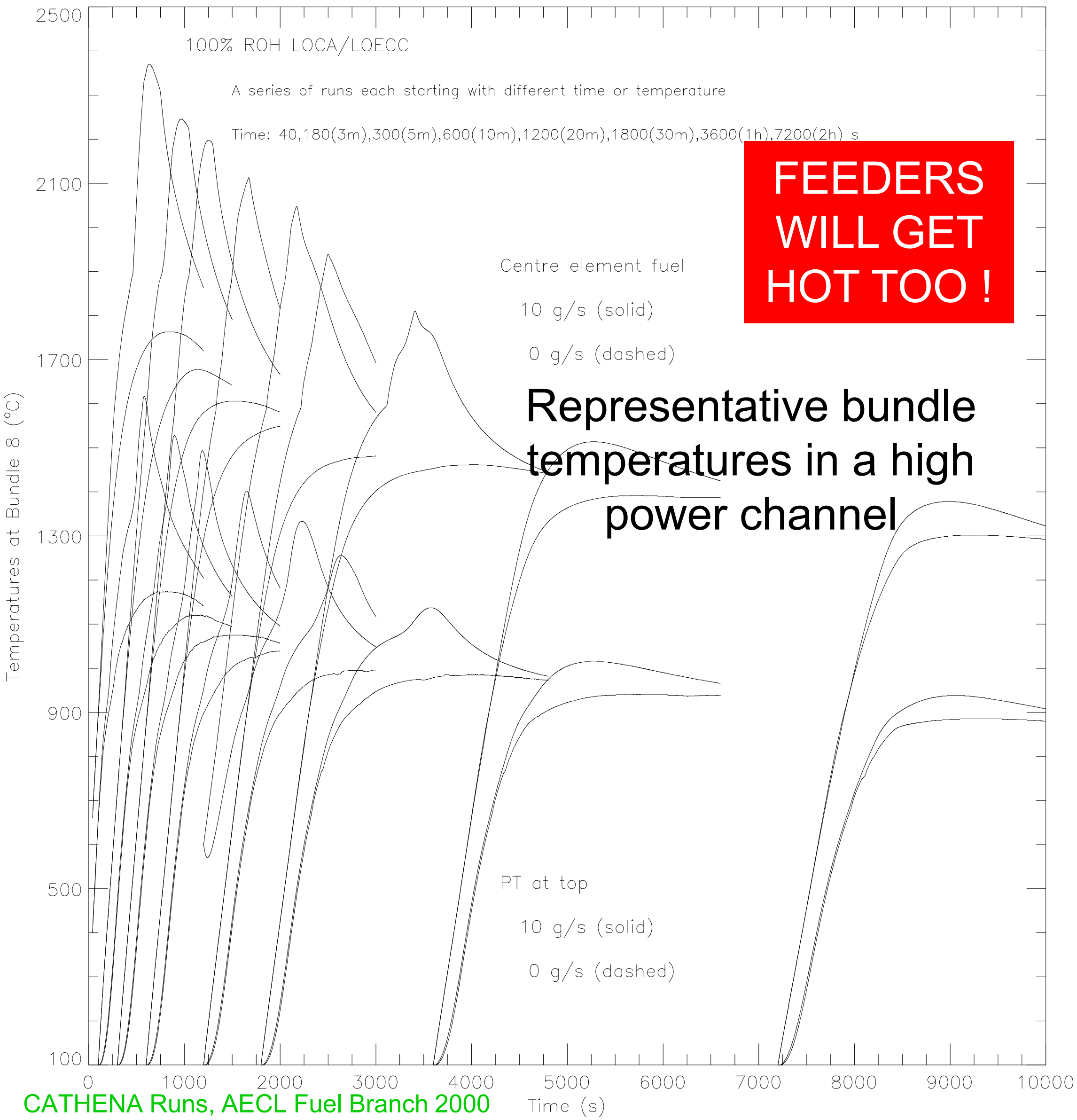


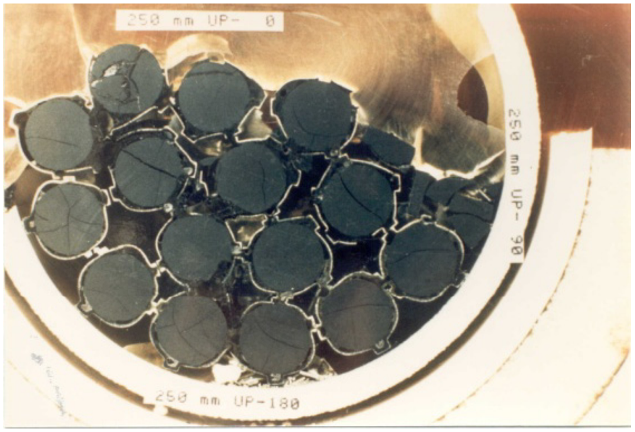
Figure 4-10 Nominal Containment Leakage Model

Source : NUREG-7110

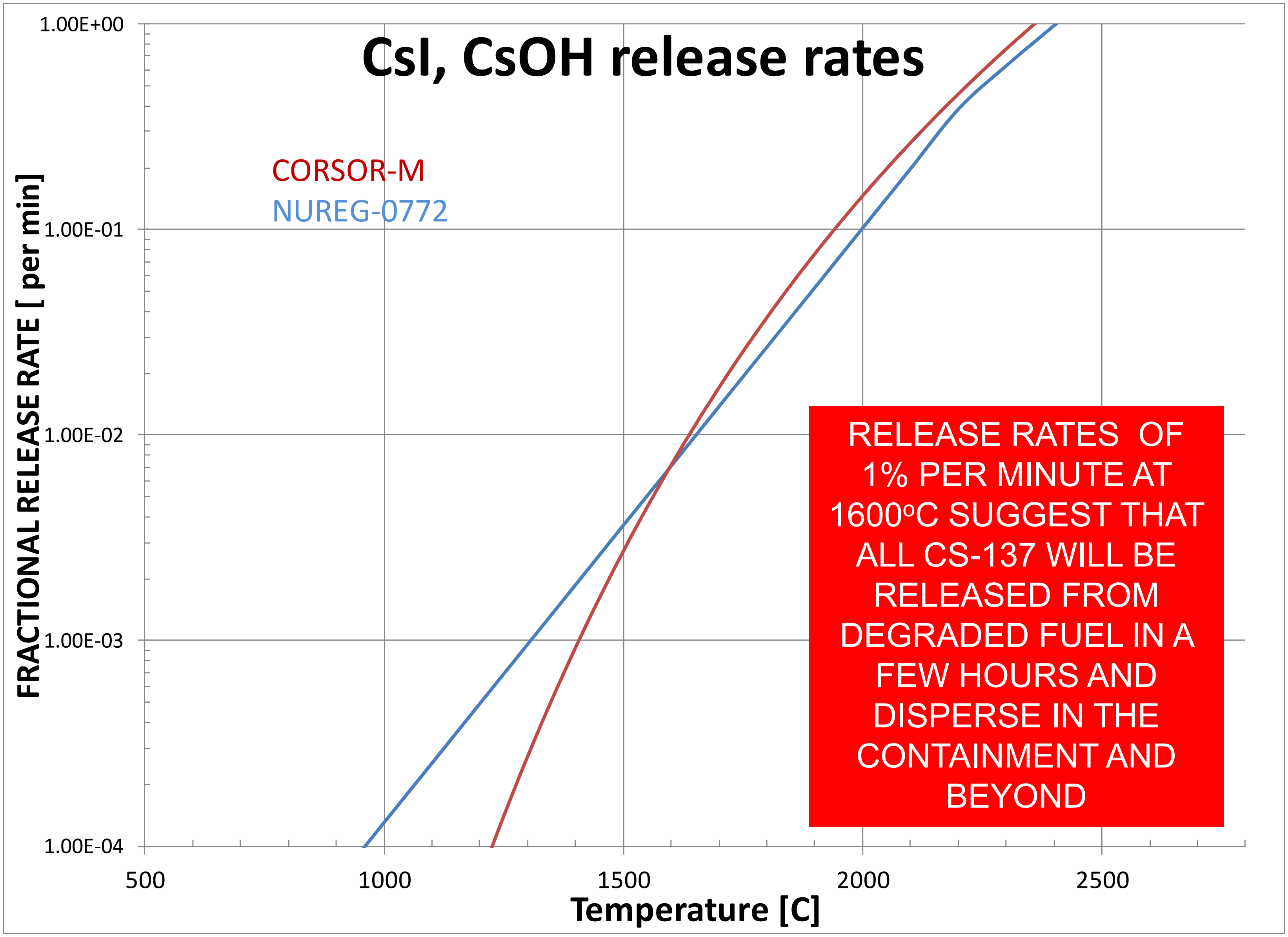
LOSS OF COOLING CAUSES HIGH TEMPERATURES



