Development of Accident Management Procedures and Guidelines

Joint IAEA-ICTP Essential Knowladge Workshop on Nuclear Power Plant Design Safety – Updated IAEA Safety Standards 9-20 October 2017

Presented by

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APoSS d.o.o.

Overview

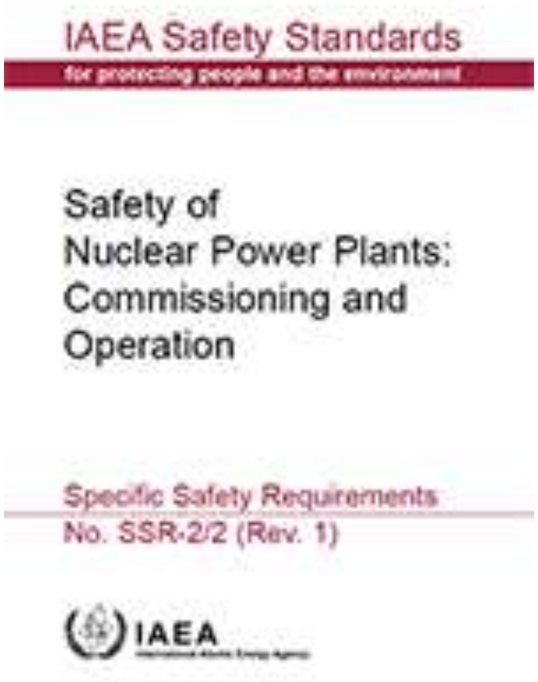
- Introduction
- Experience with Krsko AMP
  - WOG Generic SAMG Implementation
  - Plant specific SAMG
  - IPE Background
  - Background Documents – Strategies/Setpoints
  - Procedures
  - Conclusions
- References

AMP in IAEA Standards

APoS

IAEA SSR-2/2, rev.1, Req.#19 Accident Management Programme (para 5.8-5.9)

- The operating organization shall establish, and shall periodically review and as necessary revise, an accident management programme.



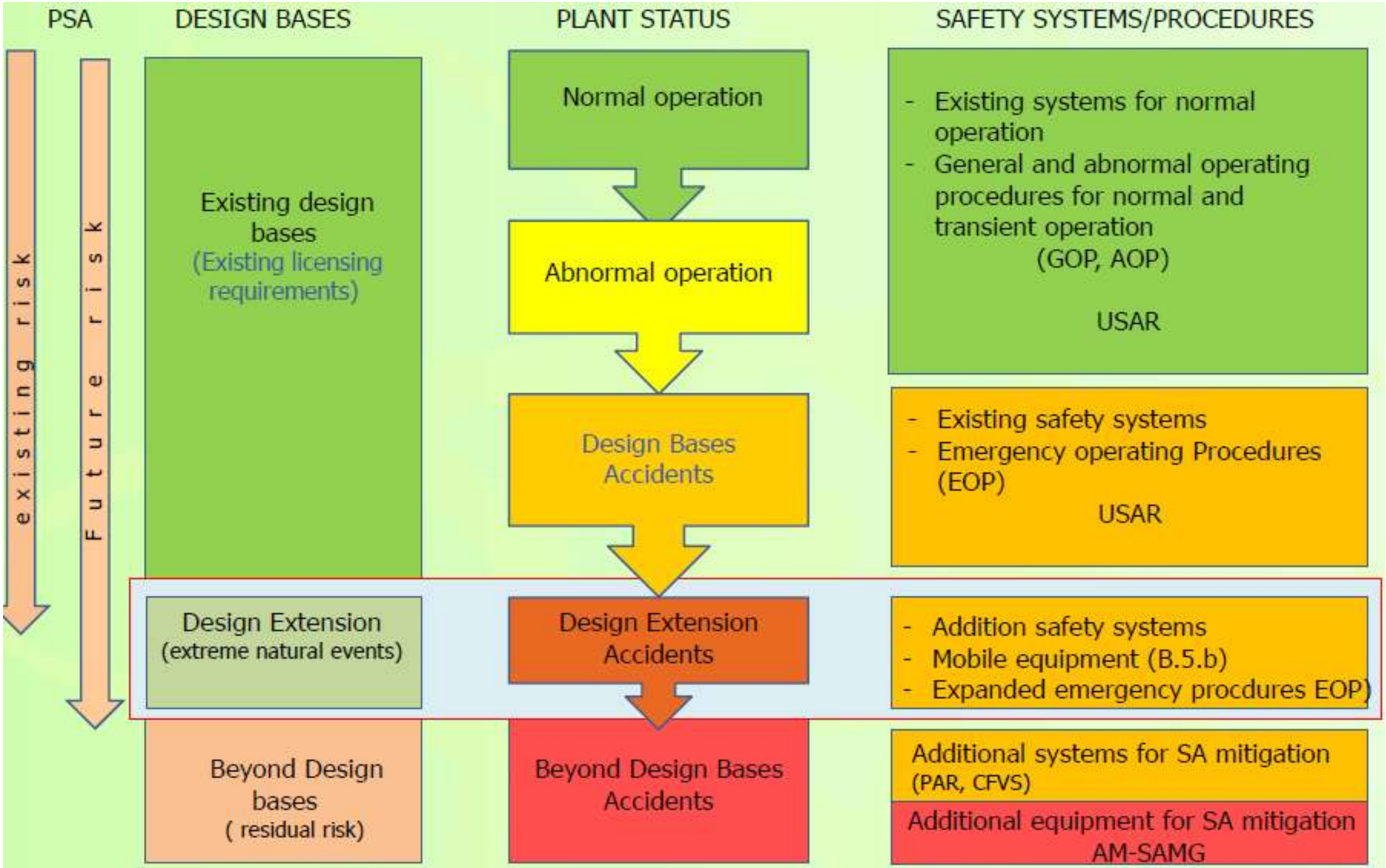
- **IAEA SSR-2/1, rev.1, para#2.10: „.. the establishment of accident
- management procedures..”

