Determination of necessary plant instrumentation, equipment and materials

Joint IAEA-ICTP Essential Knowladge Workshop on
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Overview

- Determination of necessary plant instrumentation, equipment and materials
- Approach of evaluation of instrument availability
- Conclusions
- References

Background documents

- Plant specific background documents including EIP (Emergency Implementation Procedures):

  – Evaluate applicability of generic SAMG

  – Determine for each chosen CHLA or strategy:
    - Frontline SSCs
    - Alternative SSCs
    - Mobile or FLEX

Background documents

- Plant specific background documents including EIP (Emergency Implementation Procedures)

  – Define for all chosen SSCs necessary:
    - Support systems (e.g. water, AC/DC, fuel (EDG), HVAC/VA, boron, lubrication,...)
    - Accesability (harsh environment determination) if local actions are needed
    - Surveviability or potential negative impacts of environment on SSCs
    - Spare equipment (fire pipes, tools..)

  – Define organization and necessary human resource (ERO organization, TSC/ECR staff etc)

Background Documents - Strategies

Table 2.3.4: High priority SAM insights obtained from sensitivity and phenomenological evaluation

Phenomenology or Sensitivity Case	Insight	Severe Accident Strategy	Applicable SAMG
Sensitivity 1: Wet Cavity	NEK is the plant with the dry cavity design - Wet cavity would significantly reduce the releases (the percentage of the basement penetration would decrease from 12.3 % to 3.9 % and the percentage of no containment failure would increase from 33 % to 47%)	Make the cavity wet - allow water to enter the cavity.	If the appropriate change in the design will be done then in the SAMG SAG-4 (Inject into containment) there is the consideration of the flooding of the reactor cavity as discussed in Chapter 2.2 (SAG-4).
External vessel cooling	In case of NEK vessel is not flooded from outside - due to dry cavity design - the external flooding of the reactor vessel is recommended	Make the cavity wet - allow water to enter the cavity.	If the appropriate change in the design will be done then in the SAMG SAG-4 (Inject into containment) there is the consideration of the flooding of the reactor cavity as discussed in Chapter 2.2 (SAG-4).
Debris coolability	In low pressure vessel failure cases the debris will not be cooled by overlying water (no water in the cavity due to dry cavity design) and significant MCCI in the cavity is expected	Make the cavity wet - allow water to enter the cavity.	If the appropriate change in the design will be done then in the SAMG SAG-4 (Inject into containment) there is the consideration of the flooding of the reactor cavity as discussed in Chapter 2.2 (SAG-4). SAMG SAG-3 (Control hydrogen flammability) and SAG-7 (Reduce containment hydrogen) also address this issue.

Table 2.3.2: SAM Strategies Obtained from Krško Release Category

RC no.	Release Category Definition	Severe Accident Strategy	Applicable SAMG
1	Core recovered in-vessel, no containment failure	flood the outside of the reactor vessel and thereby prevent vessel failure	SAG-4
		depressurise the primary system to allow low head injection source	SAG-2
		continue core cooling by injection into the primary system and other means	SAG-3
		avoid depressurisation of the primary system	SAG-2
2	No containment failure	continue injection of water via low head SI to quench and cover debris after vessel failure	SAG-2
		continue containment heat removal to prevent containment over pressure failure	SAG-6
		maintain the core in a controlled stable state	SAG-3
3A	Late (time frame IV) containment failure, no MCCI	establish containment heat sinks	SAG-6
		prevent hydrogen burns after 24 hours	SCG-3