CATHENA Runs, AECL Fuel Branch 2000



Fission product release rates are high



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Power Groupings		
7500	6400	
6400	6300	
6300	6000	
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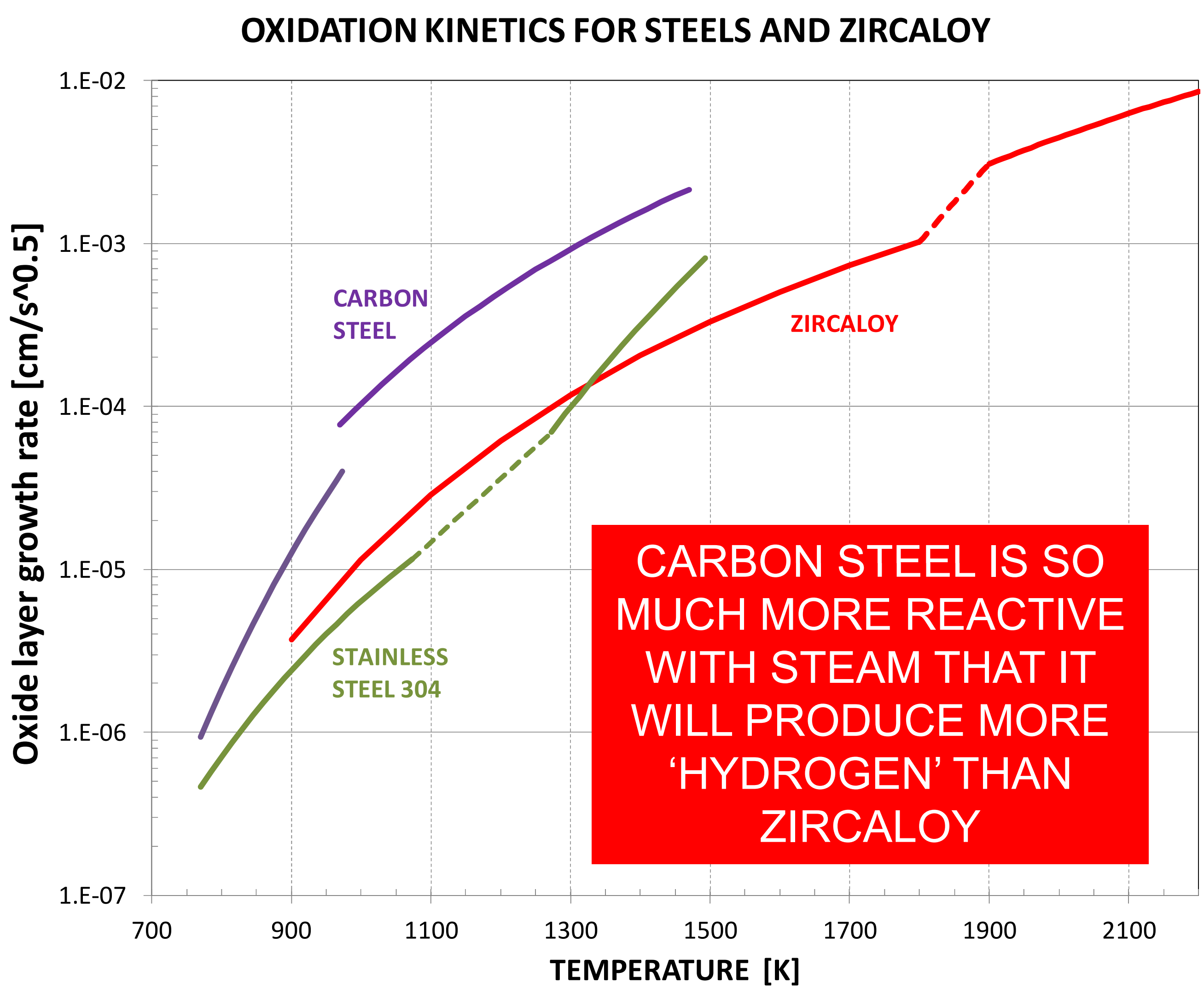
5700	5400	
5400	5000	
5000	4000	
4000	3000	

Feeder carbon steel oxidation bigger problem



Feeders are LARGE sources of flammable Deuterium

Material	Low carbon Low Cr steel (max 0.4%; actual as low as 0.04%) SA106- Grade B
Length	> 10,000 m (960 feeders)
Diameter	~2”, 2.5”,3”,3.5” non standard ID
Thickness	~5.6 to 8.1 mm
Area	> 2000 m^2
Mass	~> 110,000 kg

- Literally scores have been replaced in various plants due to thinning & actual or incipient cracking at bends
- Internal and external corrosion; thinning
- New simulations indicate very high oxidation under LOCA/LOECI and severe core damage accidents

EXTERNALLY CORRODED FEEDERS AT BRUCE

FUKUSHIMA HAS CHANGED PERCEPTION OF RISK

SEVERE ACCIDENTS & PHWR INDUSTRY

- **National Regulators**
  - Have not created specific requirements for mitigation capabilities for severe accidents or created a standard review plan for submissions.
  - Have not understood / accepted the need for and thus not demanded utilities to make meaningful design enhancements in a timely manner
  - Do not have the technical expertise to analyze severe accidents
  - Conflict with their self appointed position of promoters of reactors designs they regulate.
  - **Weak.** Regulate by consensus; suffer from Regulatory Capture.
  - Scant regulatory guidelines, rules for severe accidents.
- **Utilities**
  - Concentrate on meeting licensing requirements and operational issues.
  - Reluctant to undertake design enhancements unless specifically forced to do so.
  - **Powerful.** Got used to delaying resolution of even Generic Safety Issues
- **Designers**
  - Slow development of analytical methods for SA consequence assessments. **Only indirectly involved** in operating reactors.
  - Acknowledgment of design deficiencies hurts future sales
- **Research Organizations** – **Dependent.** Need better funding and independence.

PHWR INDUSTRY TASKS AHEAD

- **National Regulators**
  - Create specific requirements for mitigation capabilities for severe accidents and a standard review plan for submissions.
  - Acquire technical expertise to analyze severe accidents
  - Develop, firmly and fairly implement regulatory rules for severe accidents.
- **Utilities**
  - Concentrate on developing expertise on severe accident analyses, operator training and accident management
  - Examine potential design enhancements
- **Designers**
  - develop ingenious retrofits
- **Research Organizations**
  - Accelerate development of analytical methods for SA consequence assessments.
  - Undertake relevant experiments and develop better analytical tools.

TARGETS / ACCEPTANCE CRITERIA

- **Severe Core Damage Frequency**
  - Limit: 1E-04 events/year; Goal: 1E-05 events/year

- **Large Release Frequency**
  - Limit: 1E-05 events/year; Goal: 1E-06 events/year
  - Maximum off site release must be less than

    100 TBq of Cs-137 (LR); 1000 TBq of I-131 (SER).

    (There are ~90,000 TBq in each unit of Darlington at equilibrium)

    EFFECT OF OTHER RADIONUCLIDES AFTER 3MONTHS NOT MORE THAN BY Cs-137 (FINLAND)

- **No Containment Failure or Bypass**
  - The maximum containment pressure remains lower than the containment failure pressure for up to 24 hours after the onset of a severe accident.
  - The maximum pressure/temperature/radiation field at containment seals, penetrations and doors are below the failure limits for the seals and the containment, whichever is lower.
  - The hydrogen concentration remains below the limits for deflagration in any given volume of the containment.