Fission Products Barrier

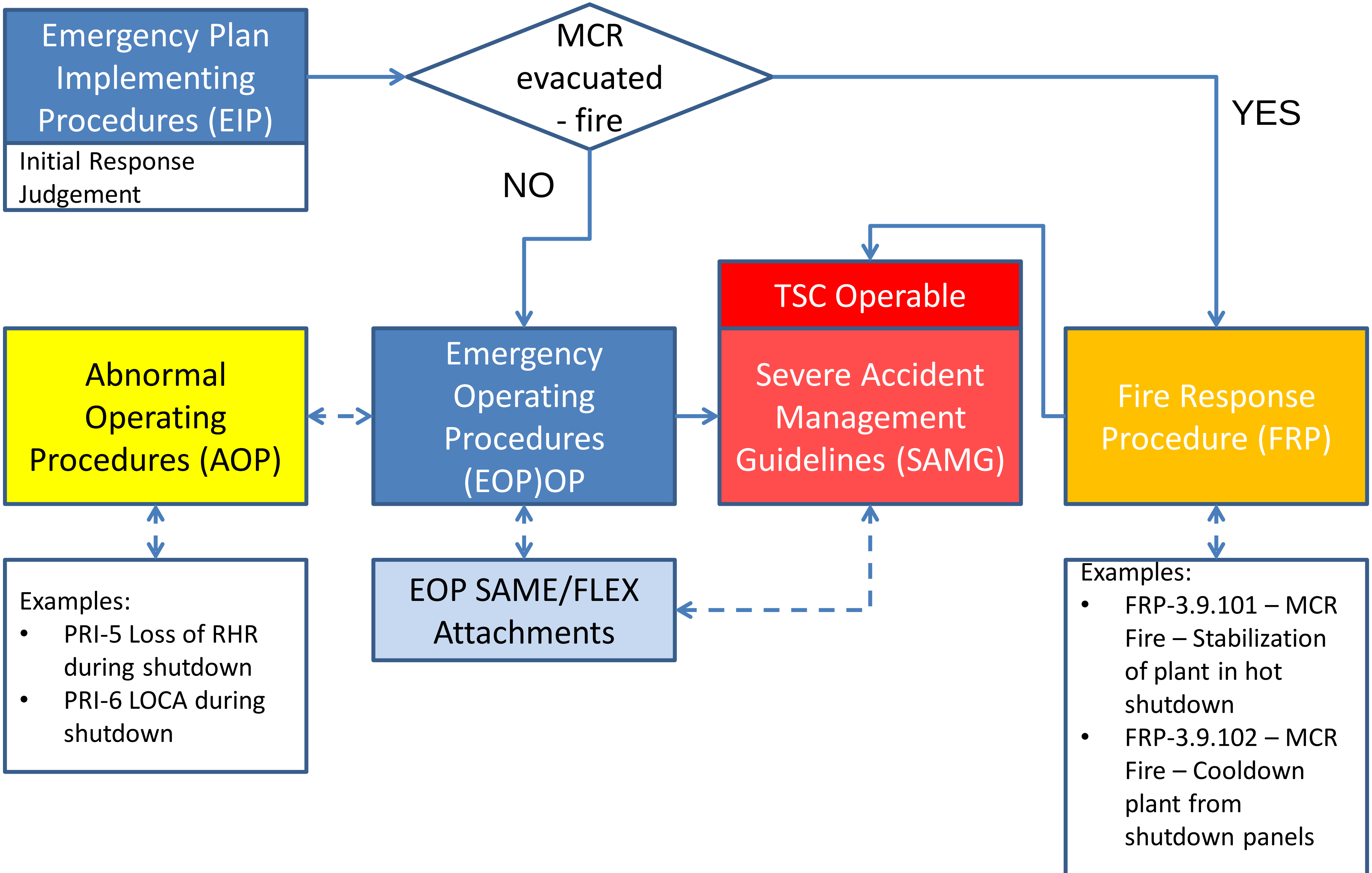
- For AM development, it is important to understand the challenges to Fission Product (FP) barriers
- Mitigating strategies may compete for resources, therefore, it is important to establish priorities

An understanding of severe accident phenomena is critical to AM

Concept of Krsko AMP



Krsko NPP Response to NEI 06-12 (B5b)



WOG Generic SAMG Implementation

- Review of WOG Generic SAMG applicability;
  - **Development of plant-specific strategies**
  - **Development of plant-specific SAMG setpoint;**
  - **Development of plant-specific computational aids;**
- Review of NEK EOPs to incorporate transitions to SAMG;
- Writing of plant-specific control room SACRGs;
- Writing of plant-specific TSC guidance, including SAGs, SCGs, DFC, SCST, and SAEGs;

candidate high level actions (CHLA) strategies and mitigate system/structure/component (SSCs) (based on OECD, IAEA and EPRI Severe Accident Management Guidance Technical Basis Reports (TBR) in comparison with NPP design, available SSCs and its applicability – NOT DIRECTLY APPLICABLE!!!

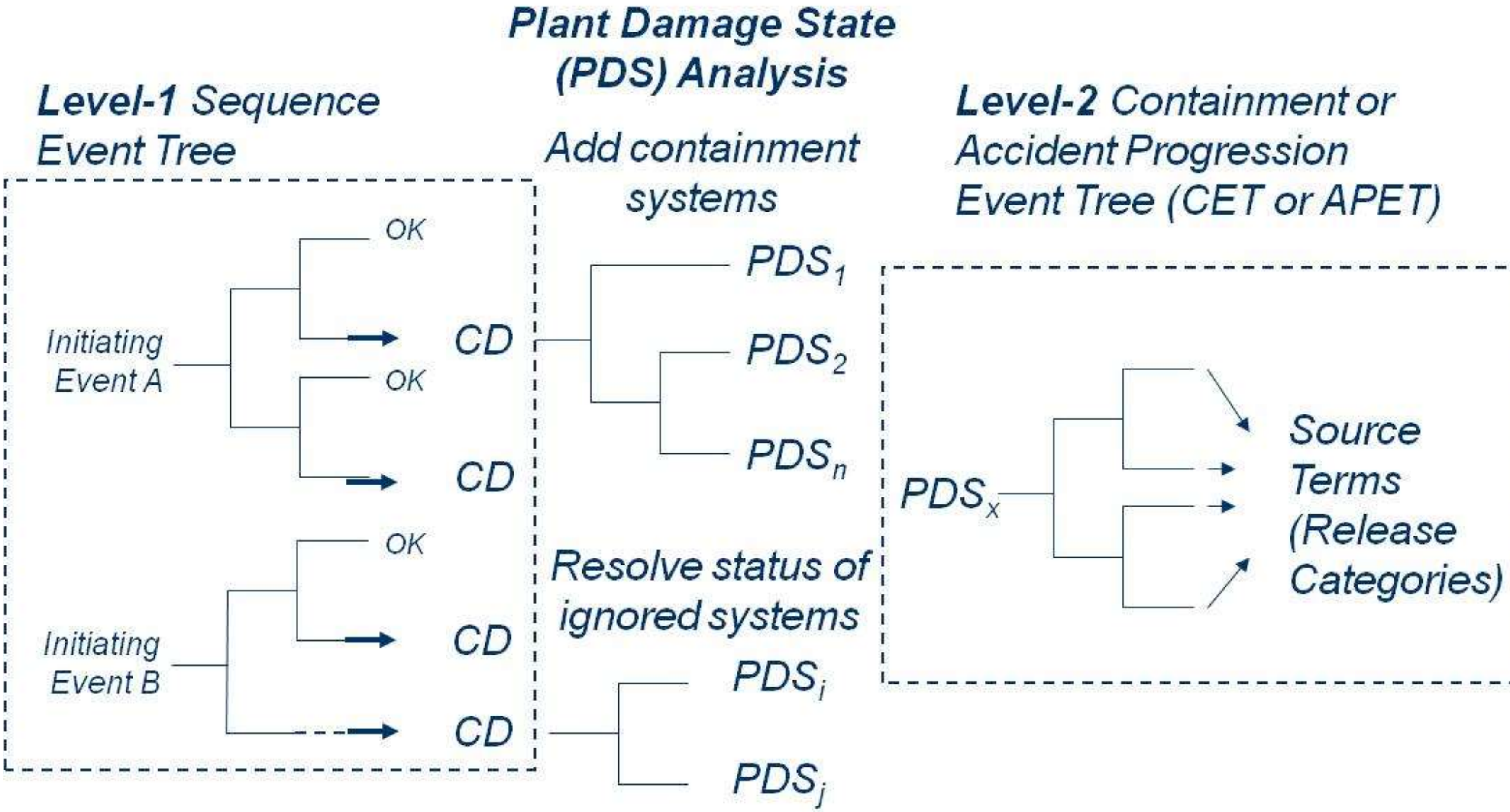
Additional Plant Specific Issues

- Definition of transition (rules of usage)
- SAMG for MCR (should be similar to FR-C1)
- SAMG for Spent Fuel Pool (not available in generic SAMG, important issue from Fukushima point of view)
- SAMG for shutdown (e.g. loss of SRH on midloop operation)
- Alternative means (mobile equipment) usage:
  - Different fire protection pumps
  - Fast connections to the systems (e.g. injection into SGs)
  - Source of waters (e.g. amount for flooding the containment to protect cavity floor from MCCI OR even flooding the Rx cavity to the top of acctive fuel to establish external cooling)

PSA Level 1 and 2

- Plant specific analysis (IPE – Individual Plant Examination) - plant response on Severe accident
  - PSA Level 1:
    - Event Trees and Fault Tree,
    - Core Damage State Evaluation
  - PSA Level 2
    - Containment Event Trees (PDS evaluation)
    - Deterministic analysis capability to simulate severe accidents (MAAP, MELCOR,..