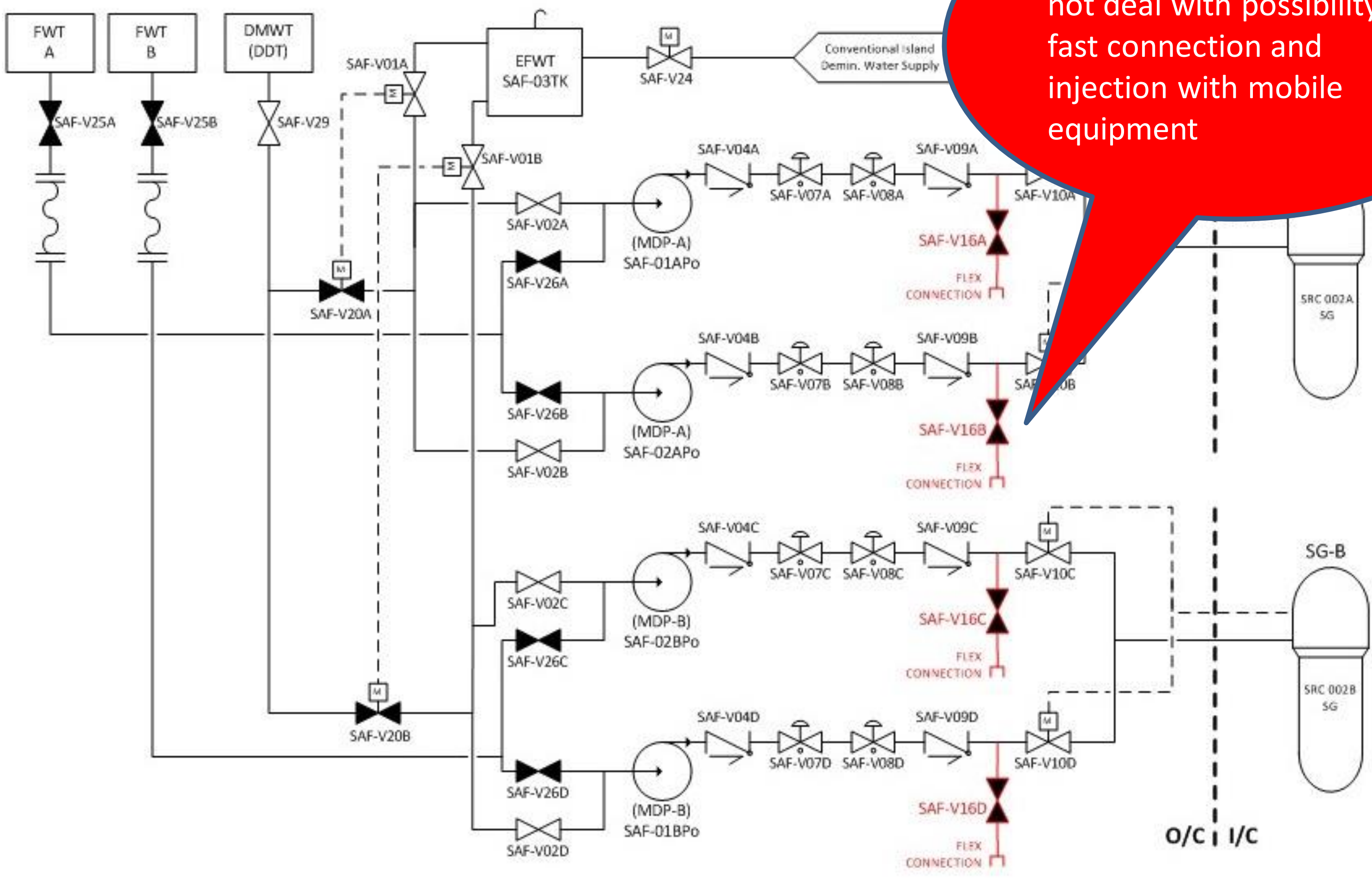
Background Documents - Strategies

COMPONENT NAME	TAG NUMBER	COMPONENT CHARASTERISTICS (Nominal flow, shutoff head, etc)	SUPPORT SYSTEMS			
			Instrument air	Cooling	AC BUS/MCC	DC BUS/BRKR
PUMPS						
Motor driven auxiliary pump 1A, 1B	AF102PMP-01A AF102PMP-02B	Rated capacity 84.14 m3/h at 104.9 kp/cm2 (1022.3m); Shutoff head 129.5 kp/cm2 (1264.9m); required NPSH 5.8m	just for AF control valves	CC train A and B	EE105SWGMD1/3 EE105SWGMD2/3	DC101PNLK101/4 DC101PNLK301/4
Turbine driven auxiliary pump 1C	AF101PMP-03C	Rated capacity 184 m3/h at 106.2 kp/cm2 (1035.7m); Shutoff head 127.8 kp/cm2 (1249m); required NPSH 6.1m	just for AF control valves	N/A	N/A (steam pressure must be greater than 5 kp/cm for pump operation)	N/A
Main feedwater pumps (1A, 2B, 3A(B)-powered from M1 or M2 bus)	FW 105 PMP 001 FW 105 PMP 002 FW 105 PMP 003	Rated capacity 2339.6 m3/h at 65.9 kp/cm2 (642.5m); Shutoff head 78.8 kp/cm2 (768 m); required NPSH 33.5m	just for MFW control valves	N/A	EE105SWGM1/6 EE105SWGM2/9 EE105SWGM1/7 or EE105SWGM2/8	DC101PNLG701/17 DC101PNLG701/2 DC101PNLG710/17 DC101PNLG710/2
Condensate pumps	CY 100 PMP 001 CY 100 PMP 002 CY 100 PMP 003	Rated capacity 1362 m3/h at 28.6 kp/cm2 (279 m); Shutoff head 33.5 kp/cm2 (326m); required NPSH 1.1m	N/A	N/A	EE105SWGM1/10 EE105SWGM2/5 EE105SWGM2/6	DC101PNLG701/1 DC101PNLG701/18 DC101PNLG701/18
Condensate transfer pump	CY 110 PMP	Rated capacity 37.5 m3/h at 6.7 kp/cm2 (65.5m); shutoff head 8.11kp/cm2; required NPSH 2.13m	N/A	N/A	EE103MCC111/6C	N/A
Demineralized water transfer pumps(2)	WT114PMP001 WT114PMP002	57 m3/h each at 6.1 kp/cm2	N/A	N/A	EE103MCC111/7A EE103MCC212/10E	N/A

Procedures - Attachments

- Example: Inject to SGs

Generic WOG SAMG does not deal with possibility of fast connection and injection with mobile equipment



Reference: PC-1-17SAF-501-S030-45-1
(Sheet 1 of 2)

Background- Intrumentation

- Bases for instrumentation used in generic SAMGs are summarised in NUREG-5691 (1991) where U.S. Nuclear Regulatory Commission (NRC) has identified accident management as an essential element of the Integration Plan for the closure of severe accident issues.
- One of the areas affecting the capability of plant personnel to successfully manage a severe accident is the availability of timely and accurate information that will assist in determining the status of the plant, selecting preventative or mitigative actions, and monitoring the effectiveness of these actions.

Not pretty new!
Today, lot of EPRI, IAEA, OCD, documents exist

Background

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Approach of evaluation of instrument availability

- **5 steps:**
  1. **Identify a set of possible severe accident sequences** that have the potential of influencing the risk for a PWR with a large dry containment.

  2. **Define the expected conditions** within the **reactor coolant system** and **containment** for important accident sequences, and identify phases of the sequences that correspond with the phenomena occurring and challenges to different instruments.

  3. **Assess instrument availability during each phase** of the severe accident sequences, based on the location of the instrument and conditions that would influence instrument performance.

Approach of evaluation of instrument availability

4. **Provide** an accident management information assessment **discussing the information needs** and the **instruments that are available**. Identify potential limitations on the information available for assessing the status of plant safety functions.

5. **Define envelopes bounding the range of parameters** that would be expected to impact instrument performance for the severe accidents identified in Step 1.

Possible severe accident sequences

- To accomplish Step 1, the **types of severe accident sequences that have the potential of influencing risk were identified** (e.g. generic SAMGs were based on the probabilistic risk assessment results presented in NUREG-1150 for the Surry and Zion PWRs.
  - These results were used in NUREG-5691 because they represent the most recent evaluation of all credible types of accidents that will dominate core damage frequency and risk to the public.
  - Although the results are specific to these two plants, the sequence categories identified in this document are sufficiently broad that they would apply to most PWRs.)
- However, the plant specific evaluation is highly recommended and necessary!

Possible severe accident sequences

Accident sequences:

- **Phase 1** - This phase begins with initiation of the sequence including the blowdown/boiloff of water inventory in the reactor coolant system and ends at the time of initial uncovery of the reactor core. Operator guidance for Phase 1 is included in the existing plant Emergency Operating Procedures.

- **Phase 2** - Core uncovery begins during this phase. Fuel heatup results from the lack of adequate cooling. This phase ends when fuel melting begins.

- **Phase 3** - Fuel melting occurs during this phase. Fuel and cladding relocation and the formation of debris beds occur. The phase ends when relocation of a significant amount of core material to the reactor vessel lower plenum begins.

Accident sequences (cont):

- **Phase 4 -** Molten core debris accumulates in the lower head of the reactor vessel during this phase. The phase ends with the failure of the lower head.