- **Long Term Cold Shutdown and Control**
  - Known and controllable reactor state;
  - The debris have adequate area to spread in the reactor vault and there is high confidence that any debris in the reactor vault are covered with water
  - No recriticality

TARGETS / ACCEPTANCE CRITERIA

- **Land use restrictions**

  Swedish studies show that a release of several thousand TBq of Cs-137 would restrict land use.

  Swedish limits were 150 TBq; now 100 TBq.

- **Increased Cancer risk ;** Acceptable risk 1E-6 to 1E-4 ?

  For chronic health effects above 0.1 Sv (100 mSv), the cancer risk can be approximated as increasing by 10%/Sv (linear no threshold dose model)

Radionuclide Cancer Morbidity - Slope Factors

Slope Factor (Morbidity Risk Coefficient)

Lifetime Excess Cancer Risk per Unit Exposure

Radionuclide	Inventory Low Burnup Curies	per Bundle High Burnup Curies	Whole Core Inventory 25%-75% estimate Curies	Half Life	Water Ingestion (risk/pCi)	Food Ingestion (risk/pCi)	Soil Ingestion (risk/pCi)	Inhalation (risk/pCi)	External Exposure (risk/yr per PCi/g soil)
Ce-144+D	5772	12680	4.99E+07	284 days	3.53E-11	5.19E-11	1.02E-10	1.1E-10	2.44E-07
Cs-137+D	223	897	3.32E+06	30.17 yrs	3.04E-11	3.74E-11	4.33E-11	1.19E-11	2.55E-06
Ru-106+D	1450	8276	3.00E+07	373 days	4.22E-11	6.11E-11	1.19E-10	1.02E-10	9.66E-07
Sb-125+D	28	139	5.07E+05	2.785 yrs	5.13E-12	7.21E-12	1.32E-11	1.93E-11	1.81E-06
Sr-90+D	174	498	1.90E+06	28.8 yrs	7.4E-11	9.53E-11	1.44E-10	1.13E-10	1.96E-08

Soil Screening Guidance: User's Guide (EPA/540/R-96/018),

- **Food contamination**

  Food specific thresholds; e.g. milk

TARGETS / ACCEPTANCE CRITERIA

Table 1: Current Council Regulation

Radionuclide	Maximum permitted levels ($Bq\ kg^{-1}$)				
	Baby foods	Dairy produce	Minor foods	Other foods	Liquid foods
Isotopes of strontium, notably ^{90}Sr	75	125	7,500	750	125
Isotopes of iodine, notably ^{131}I	150	500	20,000	2,000	500
Alpha-emitting isotopes of plutonium and transplutonium elements	1	20	800	80	20
All other radionuclides of half-life greater than 10 days, notably ^{134}Cs and ^{137}Cs*	400	1,000	12,500	1,250	1,000

* ^{14}C and ^{3}H excluded

EU Food Restriction Criteria After An Accident - 1998

Observations on Some Primary Causes of Fukushima

- Institutional and regulatory failure
- Inappropriate safety culture; over confidence on NPP safety
- Insufficient expertise with decision makers
- Insufficient understanding of severe accident phenomenology & progression
- Improper accident management
- Improper and insufficient understanding of reactor conditions
- No timely advice sought or available from external experts
- Insufficient exchange/transfer of information among and within organizations

IS CURRENT CANADIAN REGULATORY REGIME MUCH DIFFERENT ?

Source - Causes of and Lessons from Fukushima Accident, Won-Pil Baek, VP Nuclear Safety Research, KAERI, NUSSA 2012

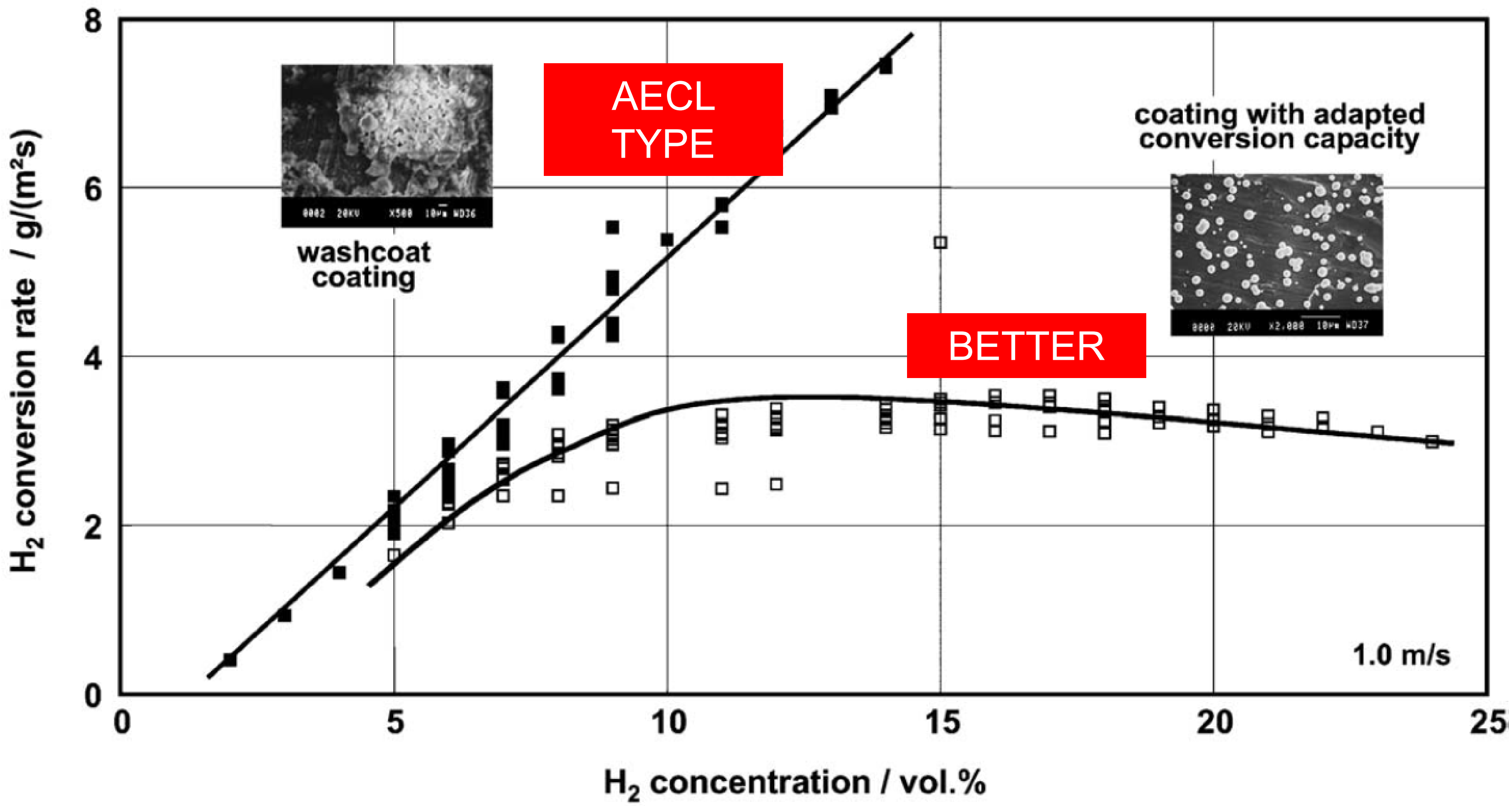
Lessons for all reactors from Fukushima

- Strengthening of safety culture, including an independent assessment system
- Practical countermeasures against severe accidents
- Improvement of NPP procedures, covering up to extreme severe accident scenarios
- Enhancement of NPP instrumentation
- Improvements in diversity & reliability of emergency power supply systems
- Reliable decay heat removal by strengthening passive safety
- Improvement and strengthening of defense in depth strategy
- Effective nuclear safety research and sharing of research outputs
- Enhancement of regulatory standards
- Strengthened independence & expertise of regulatory organizations
- Emphasized role and enhanced capability of operating organizations

HOW MUCH HAS ACTUALLY BEEN ACHIEVED IN FOUR YEARS SINCE FUKUSHIMA?