Link Level 1 Results to Level 2



Relationship between IPE and SAMG

Plant-specific Severe Accident Management insights were developed based on the following:

IPE – Individual Plant Examination

Dominant core damage sequences from Level 1 study have been grouped and assessed following the criteria set out in NUMARC 91-04, Severe Accident Issue Closure Guideline

↓

For beyond 24 hour sequence (loss of SW, loss of CCW, station blackout), insights were developed based on the accident scenarios

↓

The Level 2 results have been grouped into release categories and insights have been derived based on these categories. Also, the phenomenological evaluations have been reviewed to gather additional insights.

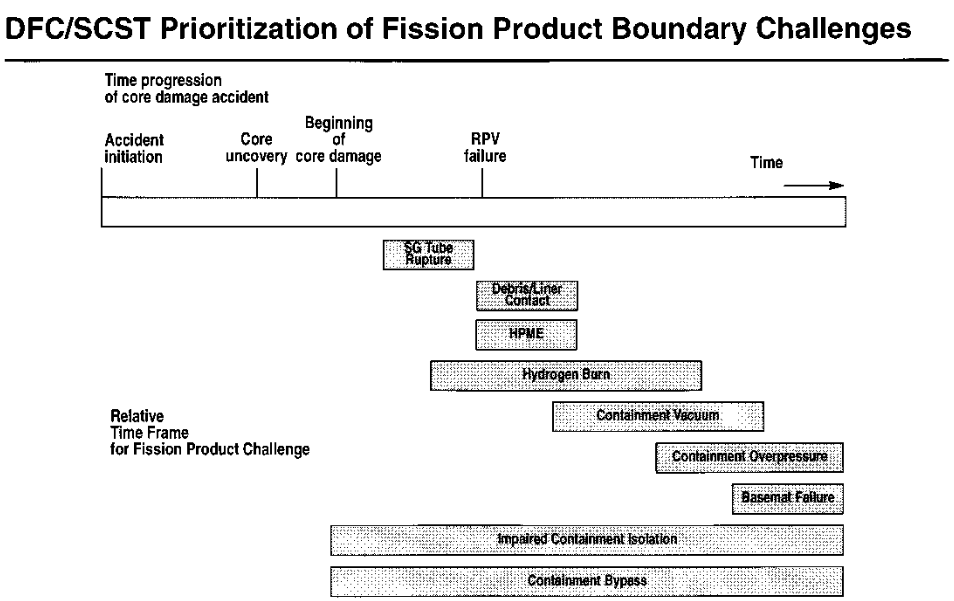
Level 1 PSA

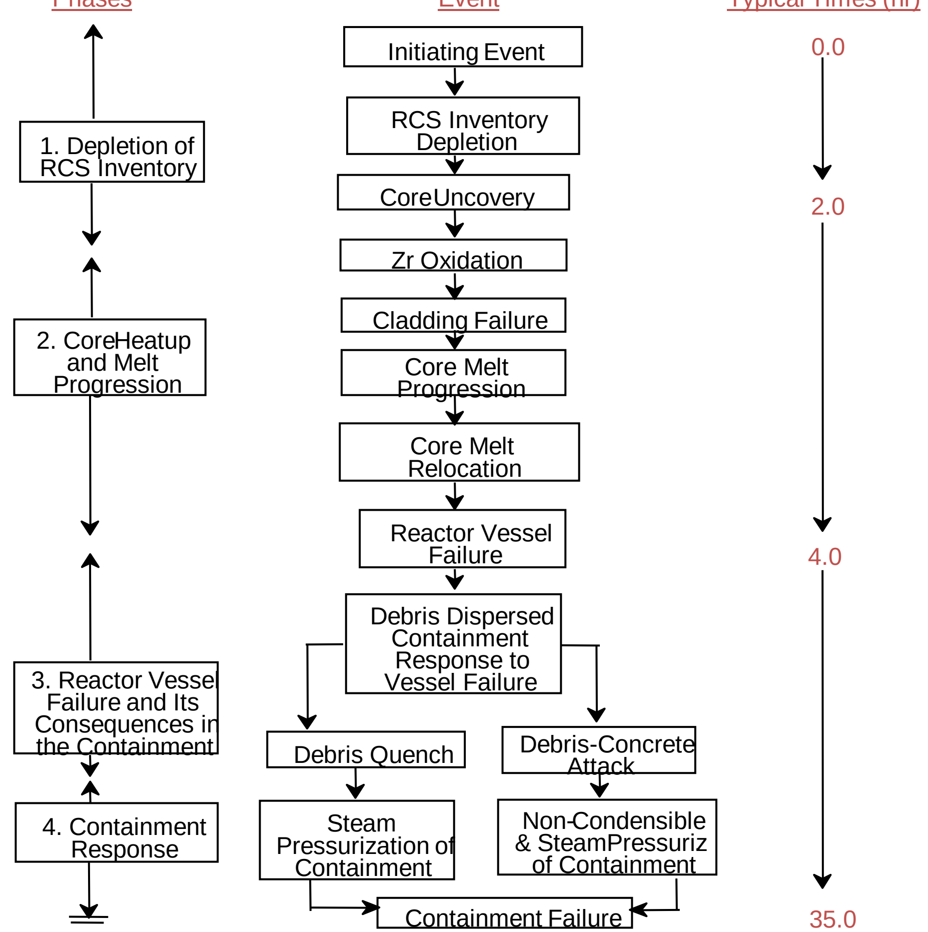
Sequences that lead to core damage after 24 hours

Level 2 PSA

Timing and severity of challenges to the barriers against releases of radioactive material - generic

- The initiating events were selected based on the dominant core melt sequences of a number of IPEs. The time sequence information was obtained from the IPE source term analyses which were performed with MAAP 3.0B, Revision 17.