- **Phase 5** - This phase is initiated when the core debris directly interacts with the containment after lower head failure. During this phase, containment failure could occur because of overpressure, hydrogen burns, or basemat meltthrough resulting from core-concrete interaction. Containment failure due to direct containment heating is also possible, depending on the reactor coolant system pressure when lower head failure occurred.

Possible severe accident sequences

Accident sequences (cont):

- Separation of the sequences into five phases allows for segregation of the information needs and instrument availability.
- **Information needs** and **instrument availability differ from phase to phase**, as **different plant safety functions** are challenged and **harsh environmental conditions** develop in various portions of the reactor coolant system, containment, and, in some sequences, the auxiliary and turbine buildings.
- Instrument availability evaluations were based primarily on the pressure and temperature qualification,location, and source of backup power for each instrument.

Severe accident conditions

- To accomplish Step 2, the conditions within the reactor coolant system and in containment are defined, based on a review of severe accident analyses available for PWR plants.

- The **BMI-2104** and **NUREG/CR-4624** analyses were used in the development of **generic SAMGs** because most of the important events expected during a severe accident, from core melt through lower head failure and beyond, are found in these reports, including possible containment failure modes. These analyses provide a baseline for gaining insight into challenges to instrument availability.
  - However, it is recognized that natural circulation is not considered in BMI-2104 and NUREG/CR-4624, which can impact performance of instruments in the reactor coolant system.

- Still, the plant specific evaluation is highly recommended and necessary!

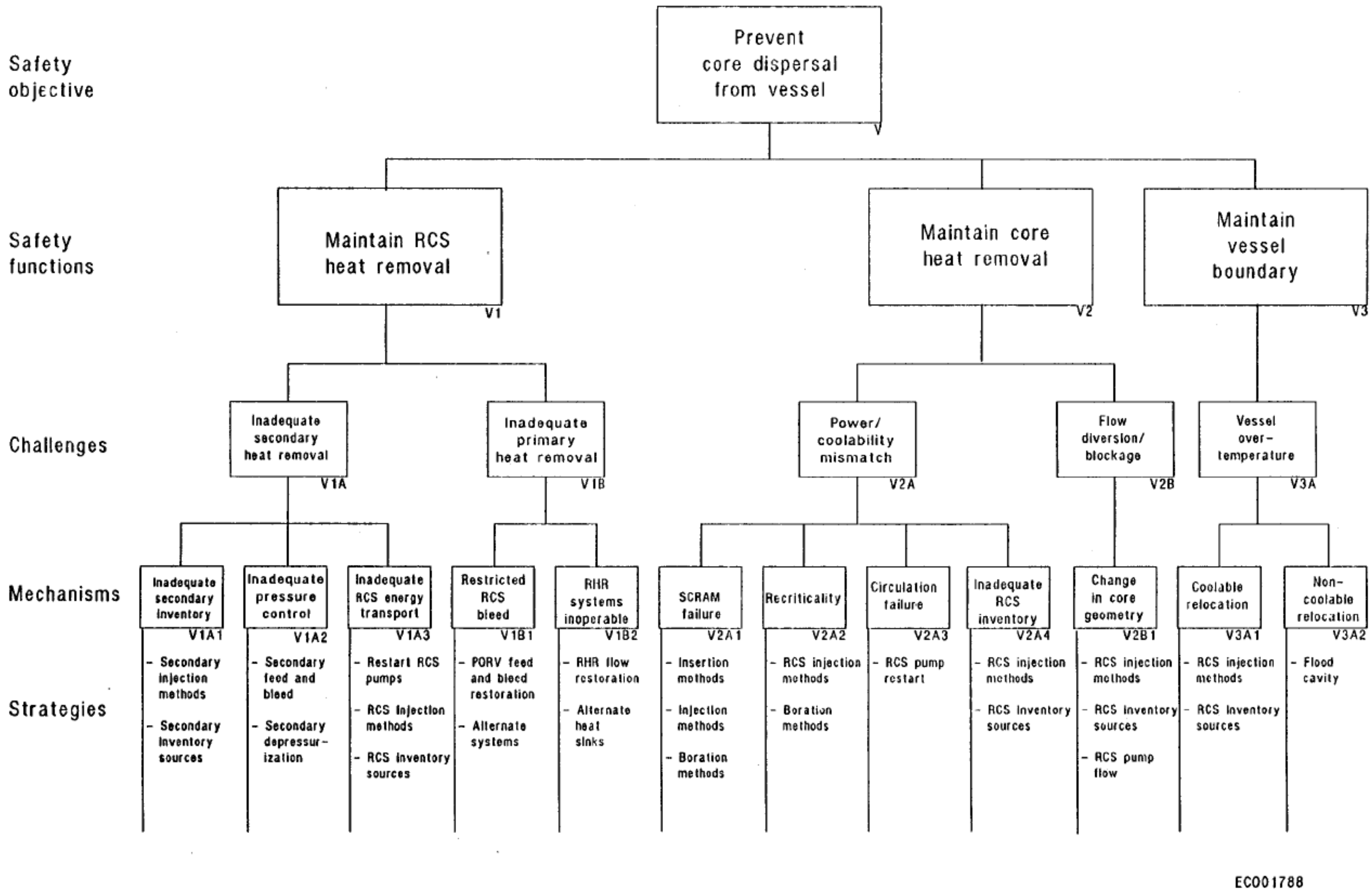
Accident management information assessment

The Safety Functions information needs to be identified for each mechanism are summarized as follows:

- Determination of the status of the safety function in the plant, that is, whether the safety functions are being adequately maintained within predetermined limits.
- Identification of plant behaviour (mechanisms) or precursors to this behaviour that indicate that a challenge to plant safety is occurring or is imminent.
- Selection of strategies that will prevent or mitigate plant behavior that is challenging plant safety.
- Monitoring the implementation and effectiveness of the selected strategy.