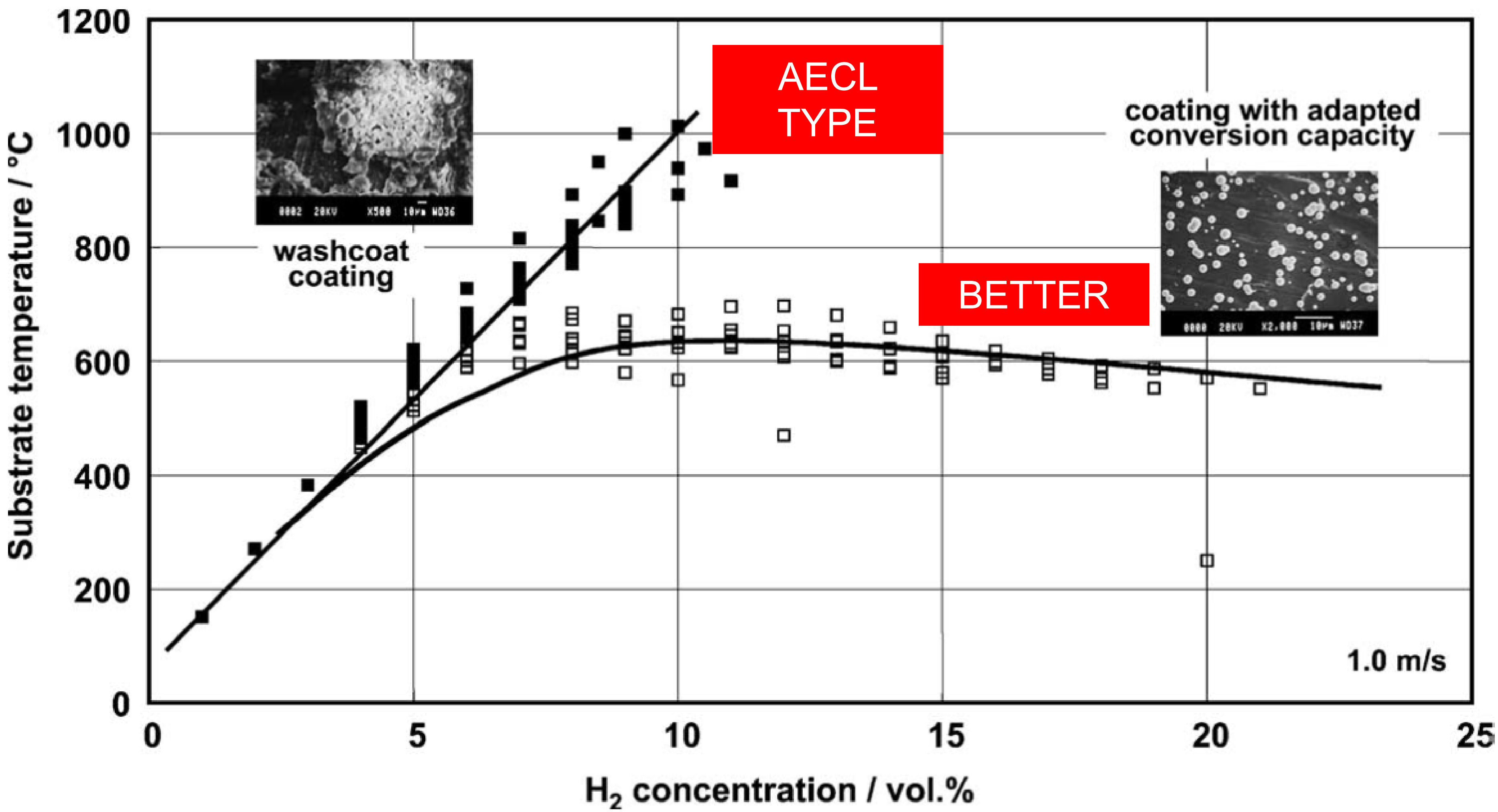
Source - Causes of and Lessons from Fukushima Accident, Won-Pil BAEK, VP Nuclear Safety Research, KAERI, NUSSA 2012

AECL RECOMBINERS WILL CAUSE D2 EXPLOSIONS



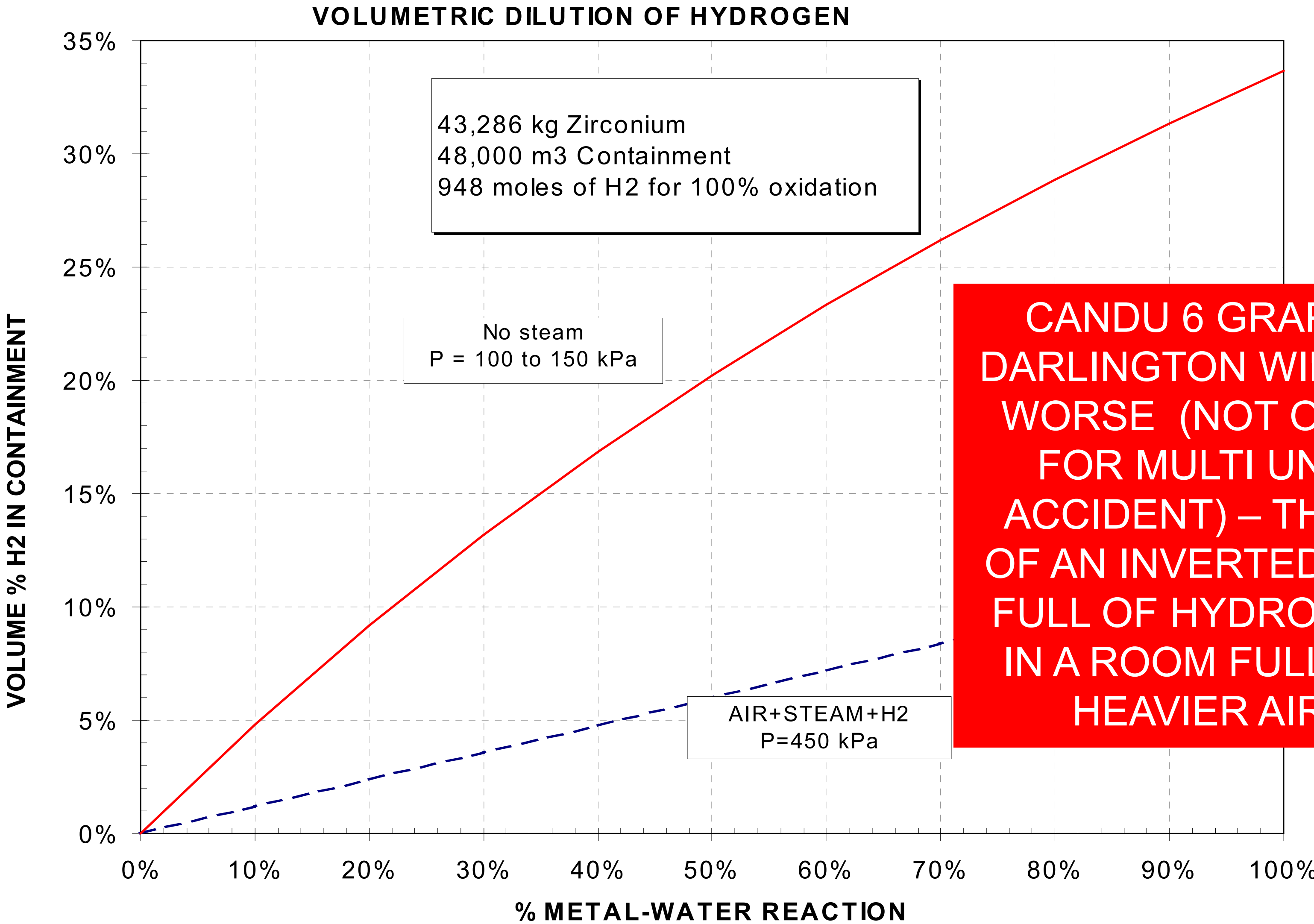
Studies on innovative hydrogen recombiners as safety devices in the containments of light water reactors