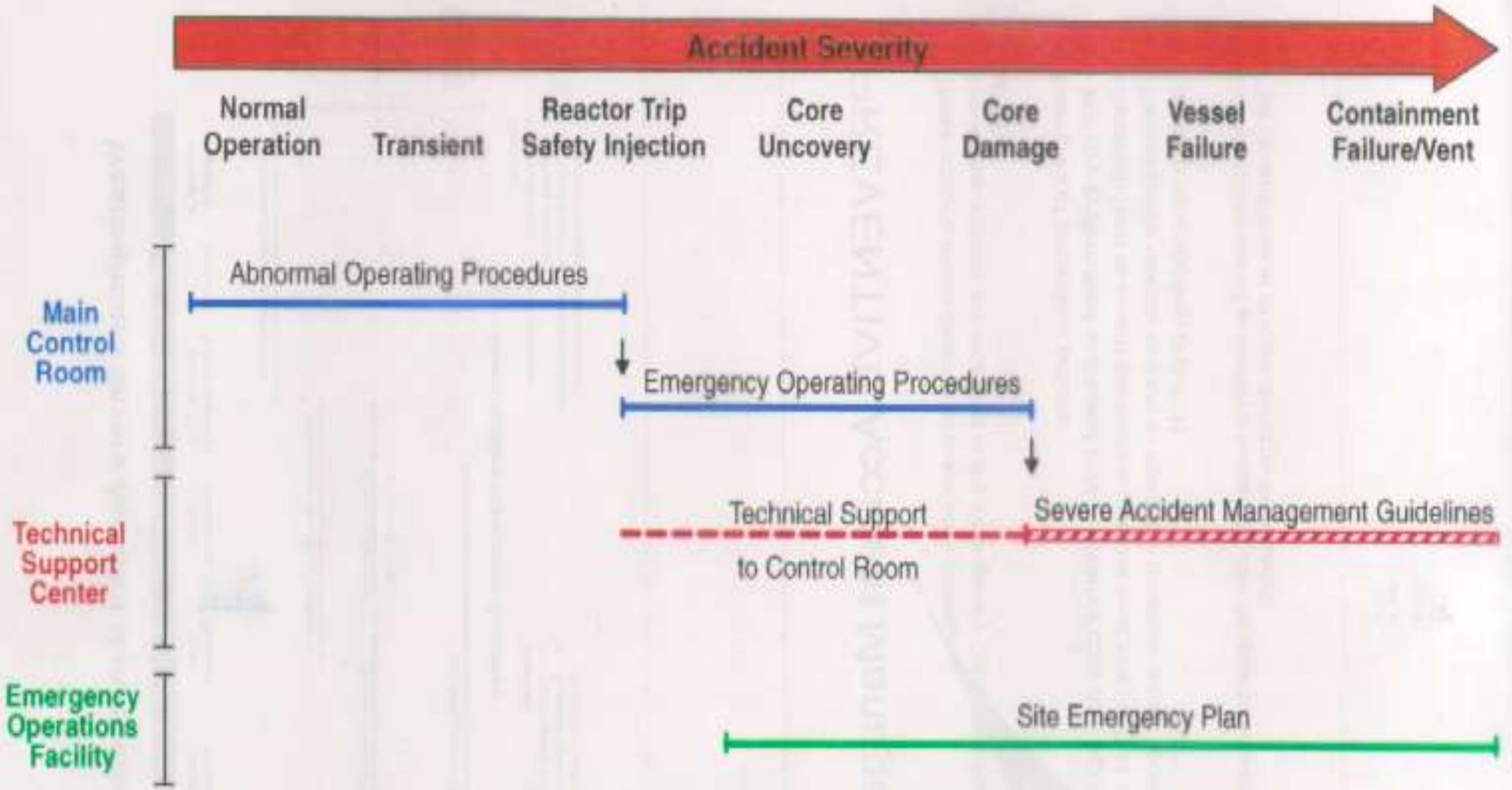


Development of NEK specific SAMG based on WOG generic guidelines:

- Generic Strategies defined (an action /set of actions) to be taken; a challenge that is to be mitigated, and the equipment that will be used);

- Many steps needed to developed plant specific procedures (development of plant specific background documentation, procedures, implement required changes in EP,..)

Westinghouse Severe Accident Management



Severe Accident Management Guidelines

Purpose

- Protect fission product boundaries
- Mitigate releases
- Mitigate severe accident phenomena
- Restore controlled stable condition

Features

- Implemented by TSC
- Separate from EOPs
- Symptom based

WOG Generic SAMG Implementation

- Review of WOG Generic SAMG applicability;
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- Review of NEK EOPs to incorporate transitions to SAMG;
- Writing of plant-specific control room SACRGs;
- Writing of plant-specific TSC guidance, including SAGs, SCGs, DFC, SCST, and SAEGs;

Background Documents - Strategies

Purposes were:

- Identify if all **<u>generic strategies</u>** are applicable to NEK - can successfully be applied;

  Accident Management measures or strategies may be PREVENTIVE (delay or prevent core damage) or MITIGATIVE (mitigate core damage and protect fission product boundaries) or BOTH

  SAMG

- Verify if **IPE insights are <u>addressed</u>** in generic strategies;

- Identify the plant **<u>specific capabilities</u>** (equipment that will be used), action to be taken to mitigate the challenge

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

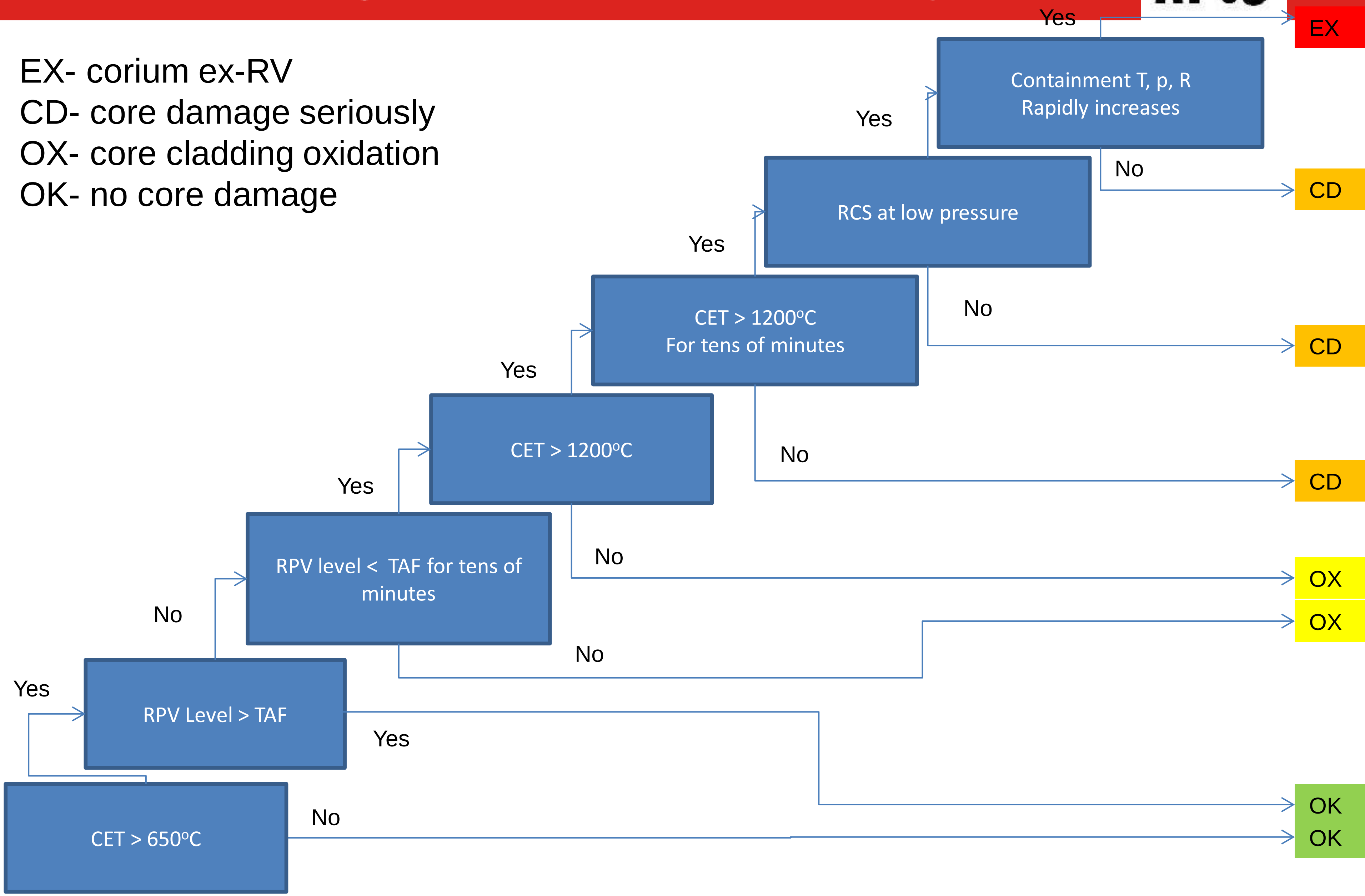
Background Documents - Setpoints

The developed report served as a reference and source of data for plant specific SAMG procedures taking into account:

- Plant specific equipment, system design basis and characteristic;
- Finding and results of the phenomenological evaluations and plant specific analysis done in the scope of IPE Level 2 (example: the pressure at which there is a low probability for containment failure);

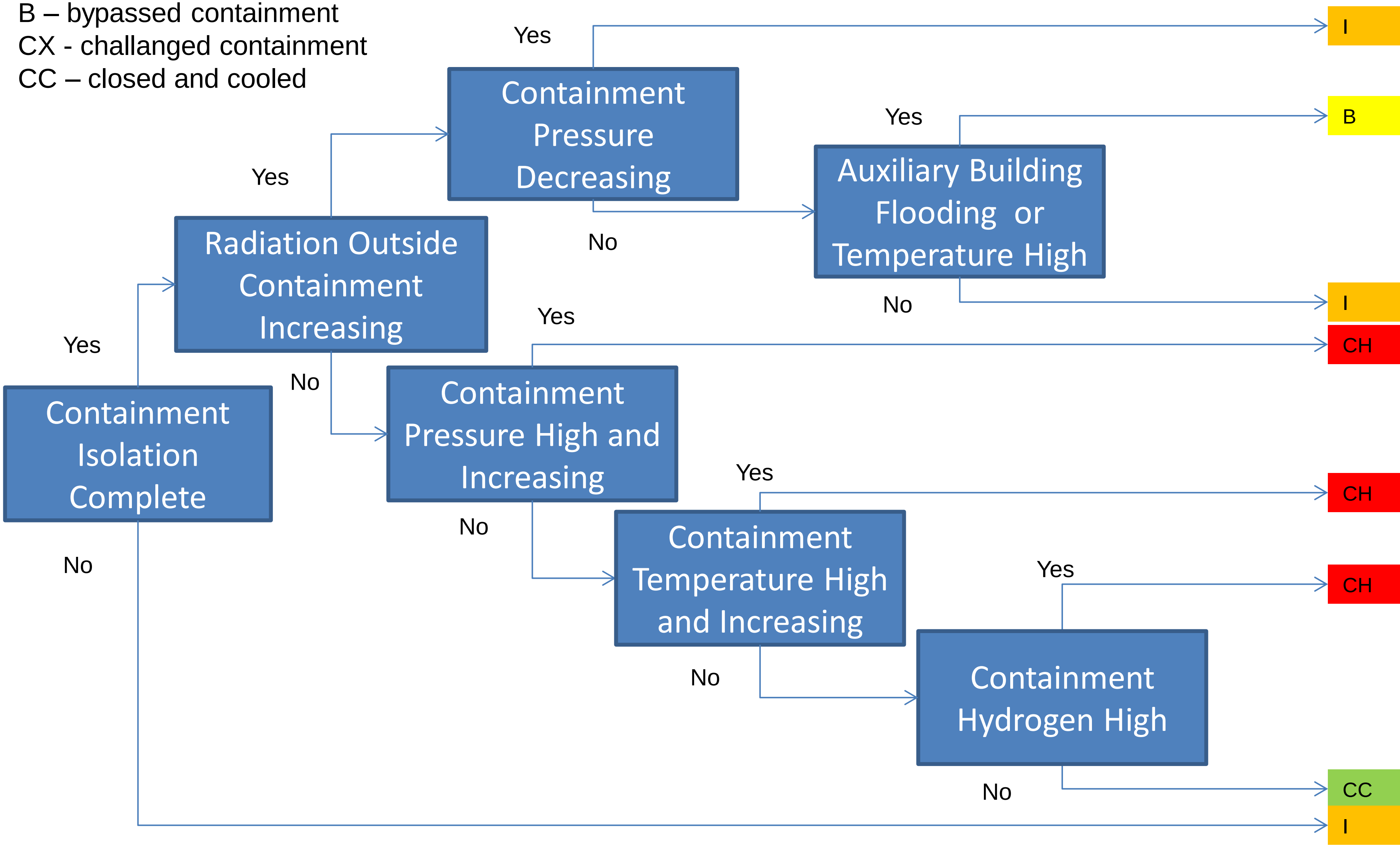
Core Damage Condition Status Tree example

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



Background Documents – Setpoints, examples

Determination of NEK Specific Value:

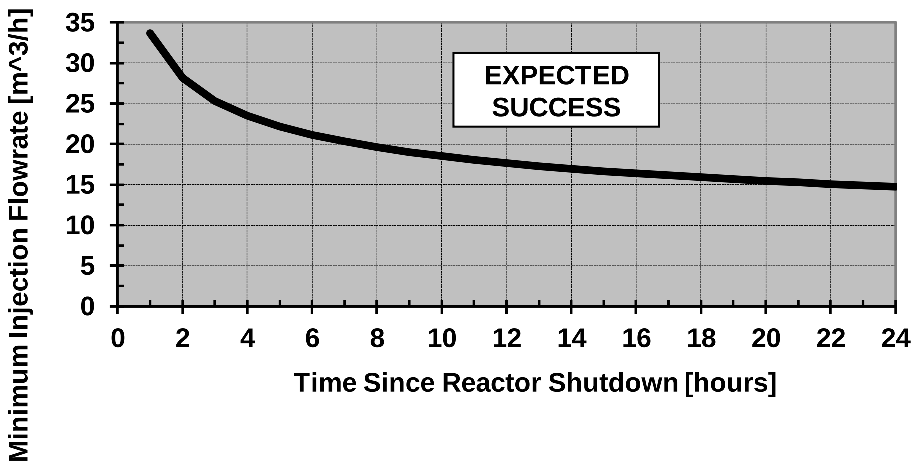
The available NPSHs from the VCT calculated for charging pumps, exceed the required NPSHs independently to the water level in the VCT (Document Id:630-7).

Based on discussion for L04, vortexing formation is limiting for the CVCS pumps. Water velocity in VCT outlet nozzle (4-CS-151R, sch. 40 pipe) due one centrifugal pump running is 1.22908 m/s (based on 160 gpm flowrates [36.34m^3 /h] - USAR Table 9.3-2, outlet nozzle cross section area of 8.213E-3 m^2). Based on curve of Hydraulic Standard required relative submerge is 0.632m.

(1) Centerline of pipe = 116.786m (dwg. E-304-680)
(2) Centerline of level tap = 117.355m (dwg. B-814-670, sh. 19)
(3) Radius of outlet pipe = 0.05113m
(4) Relative submerge = 0.632m
(5) (2)-(1)=0.569m
(6) Relative submerge toward level tap = (4)+(3)-(5)=0.114m
(7) 100% of level span is 1.906m
(8) Relative submerge toward level tap in % = (6)/(7)*100= 5.988%