Accident management information assessment

Generic SAMGs accident management information assessment relies principally on the safety objective trees (e.g. prevent core dispersal from vessel, prevent containment failure and mitigate fission product release from containment) and information needs tables developed in NUREG/CR-5513



Accident management information assessment

APiS

PWR - CHLA Information Needs (Ref: TBR)

CHLA	Measures of Effectiveness (Information Needs)
Inject into RCS	Decreasing Core exit thermocouples Increasing Reactor vessel level indication Decreasing containment pressure Decreasing containment temperatures Decreasing hot or cold leg temperature
Depressurize RCS	Decreasing RCS pressure Decreasing Core exit thermocouples Increasing containment hydrogen concentration Increasing containment radiation Increasing containment pressure Decreasing hot or cold leg temperature

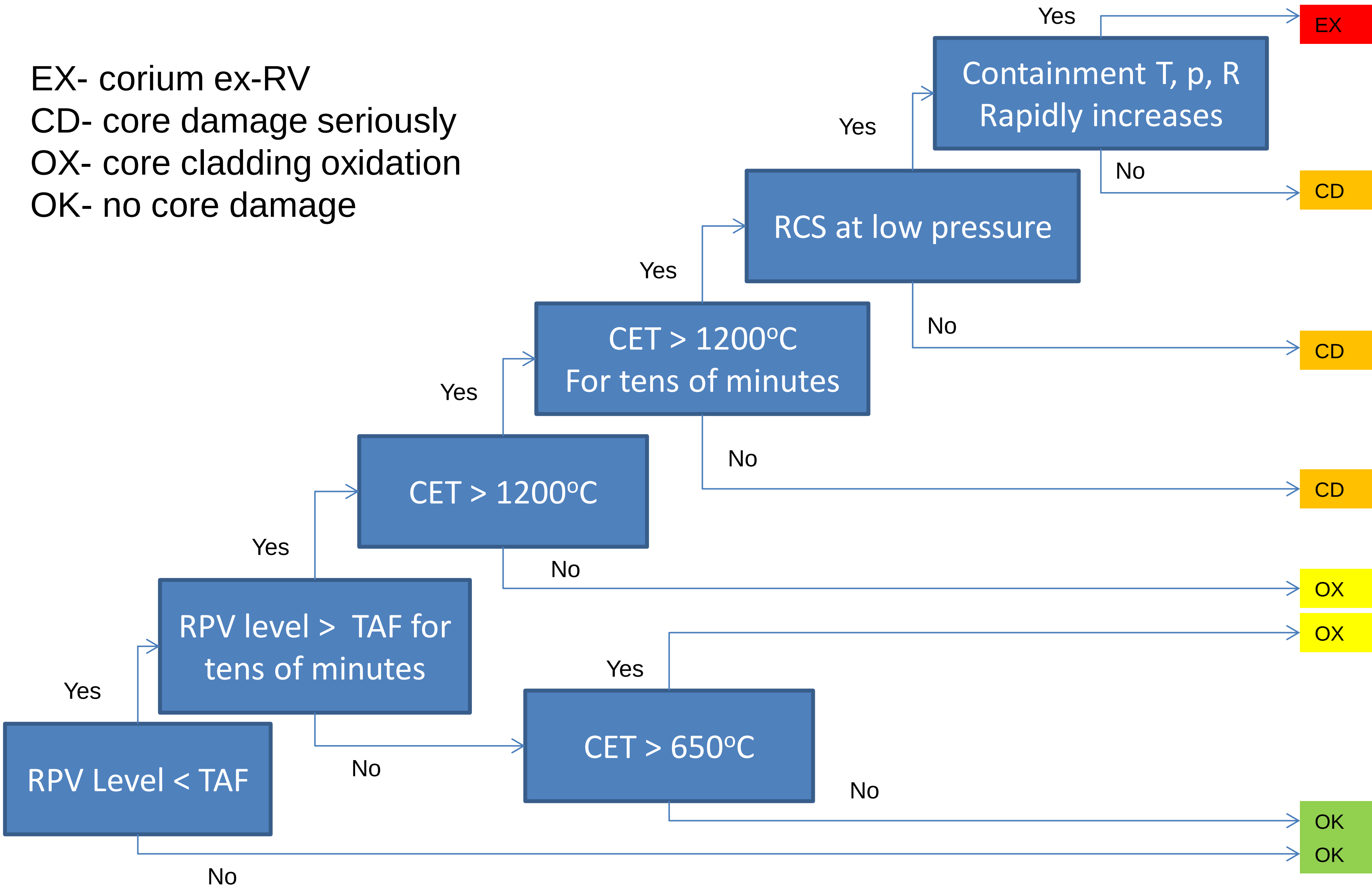
DIAGNOSTIC FLOW CHART/SEVERE CHALLENGE STATUS TREE INSTRUMENTATION

Parameter	Instrumentation	Number of Channels	Instrument I.D. or TAG Number	Instrument Range
RCS Pressure	Wide Range RCS Pressure	1/loop (hot leg)	PT405 [PT406[Note 1]]	0-210kp/cm^2
	Pressurizer Pressure	4	PT455, PT456, PT457, PT458	120-180kp/cm^2
	Accumulator Pressure	2/accumulator	PT960, PT961 [PT962, PT963]	0 - 56 kp/cm^2
	SI Header Pressure			
	Charging Pumps	1/pump	MDP01: PI287B MDP02: PI288B TDP: PI286B PI204 (header)	0 - 250 kp/cm^2
	SI Pumps	1/pump	PI904 [PI905]	0 - 210 kp/cm^2
	RHR Pumps	1/pump	PI861 [PI862]	0 - 50 kp/cm^2

Instrument	Typical Range	Usefulness	Comments
Pressurizer Pressure	1700-2500 psig	Limited in usefulness during core damage.	Lower indication range (1700 psig) limits usefulness during severe accidents.