E.-A. Reinecke et al. / Nuclear Engineering and Design 230 (2004) 49–59



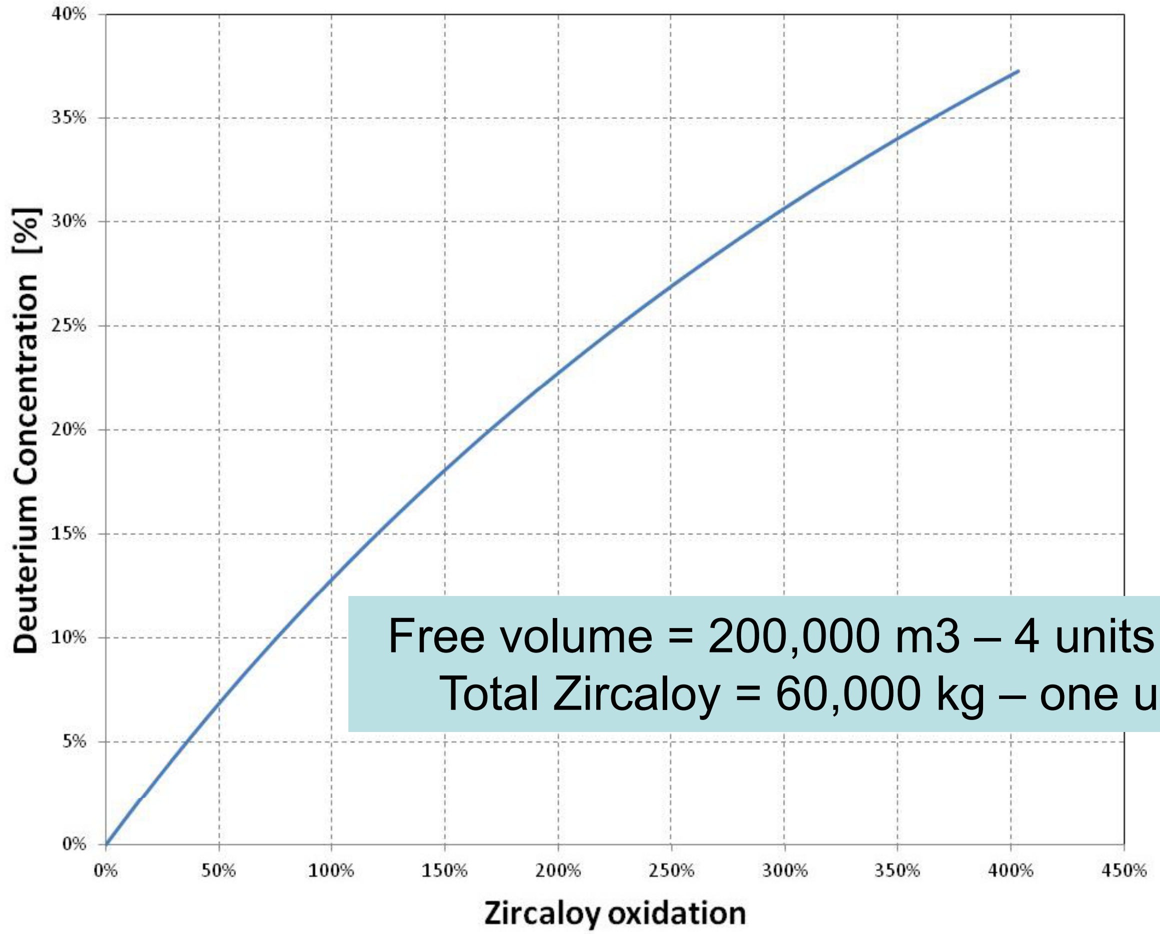
E.-A. Reinecke et al. / Nuclear Engineering and Design 230 (2004) 49–59

Enough hydrogen to burn or detonate ?



CANDU 6 GRAPH; DARLINGTON WILL BE WORSE (NOT ONLY FOR MULTI UNIT ACCIDENT) – THINK OF AN INVERTED JAR FULL OF HYDROGEN IN A ROOM FULL OF HEAVIER AIR