NEK Specific value for L06 = 6.0 % of VCT level

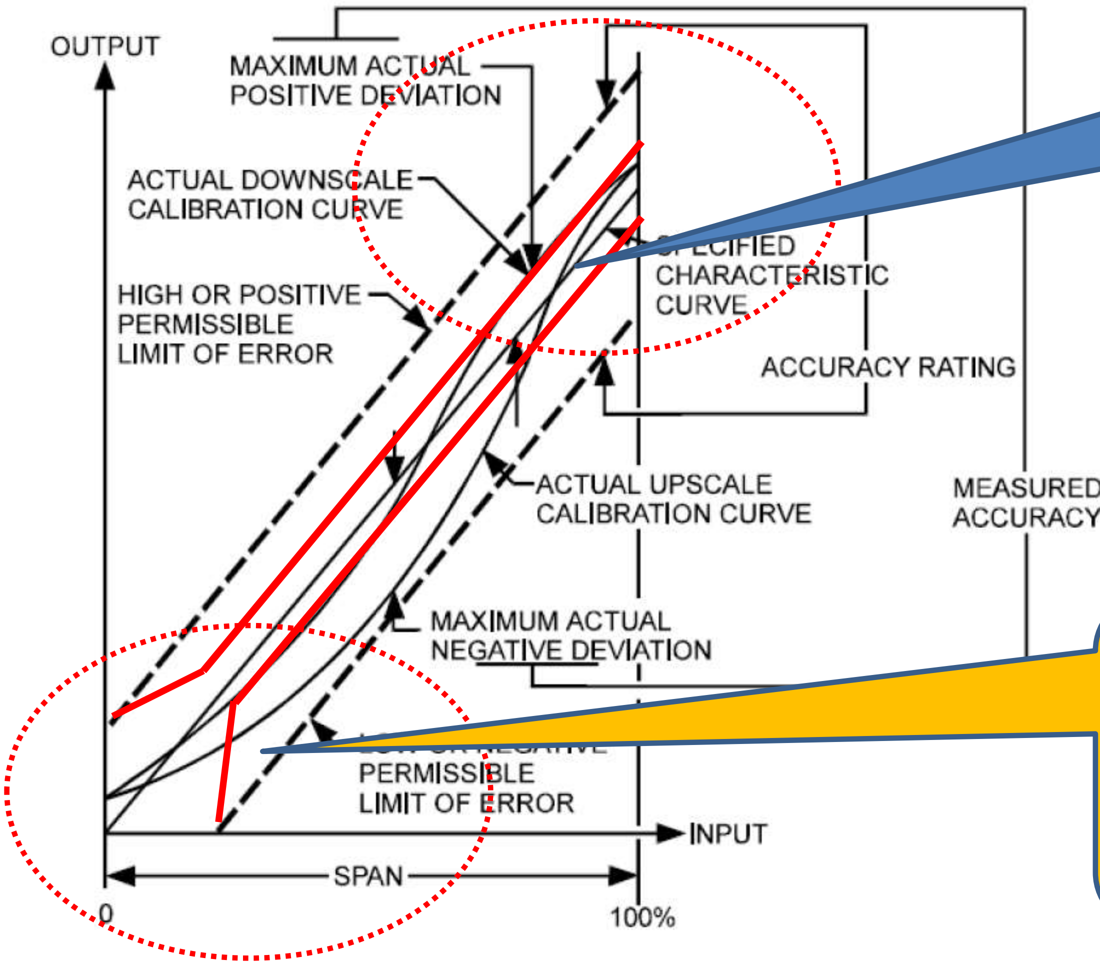
Figure 2-1
Minimum Injection Flowrate for Long Term Decay Heat Removal



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!

Procedures

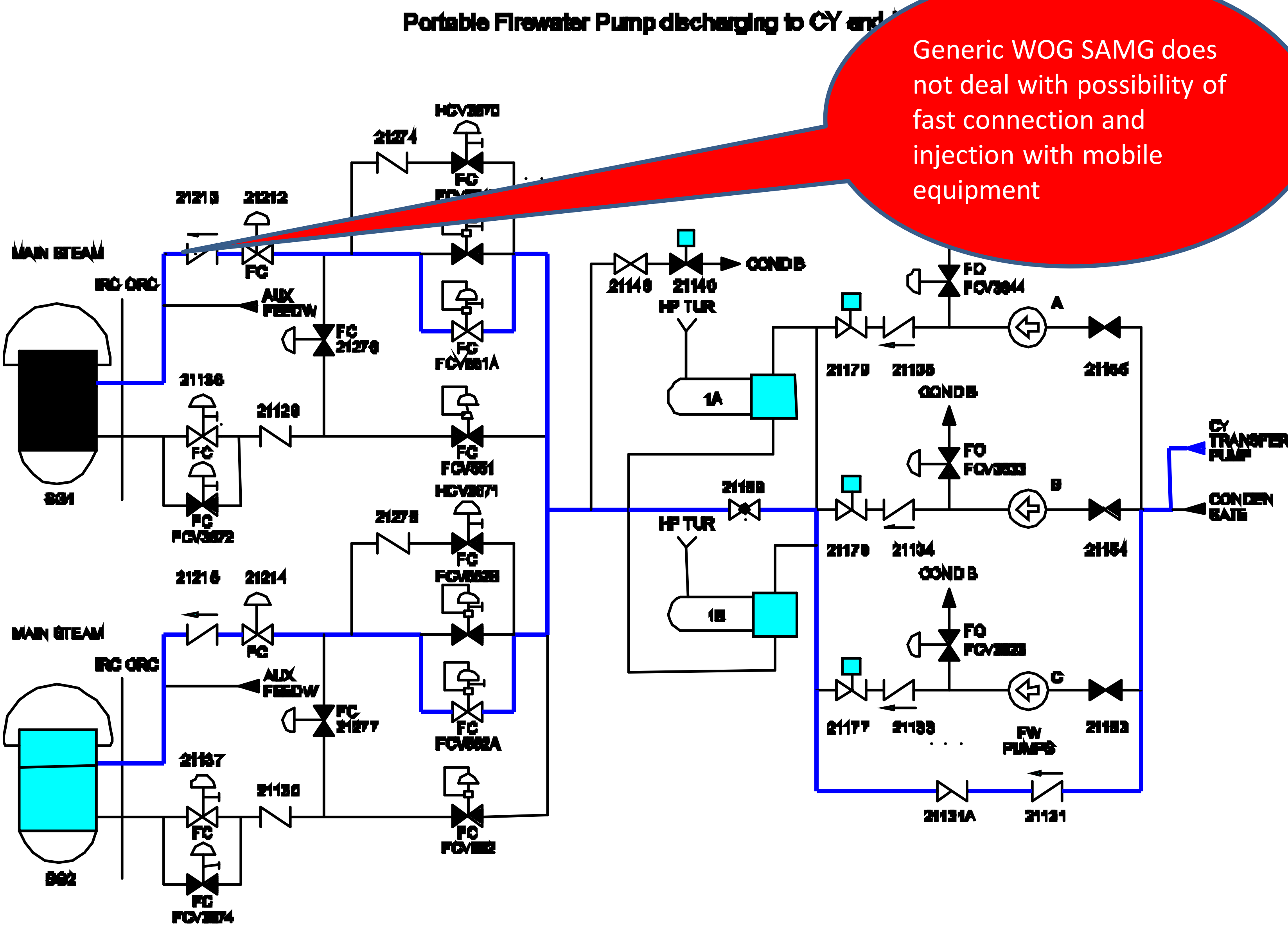
Writing Team established from:

- Operations personnel (SE and SS to assure proper linkage to EOP and up front familiarization);
- Engineering personnel (people included into preparation of SAMG background documentation - Setpoint, Plant Capabilities, CA);

Procedures and background documentation were reviewed:

- internally (NEK TSC members
- externally (Westinghouse - WENX-00-05 and IAEA RAMP mission , IAEA-TCR-00959)

Procedures - Attachments



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

Review of NEK E- plan

- Transition from ERGs to SAMG
- Termination of SAMG
- Identification of personnel to evaluate SAM actions
- Special approval for intentional fission product releases

VALIDATION - WENX-00-29

- Validation performed on WCAP-14213
  - Training and Integral Exercise in March 2001;
- Selected scenarios have been examined by KFSS

Validation of NEK SAMG benefits:

Identifies plant-specific technical inaccuracies

Demonstrates “usability” of guidelines

Identifies potential conflicts and problems

Provides training and experience

Example for Validation Acceptance Criteria

EOP - SAMG INTERFACE

- Is the EOP-SAMG transfer point clear and useable?
- Is the timing appropriate?
- Is the responsibility for the EOP-SAMG transition clearly defined?