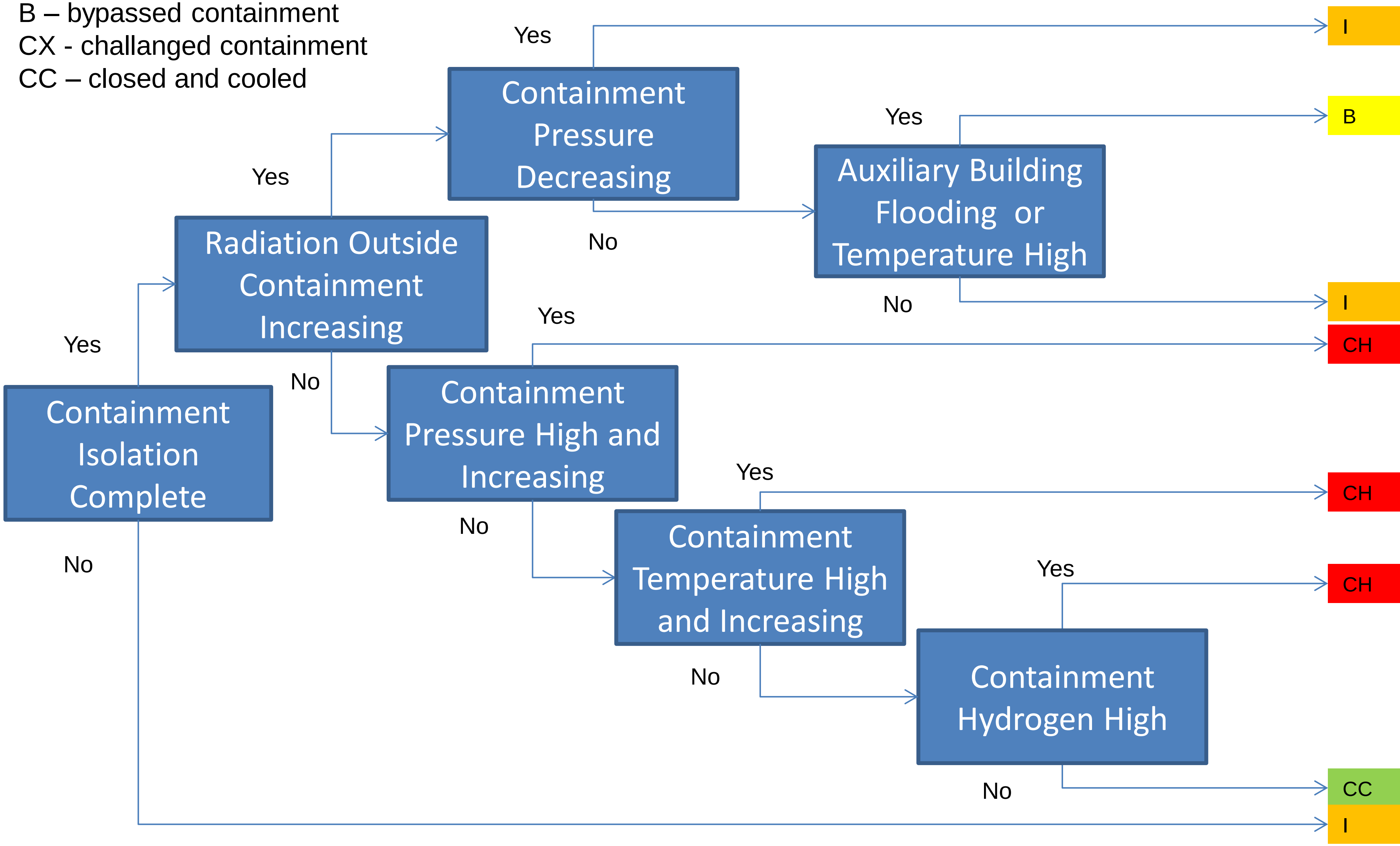
Core Damage Condition Status Tree example

EX- corium ex-RV
CD- core damage seriously
OX- core cladding oxidation
OK- no core damage



Containment Condition Status Tree example

I – impaired containment
B – bypassed containment
CX - challanged containment
CC – closed and cooled



Instrument availability during severe accidents

- The conditions affecting instrument availability are:

  – **Harsh pressure, temperature, humidity and radiation containment environments**, causing instrument performance to degrade.

  – **Electrical power failure** resulting from station blackout, loss of a dc bus, or other power interruptions, causing instruments to be unavailable.

  – **High radiation fields** resulting from an interfacing system LOCA or steam generator tube rupture, impeding access to instruments or sampling stations located in the auxiliary building or turbine building.

Instrument qualification assessment

- **Instrument information should be based on the Regulatory Guide 1.97**
- Typical instrument systems consist of transducers, cabling, electronics, and other instrument system components.
  - For instruments located in the reactor coolant system, evaluation is focused on the sensors, because of the harsh temperature conditions that these sensors could be exposed to during a severe accident.
  - For instruments located in the containment, consideration is given to cabling, splices, and other components of the instrument systems.

Instrument qualification assessment

- The basic instrument system performance is not well known when qualification conditions are exceeded!
  - An assessment of the relationship between the instrument uncertainties and the timing and degree to which the qualification conditions are exceeded would require a detailed study of basic instrument capabilities and failure modes.
  - It should be noted that operators may not recognize that instrument performance has degraded. One possibility is that an instrument reading appears to be normal or the trends may be plausible, when, in actuality, the plant conditions and trends are different.
  - Cabling is expected to be particularly vulnerable to the high-temperature conditions that develop during multiple hydrogen burns.

Instrument qualification assessment

- Envelope of severe accident plant conditions and event timing – the thermal hydraulic and timing data (e.g. MAAP, MELCORE, RELAP/SCADAP calculation) are intended to provide an indication of the conditions to be expected for a broad range of severe accidents
  - Envelope definition is defined as an upper limit that covers the expected pressure and temperature (and humidity/radiation) for each accident phase for any sequence.
  - Envelope Uncertainty: There are three aspects to the uncertainty of analytical predictions of severe accident conditions that affect instrument availability: (1) the occurrence of a severe accident event, such as lower head failure or hydrogen burns, which causes instrument failure; (2) timing of major severe accident events; and (3) predicted pressure and temperature (and humidity/radiation) conditions at various locations in the plant.

Instrument qualification assessment

Instrument Survivability:
- Inside process
- Harsh Environment!