DARLINGTON 'HYDROGEN' CONCENTRATION



Free volume = 200,000 m3 – 4 units +VB
Total Zircaloy = 60,000 kg – one unit

OVERPRESSURE FAILURES UNIQUE TO CANDU

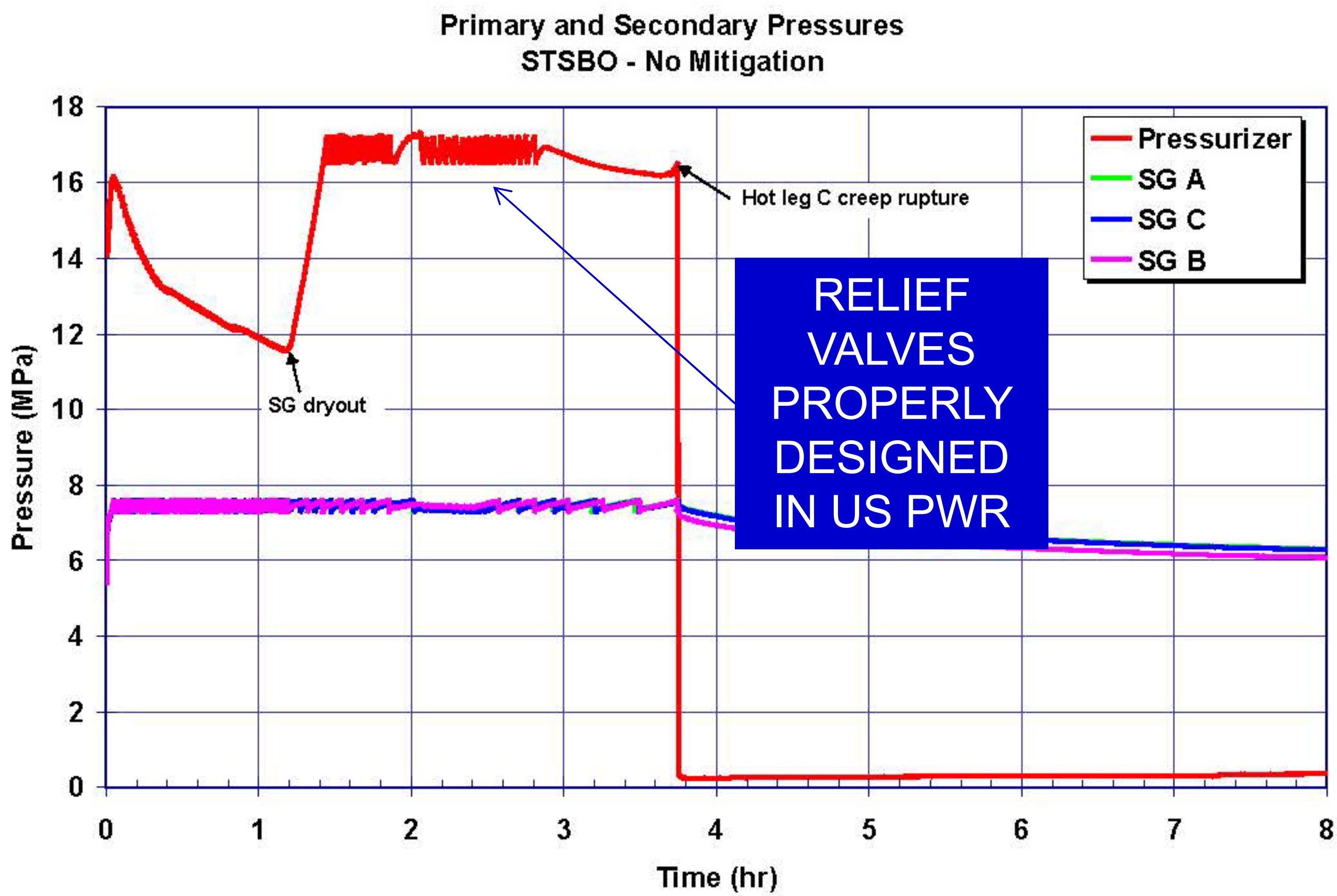
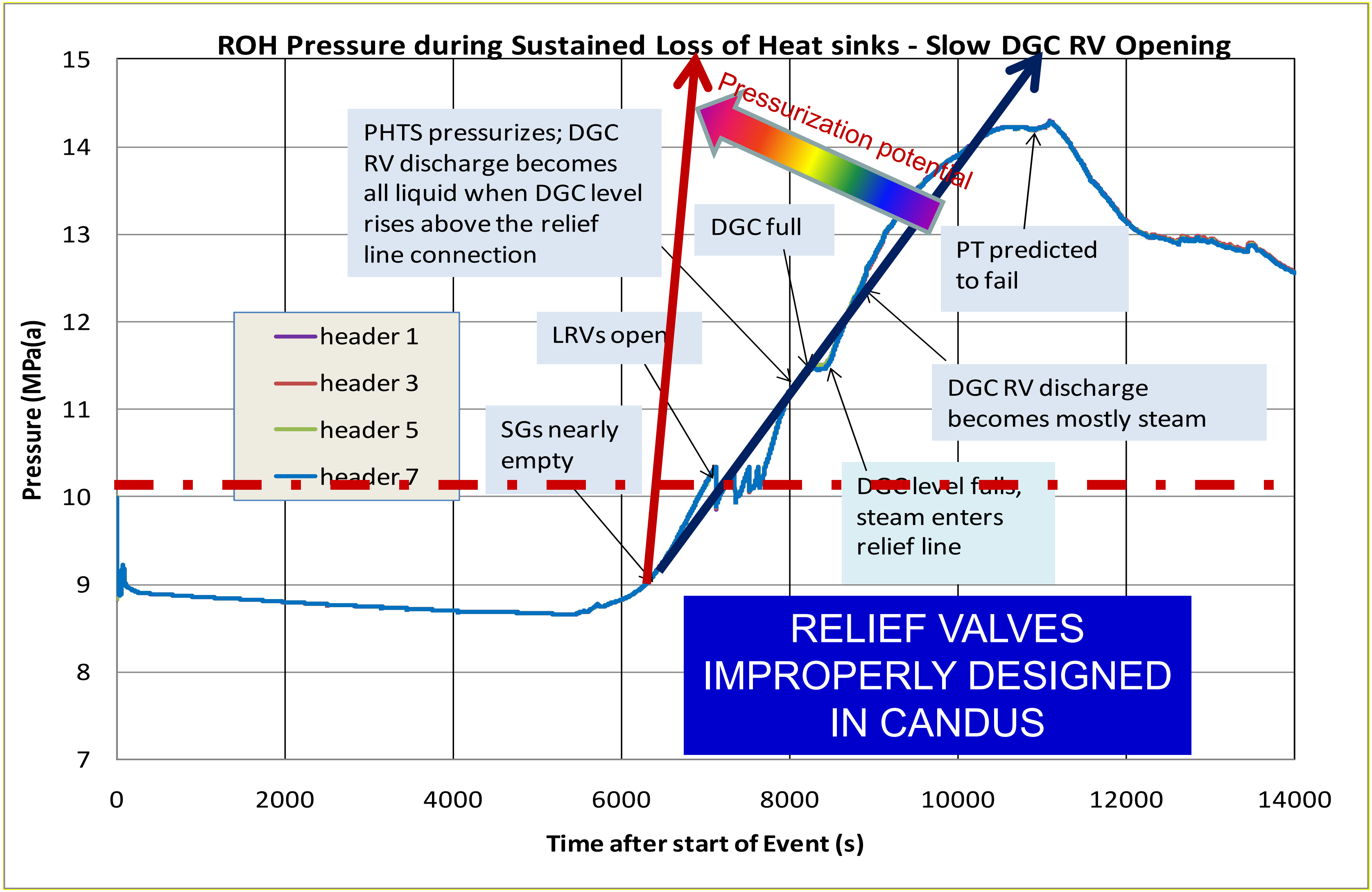