CONTROL ROOM GUIDELINES

- Can the needed plant parameters be obtained?
- Are the decision steps logically ordered?
- Are there extraneous or missing steps?
- Can each of the steps be completed?
- Are the instructions clear and understandable?
- Is the communication between the control room and the TSC emphasized enough?

SAMG Contents

SACRGs Severe Accident Control Room Guidelines
SACRG-1 Severe Accident Control Room Guideline Initial Response
SACRG-2 Severe Accident Control Room Guideline for Transients After the TSCis Functional

DFC TSC Diagnostic Flow Chart

SAMGs	Severe Accident Guidelines
SAG-1	Inject into the Steam Generators
SAG-2	Depressurize the RCS
SAG-3	Inject into the RCS
SAG-4	Inject into Containment
SAG-5	Reduce Fission Product Releases
SAG-6	Control Containment Conditions
SAG-7	Reduce Containment Hydrogen
SAG-8	Flood Containment
SCST	TSC Severe Challenge Status Tree

SCGs	**Severe Challenge Guidelines**
SCG-1	Mitigate Fission Product Releases
SCG-2	Depressurize Containment
SCG-3	Control Hydrogen Flammability
SCG-4	Control Containment Vacuum

SAEGs	**Severe Accident Exit Guidelines**
SAEG-1	TSC Long Term Monitoring Activities
SAEG-2	SAMG Termination

RAMP Reanalysis

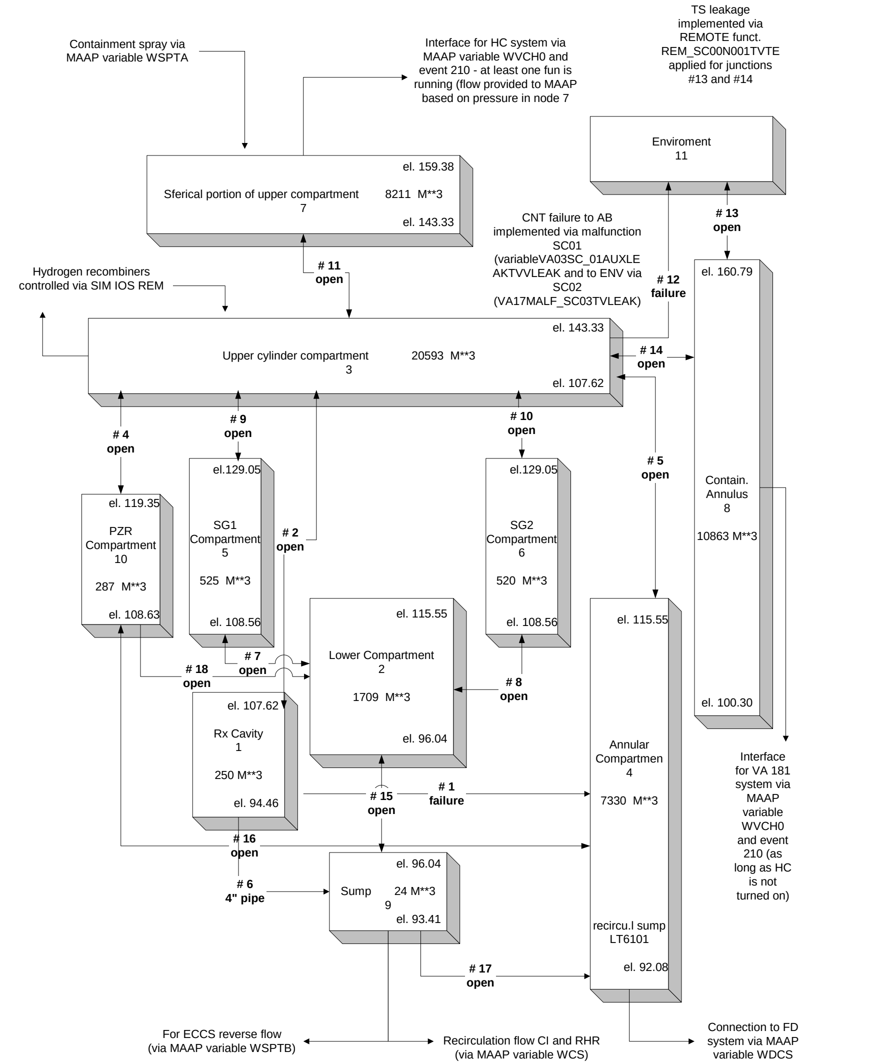
During NPP Krsko RAMP mission in 2001 some hydrogen related questions were raised and they are addressed again later in plant's PSR. It was attempted to critically review plant safety taking into account increased knowledge of the subject, the way how problem was treated in advanced LWR designs, and changes in regulations. Following three actions were identified:

- PSR 5.3-6: RAMP analyses of possible non-uniform distribution of hydrogen within the containment space
- PSR CH2.12_P18-3: RAMP analyses of Passive Autocatalytic Recombiners (PAR)
- PSR CH2.2_P7-3: RAMP analyses of potentially decreased corium coolability for the burnable gas management and containment long term pressure management.

Usage of MAAP 4.0.5 for hydrogen evaluation

At the time of preparation referenced paper, the new Subroutine OXIPAR for Debris Particle Oxidation was added in Subroutine DEBRIS, DBJET and DCH1/2. Metallic debris oxidation is a key mechanism that allows hydrogen to be produced which subsequently burned during DCH in the high-pressure IET experiments.
The purpose of the new subroutine is to provide a hydrogen-producing mechanism during the event needed to match what actually happened in the experiments.
It is used for metal-water reactions during steam explosions and during DCH. OXIPAR is called by DEBRIS during steam explosions, by DBJET during corium fragmentation, and by DCH1/2 during DCH.

Nodalization scheme



Usage of MAAP 4.0.5 for hydrogen evaluation

Accident sequences:

- HSBO01: In the station blackout sequence, HPI, LPI, AFW and MFW are turned off. The sequence is analyzed for the hydrogen study since it represents a case with large hydrogen generation in the reactor vessel which ends with reactor pressure vessel failure from high RCS pressure.

- HLLOCA3: The large LOCA accident is a 27.5-inch break (double ended) in the cold leg. One HP1 pump is assumed available. LPI, AFW and containment sprays are unavailable from beginning of the sequence. ECCS recirculation is assumed unavailable, and so core melt occurs following emptying of the RWST.

- HSLOCA2: The small LOCA accident is a small break (0.5-inch) in the cold leg. LPI, HPI, AFW and containment sprays are assumed unavailable at the beginning of the sequence.