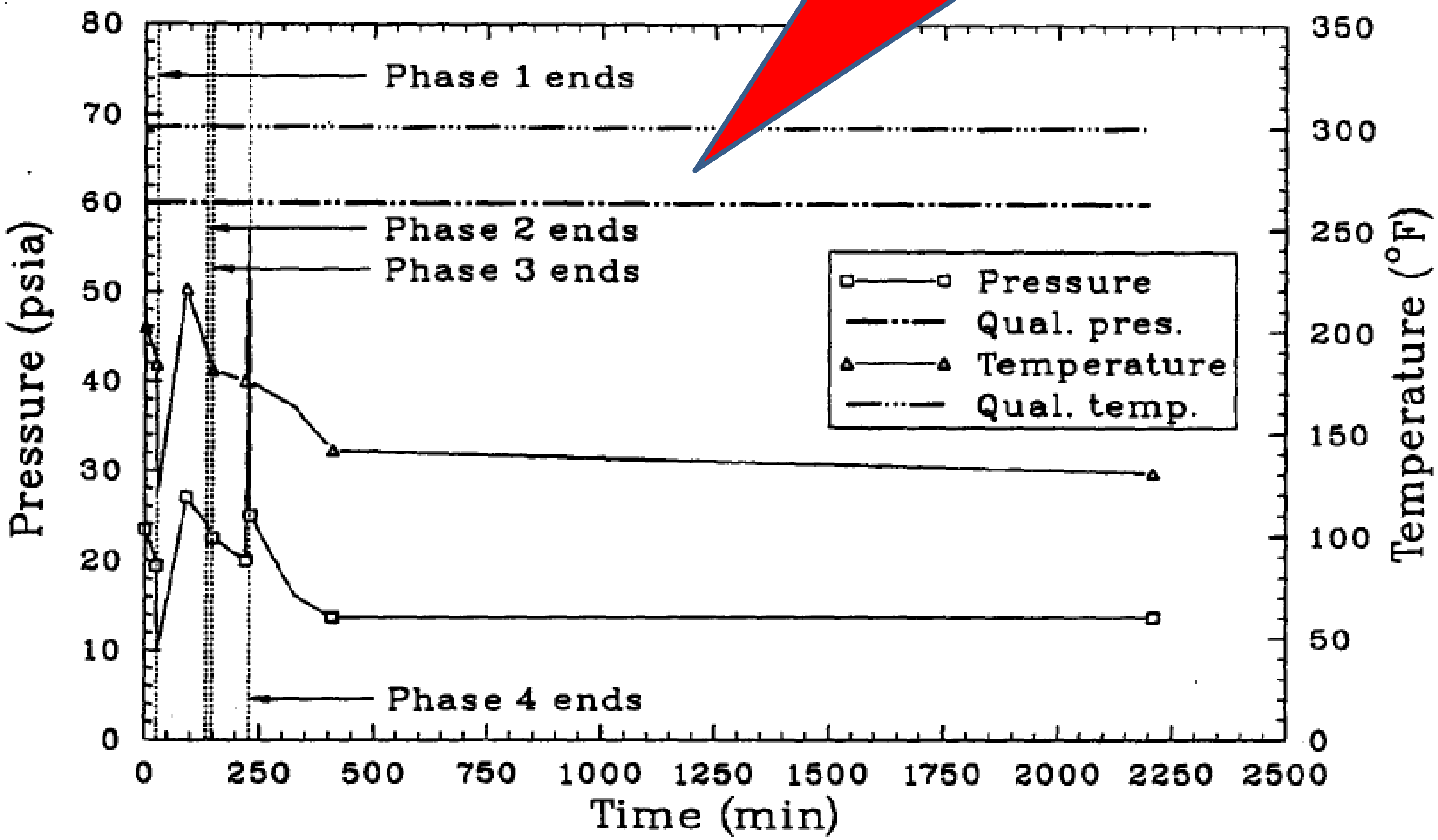


Figure B-4. Surry S_2D, basemat melt, containment data.

Instrument qualification assessment

- There is little uncertainty in the conclusion of degraded performance or failure of instruments located:
  - in the reactor vessel if exposed to the temperatures expected during a core melt, which are well in excess of the qualification temperatures.
  - in the reactor cavity which would be subjected to temperature conditions well in excess of their qualification limit upon lower head failure.

- There is more uncertainty in assessing the performance of instruments located in the reactor coolant system outside the reactor vessel, because of hot gases being transported through the reactor coolant system due to PORV actuation or natural circulation. The uncertainty here is in the temperature predictions in the reactor coolant system, which are sensitive to the analytical assumptions made.

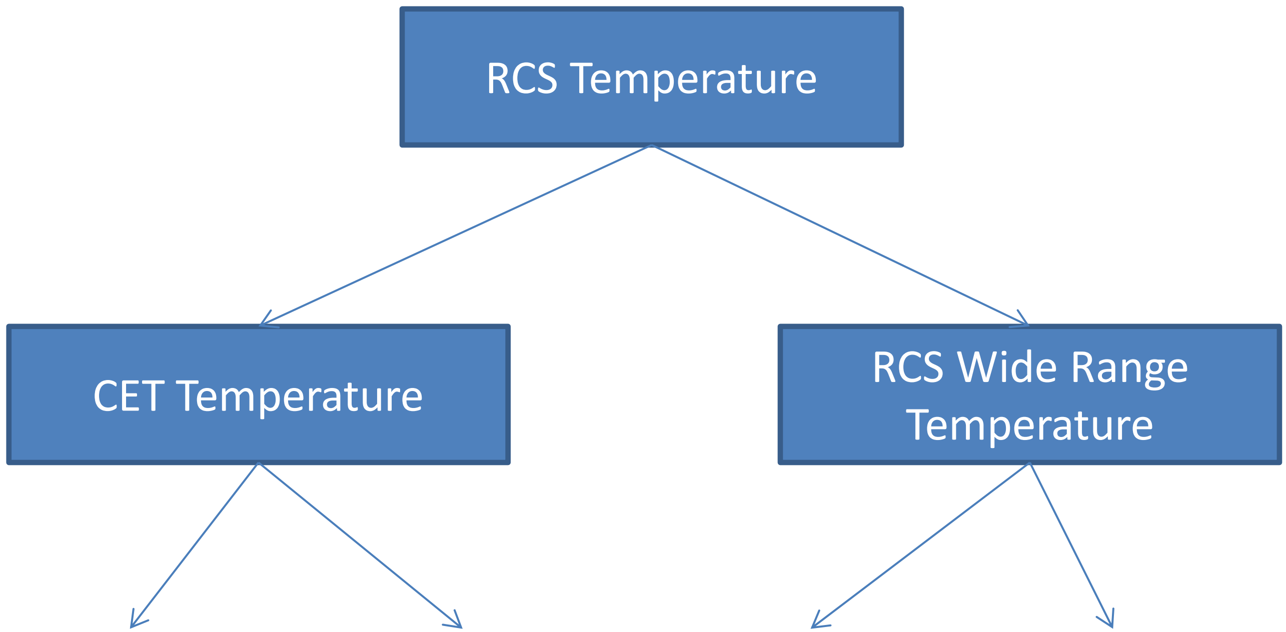
Instrument qualification assessment

- The occurrence and timing of **hydrogen burns** or **direct containment heating** can produce temperatures well in excess of qualification limits of instruments located in the containment.
  - However, the analytical uncertainty has a greater impact because of the dynamics of hydrogen transport and ignition in containment.
    - The uncertainty issue regarding hydrogen burns in the containment is the location and magnitude of these burns.
    - If hydrogen bums occur near the top of the containment, instruments located in the reactor cavity or near the containment floor may survive because of dissipation of the thermal energy.
    - The occurrence of hydrogen bums in the containment does not automatically mean that the performance of instruments located in the containment will degrade. The issues are similar for direct containment heating.
- Evaluation of instrument performance during hydrogen burns or direct containment heating should be evaluated on a plant specific basis.