Figure 5-28 Unmitigated STSBO primary and secondary pressures history

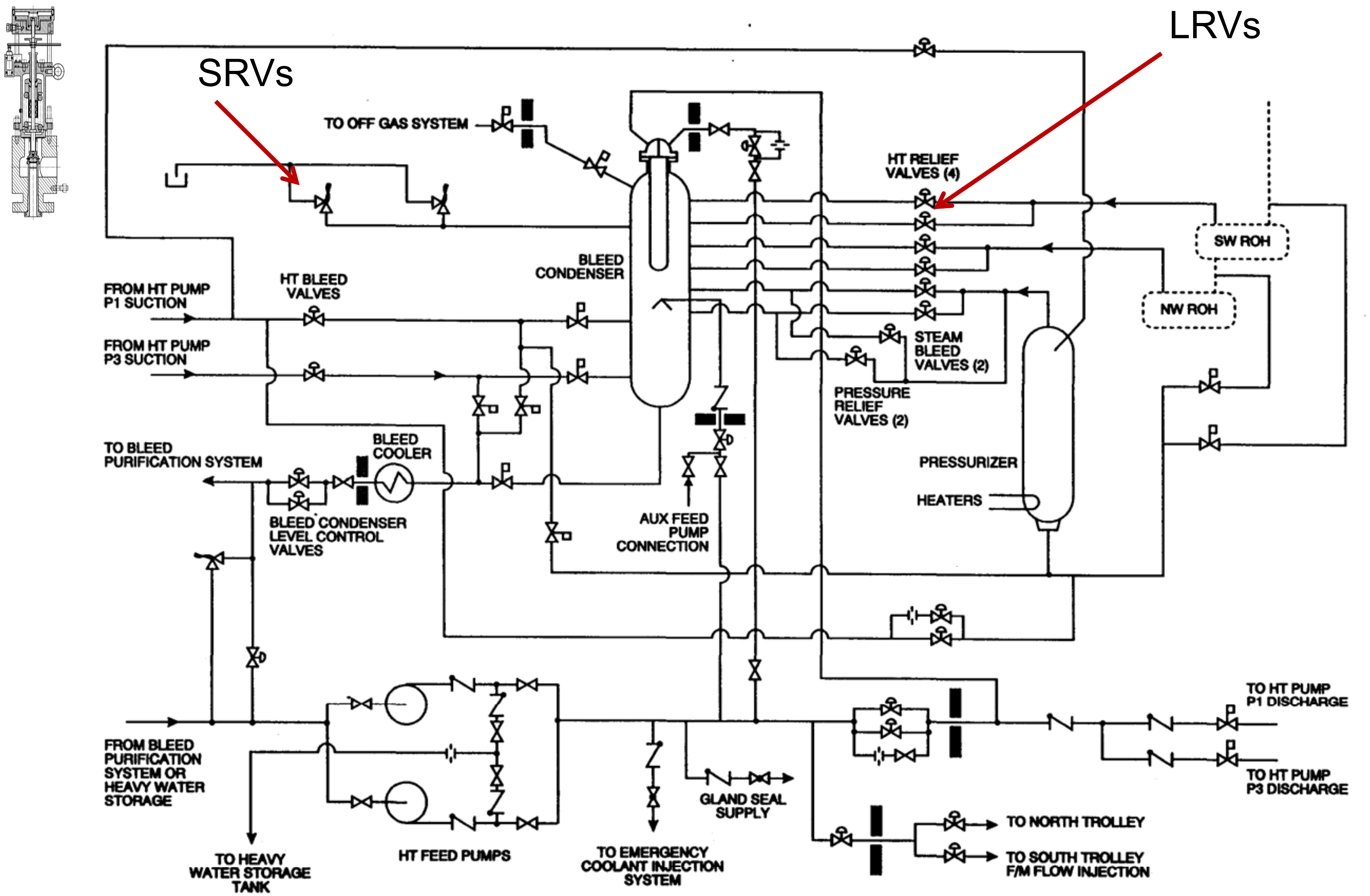
Source – NUREG/CR 7110

EXAMPLE OF UNDESIRABLE RESPONSE – CANDU 6; SAME EXPECTED FOR DARLINGTON

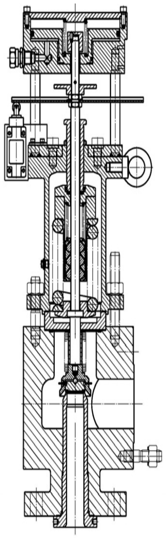


Source : AECL 2011

DARLINGTON HTS OVER-PRESSURE PROTECTION IS WEAK



DARLINGTON HTS OVER-PRESSURE PROTECTION IS WEAK

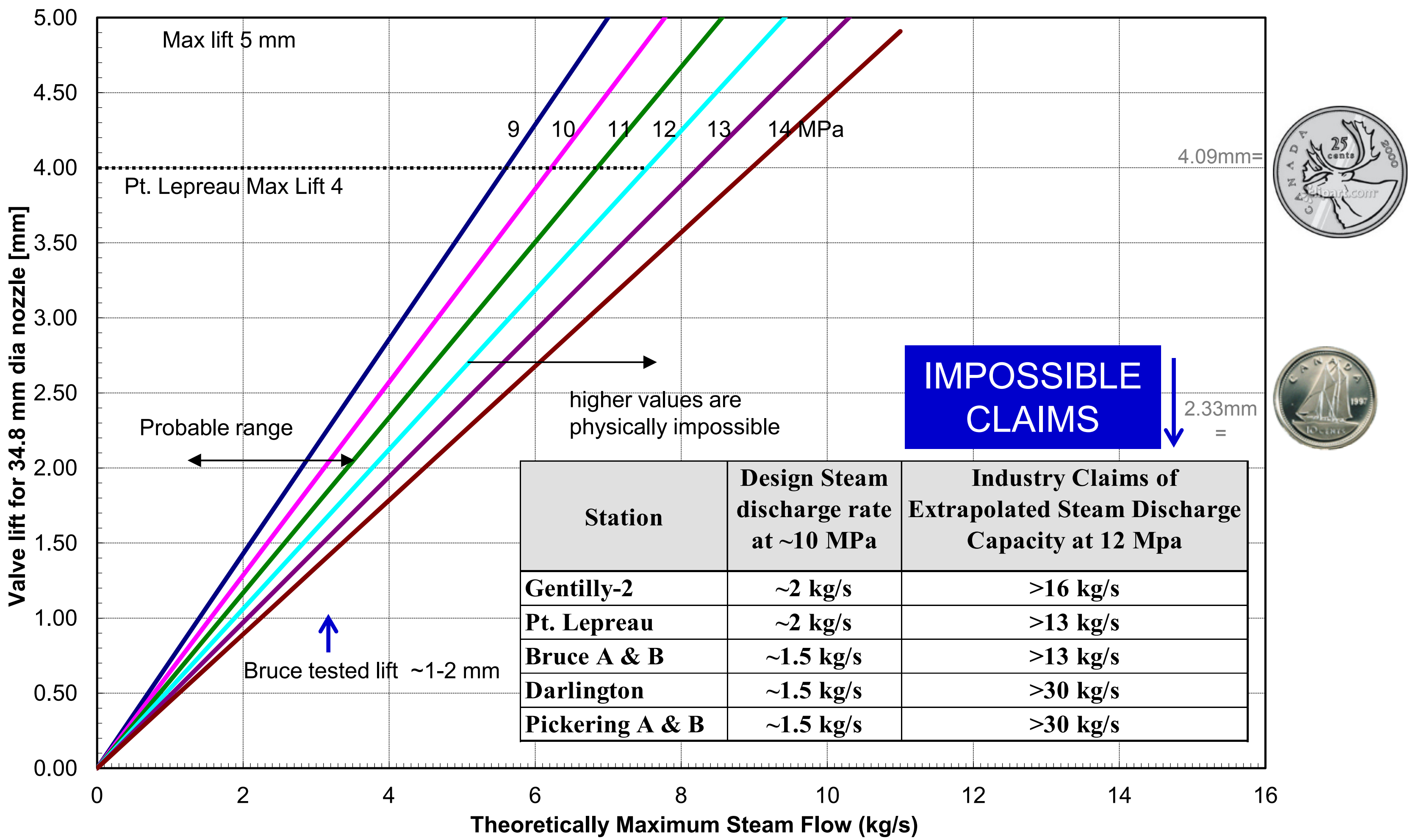


SRVs

- TWO '100%' SAFETY RELIEF VALVES ON BLEED CONDENSER (63332-RV25,26, NPS 4x6)
- RATED CAPACITY 100 l/s of LIQUID D_2O
- STEAM D_2O CAPACITY REPORTED TO CNSC IN 2003 BY OPG AT 1.5 kg/s per VALVE
- ONLY ONE VALVE REQUIRED FOR OPERATION
- VALVES ARE IMPROPERLY DESIGNED FOR LOSS OF HEAT SINKS

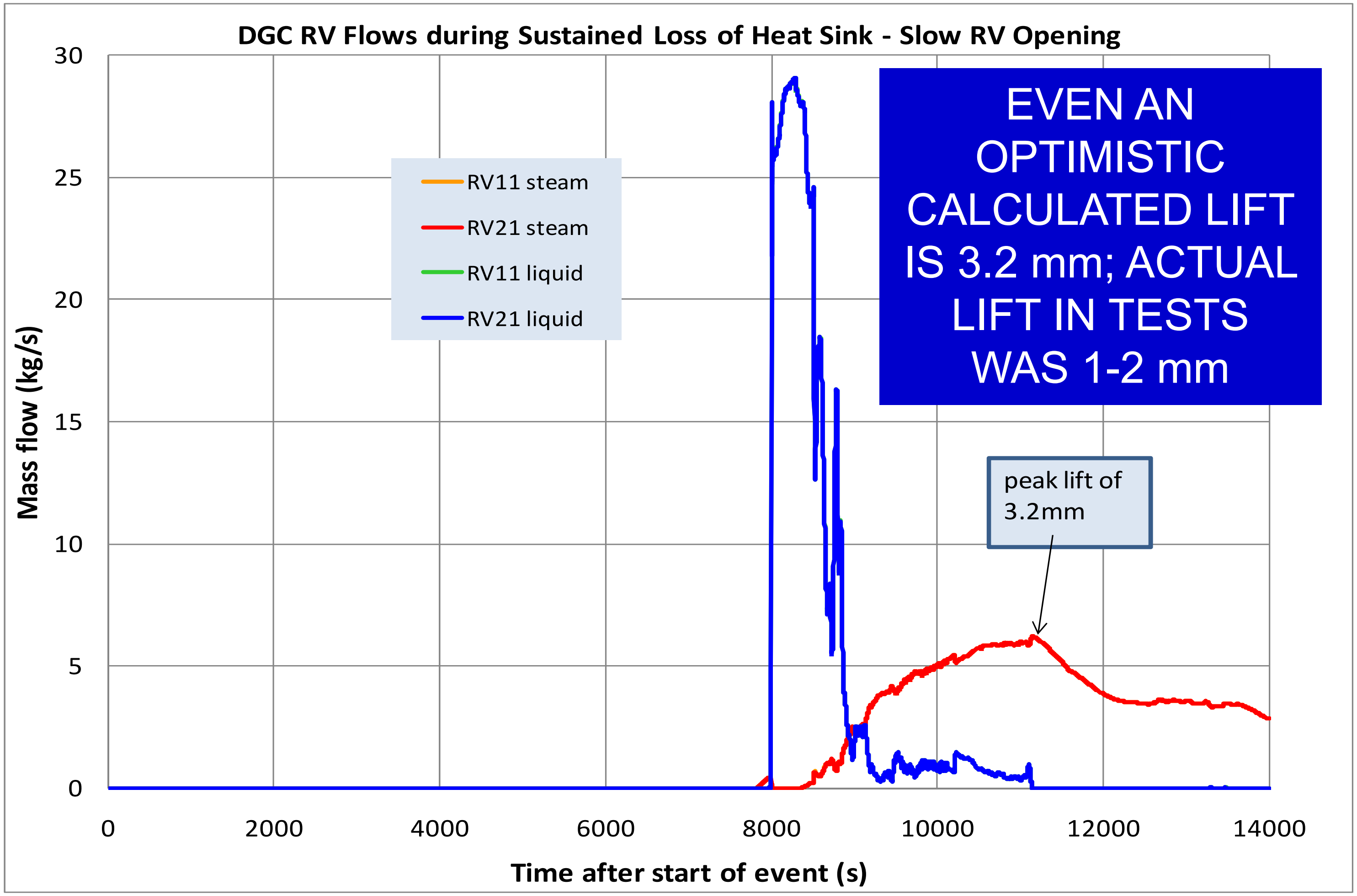
INSTALLED SAFETY VALVES CANNOT RELIEVE DECAY HEAT