Diagnostic Assessment

Necessary Information

Parameters

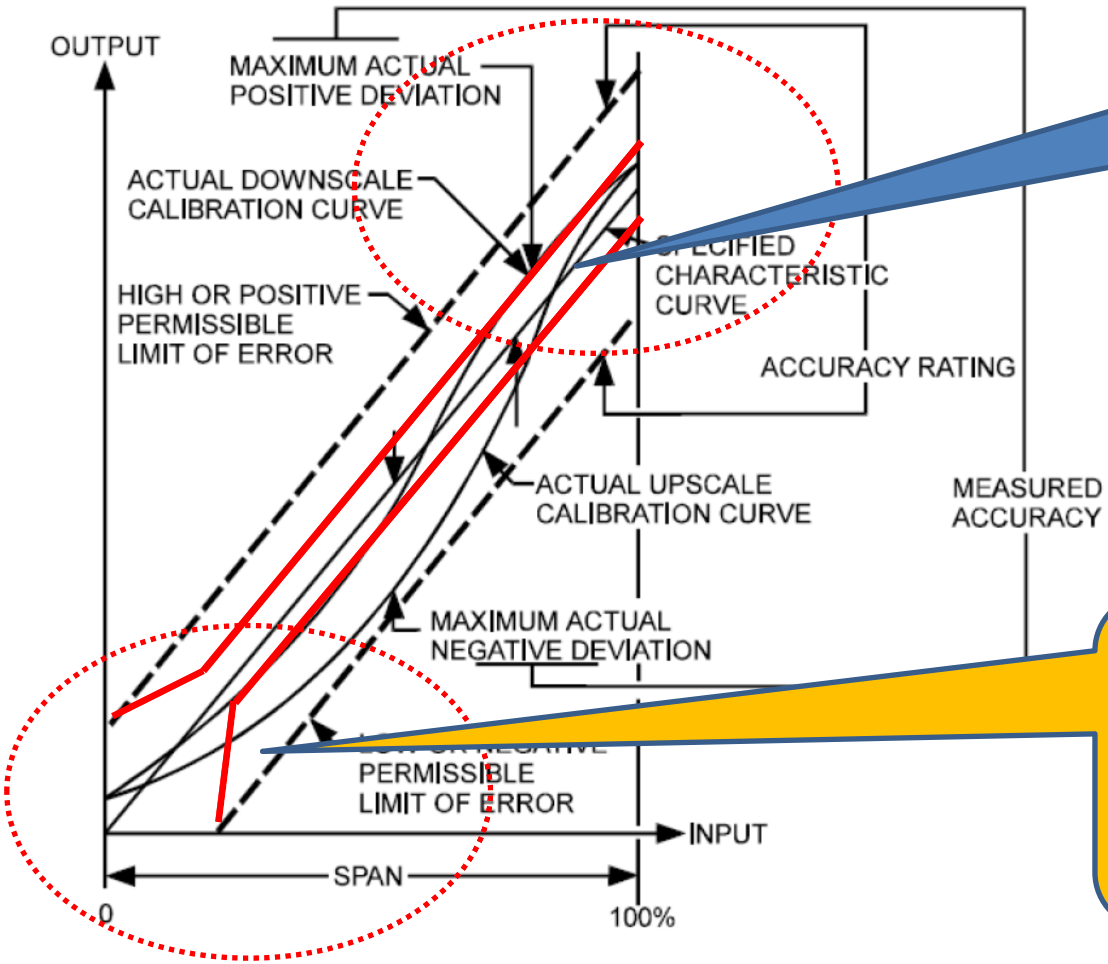


Instrument Loops	**TCAxxx**	**TCBxxx**	**TRCAxxx**	**TRCBxxx**
Loop Components	TLAyyyy	TLByyyy	TLAzzz	TLBzzz
	TQAyyy	TQByyy	TQAzzz	TQBzzz
	TIAyyy	TIByyy	TIAzzz	TIBzzz

Diagnostic Assessment

Sensor reference accuracy

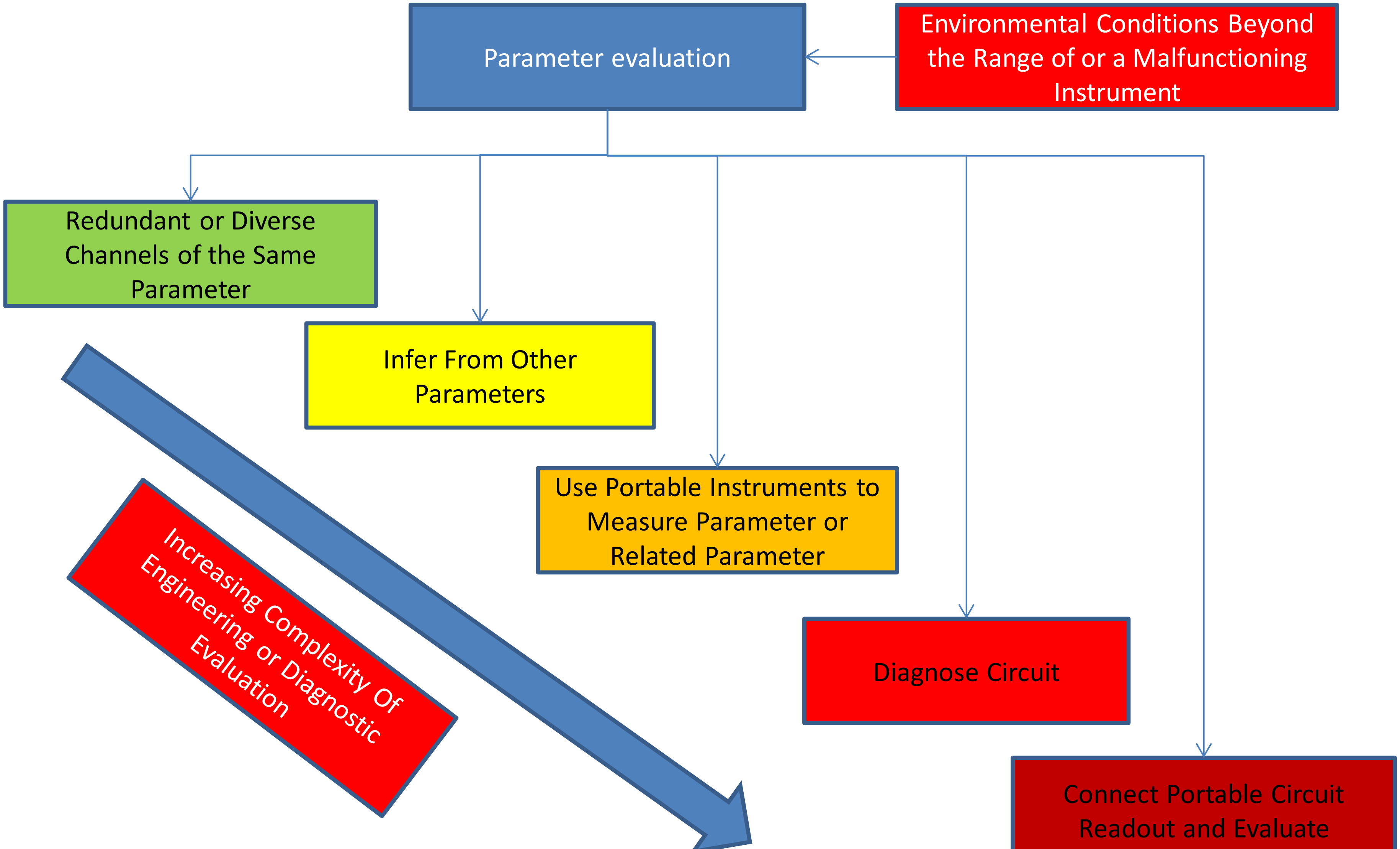
"In process instrumentation, a number or quantity that defines a limit that *errors* will not exceed when a *device* is used under specified *operating conditions*"



Typically, uncertainty for parameters during normal operation is low

SAMGs often use setpoints where uncertainty is bigger and affected by harsh environment conditions!

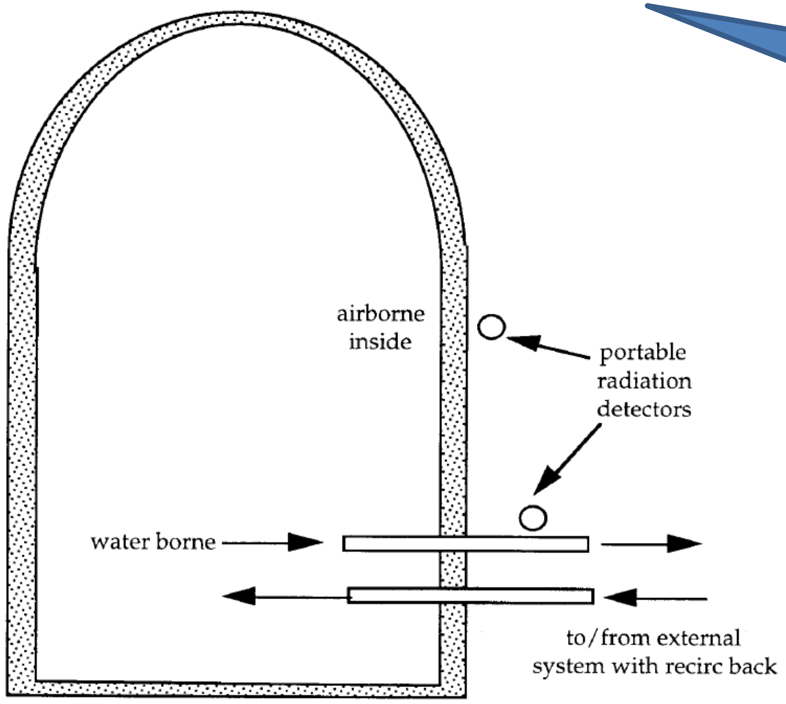
Diagnostic assessment

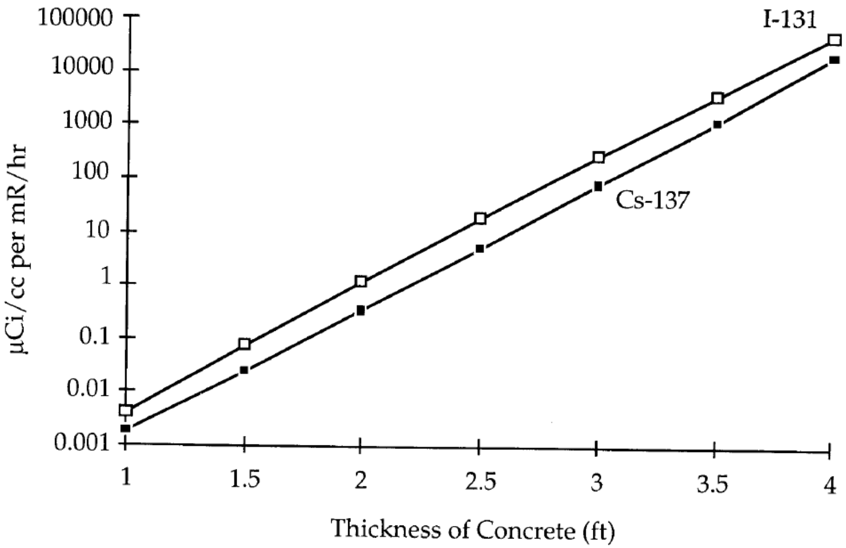


Diagnostic assessment

Example: Portable Radiation Detection of Containment Internal Radiation and Necessary Correction due to the Thickness of Concrete

Necessity for Technical Support Centre (TSC) and Operational Support Centre (OPC) training and drills!





Conclusion

- The **role of plant instrumentation is significant** and has to be carefully evaluated in the process of the development of the SAMGs.

- The plant instrumentation provides the **vital link** between:
  - the **severe accident conditions inside the plant** and
  - the **decision making process for severe accident management activities**.

- Because the correct **use and interpretation of instrumentation** is fundamental to the **successful diagnosis and management** of a severe accident, instrumentation should be an **integral part of severe accident training**.

References

[1] NUREG/CR-5691, "Instrumentation Availability for a Pressurized Water Reactor With a Large Dry Containment During Severe Accidents," March 1991.

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[3] EPRI TBR, "Severe Accident Management Guidance Technical basis Report, Volume 1,”September 1992.

[4] "NPP Krško Severe Accident Management Guidelines Implementation”; paper presented at the international conference “Nuclear Option in Countries with Small and Medium Electricity Grids 2002"; Dubrovnik, Croatia, June 17-20,2002., I. Basic, J. Spiler, B. Krajnc, T. Bilic-Zabric (NEK);