Theoretically max choked Steam Flow through a hole for a range of pressures (kPa)



Station	Design Steam discharge rate at ~10 MPa	Industry Claims of Extrapolated Steam Discharge Capacity at 12 Mpa
Gentilly-2	~2 kg/s	>16 kg/s
Pt. Lepreau	~2 kg/s	>13 kg/s
Bruce A & B	~1.5 kg/s	>13 kg/s
Darlington	~1.5 kg/s	>30 kg/s
Pickering A & B	~1.5 kg/s	>30 kg/s

AECL August 2011 CATHENA analysis



OPG NEEDS TO DEMONSTRATE EFFORTS ALREADY TAKEN TO…

1. Further reduce the likelihood of a station blackout scenario that starts with a loss of off-site power or a malevolent act.
2. Reduce the likelihood of events and failures that create permutations of failures that may lead to severe core damage accident from other internal and external events
3. Reduce the likelihood of incidents progressing to a core damage state by measures such as external and internal hookups for adding power and water; daerator hookup.
4. Reduce the likelihood of an uncontrolled rupture of heat transport system pressure boundary at the onset of boiler dryout in case of a station blackout as at Fukushima.
5. Correct the inadequacy of heat transport system over pressure protection
6. Reduce the likelihood of containment bypass in boilers

OPG NEEDS TO DEMONSTRATE EFFORTS ALREADY TAKEN TO….

7. Reduce the likelihood of containment failure by pressure, temperature, radiation and fluid/gas interactions with containment penetrations given that certain reactor units have weak confinement structures and no pressurizable containments.
8. Evaluate and document the effect of recovery actions including power restoration, water injection as a function of time since onset of core damage
9. Install additional and independent of that available before Fukushima, instrumentation to detect and help control the progression of a severe core damage accident
10. Reduce likelihood of recovery actions exasperating the accident consequences by enhanced severe accident specific instrumentation and display of state of the reactor
11. Reduce likelihood of fuelling machine adversely affecting the outcome upon restoration of cooling functions
12. Modify Calandria vessel overpressure system to avoid fluid loss through rupture disks; delay onset of severe core damage

OPG NEEDS TO DEMONSTRATE EFFORTS ALREADY TAKEN TO…..

13. Modify moderator cooling system to install recovery system hookups for inventory replenishment and reinstatement of cooling functions
14. Investigate potential of in-situ design enhancements to avoid Calandria vessel failure by hot debris to avoid catastrophic failure of reactor structures
15. Increase the likelihood of successful external water injection by manual depressurization of the heat transport system
16. Increase the likelihood of core inventory degradation by ultra high pressure water addition to pressurized HTS before core degradation and prior to an in-core rupture
17. Increase the likelihood of reactor heat transport system heat removal by thermosyphoning by adding systems to remove non condensable gases that can degrade thermosyphoning
18. Reduce the likelihood of ECC injection failure

OPG NEEDS TO DEMONSTRATE EFFORTS ALREADY TAKEN TO…

19. Modify shield tank over pressure protection system to conform to anticipated heat loads to avoid catastrophic failure of shield tank vessel.
20. Install hookups for water addition to the shield tank
21. Obtain a more realistic evaluation of accident progression by using analytical methods that are more modern than the MAAP4-CANDU code that is 25 years old and obsolete in light of new information; and model the event with :
    - More detailed modelling of reactor core by differentiating between different bundles by modelling all reactor channels and incore devices
    - More appropriate modelling by using D_2O properties
    - More appropriate modelling by evaluating Deuterium (D_2) gas production, transport, recombination and burns. Has the utility considered that Deuterium gas properties differ greatly from hydrogen (H_2).
    - Considers oxidation of end fittings and feeders as sources of flammable D2 gas during a severe accident
    - Consider a more representative inventory of fission products
    - Consider concurrent fires (e.g. In feeder cabinets) as core voids, heats up and degrades
    - Consider failure of Calandria vessel at welds with hot debris
    - Consider failure of Calandria vessel penetrations at the bottom of the vessel (moderator outlet)
    - Consider explosive interaction of water with melt in Calandria vessel
    - Consider explosions caused by interaction of deuterium gas with PARS

OPG NEEDS TO DEMONSTRATE EFFORTS ALREADY TAKEN TO….

22. Consider alternate hydrogen mitigation measures as PARS may become ignition sources; consider upgraded catalyst plates with electrolytic deposition that limit gas temperatures.
23. Installation of measures to avoid ignition in existing PARS
24. Consider D_2 mitigation system optimization for a 100% Zircaloy oxidation (also to include effect of feeder oxidation)
25. Consider enhanced deuterium concentration monitoring systems within containment and Calandria vessel
26. Consider advanced video surveillance systems
27. Consider measures for mitigation of consequential fires during the progression of core disassembly
28. Consider post accident monitoring system instrumentation and control survival and functionality for severe accident conditions
29. Modify proposed emergency filtered containment venting for more realistic severe accident progression and fission product loads

OPG NEEDS TO DEMONSTRATE EFFORTS ALREADY TAKEN TO….

30. Consider improvements to pressure suppression system in reactor building as the vacuum building may be inadequate to avoid building failure for multi unit accidents
31. Consider reactor building reinforcements to avoid building failure; special emphasis on confinement on top of reactivity decks in multi unit station
32. Consider deploying on-site and off-site radiation detection equipment that actually detects the source characteristics and differentiates between incident radiation species by measuring the energy of incident radiation; does not get saturated by incident particulates as happened for Chernobyl at Leningrad station a thousand km away.
33. Develop methods and acquire instrumentation to help deduce source terms from radiation measurements so that prediction of radiation effects can be made for different locations and changing weather conditions
34. Develop simulators to train the operators in progression of a severe core damage accident and develop experimental basis & analysis to help avoid potential adverse outcomes of various mitigation measures.

MAIN CNSC ACTION ITEMS SUMMARIZED

1. Bleed condenser / degasser condenser relief capacity
2. Shield Tank relief capacity.
3. Means to protect containment integrity.
4. PARs installation
5. Potential hydrogen generation in the IFB area
6. Structural response of the IFB structure to high temperatures
7. Provisions for coolant makeup to PHTS, SGs, moderator
8. Assessments of equipment survivability under severe accident conditions.
9. Habitability of control facilities

ACTION ITEMS REQUIRED NO IMPLEMENTATION, MOSTLY JUST PAPER PLANS FOR 'ACTION'

MAIN CNSC ACTION ITEMS SUMMARIZED

10. Enhanced electrical power for key instrumentation and control (8 hour target)
11. Re-evaluation of external events using better analyses
12. Station specific SAMG
13. Modeling of severe accidents in multi-unit stations.
14. Upgraded analyses and experimental support
15. Evaluation of emergency plans and programs.
16. Backup power for emergency facilities and equipment.
17. Identification of external support and resources
18. Development of source term and dose modeling tools.

ACTION ITEMS REQUIRED NO IMPLEMENTATION, MOSTLY JUST PAPER PLANS FOR 'ACTION'

LET US ASK OPG – HOW MANY ACTUALLY IMPLEMENTED IN OPERATING REACTORS?

SUMMARY

1. NO COMPELLING ARGUMENTS TO ISSUE A 13 YEAR LICENCE EXTENSION

2. NEED TO FIRST UPDATE RISK EVALUATIONS

3. NEED TO FIRST UPGRADE REACTORS BEFORE REFURBISHMENT

4. NEED TO UPGRADE REGULATORY REVIEW FOR SEVERE ACCIDENTS