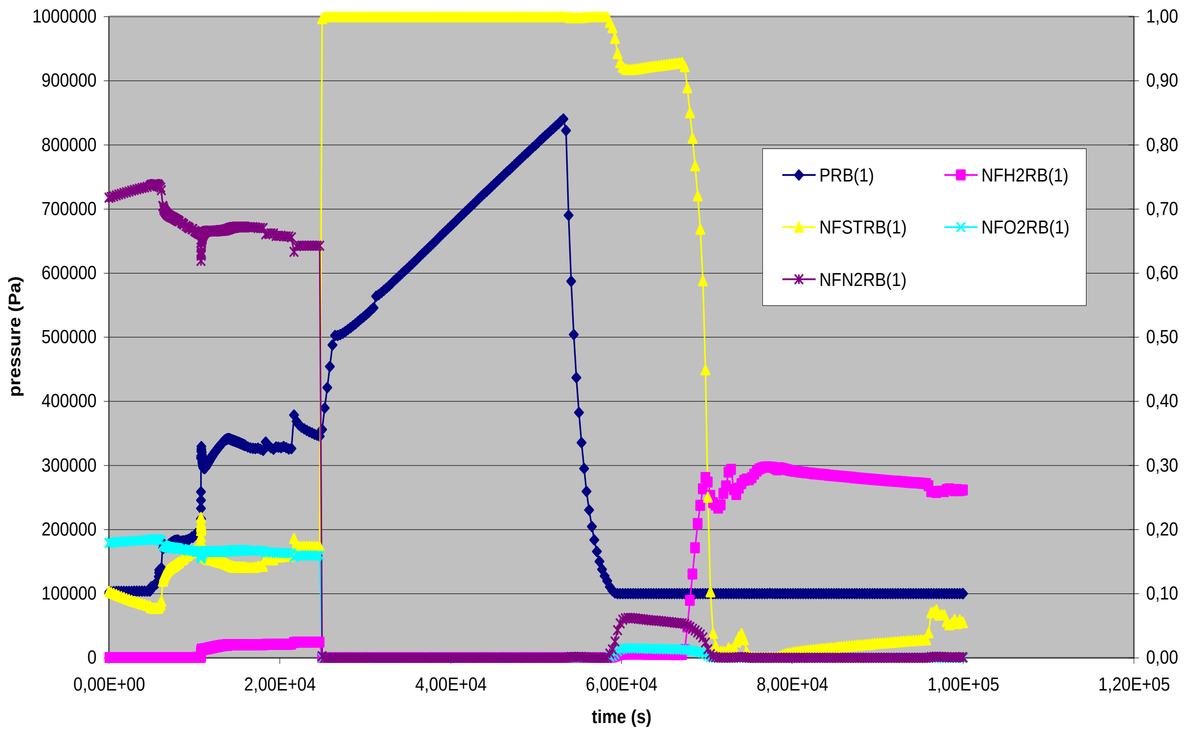
The assumptions used in severe accident scenarios were very conservative and no active strategies such as Severe Accident Management Guidelines (SAMG) have been applied in order to assess limiting hydrogen concentrations.

Usage of MAAP 4.0.5 for hydrogen evaluation

APoS

Supporting Accident Analysis (generic & plant specific)

"Hydrogen Behaviour in PWR Containment Evaluated by MAAP4.0.5"; paper presented at the "5th International Conference on Nuclear Option in Countries with Small and Medium Electricity Grids"; Dubrovnik, Croatia, I. Bašić, T. Bilic-Zabric and J. Spiler (NEK);

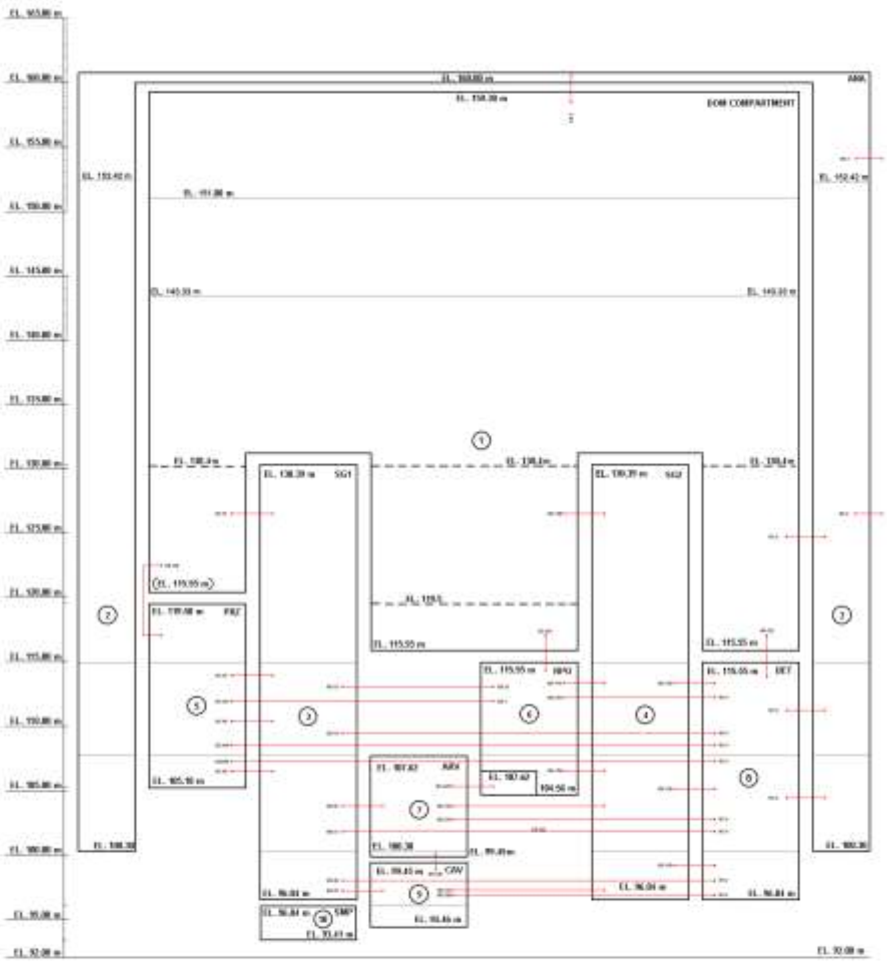


Station Blackout (HSBO1) MAAP Results, Containment Pressure, Hydrogen, Nitrogen, Steam Mole Fraction in the Cavity

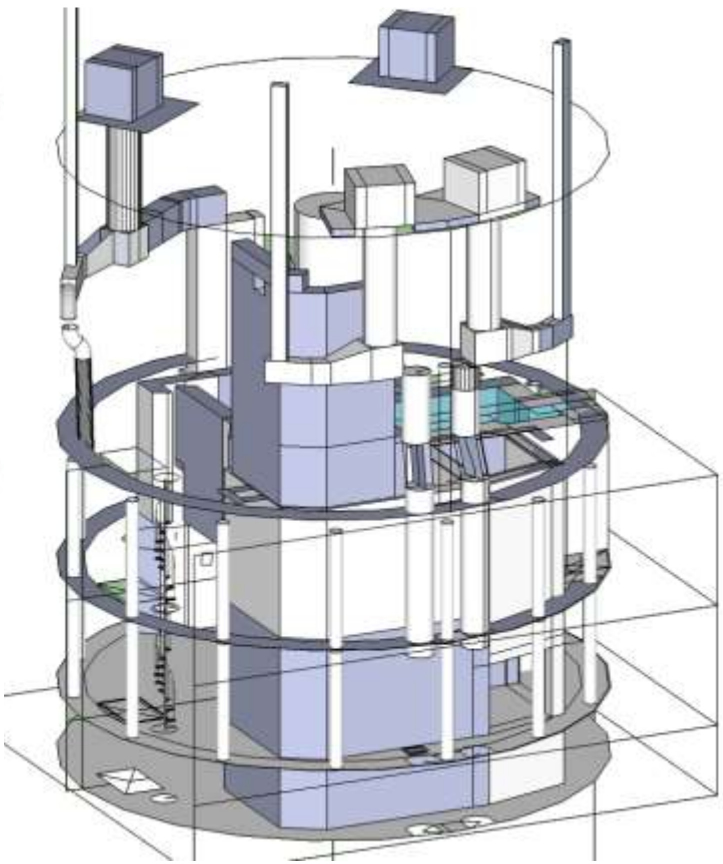
Run ID	MAAP 3B H2 RX vessel failure (kg)	MAAP4.0.5 H2 at time of RX vessel failure kg)	MAAP 3B H2 at end of the transient (kg)	MAAP4.0.5 H2 at end of the transient (kg)
SBO (HSBO1)	180	261	255	266
LB LOCA (HLLOCA3)	80	185	103	185
Small LOCA (HSLOCA2)	190	274	320	280

Analysis Results

Reference: Hydrogen Distribution in NPP Krško Containment Report number (NEK ESD TR 13/10), D. Grgic and T. Fancev (FER)

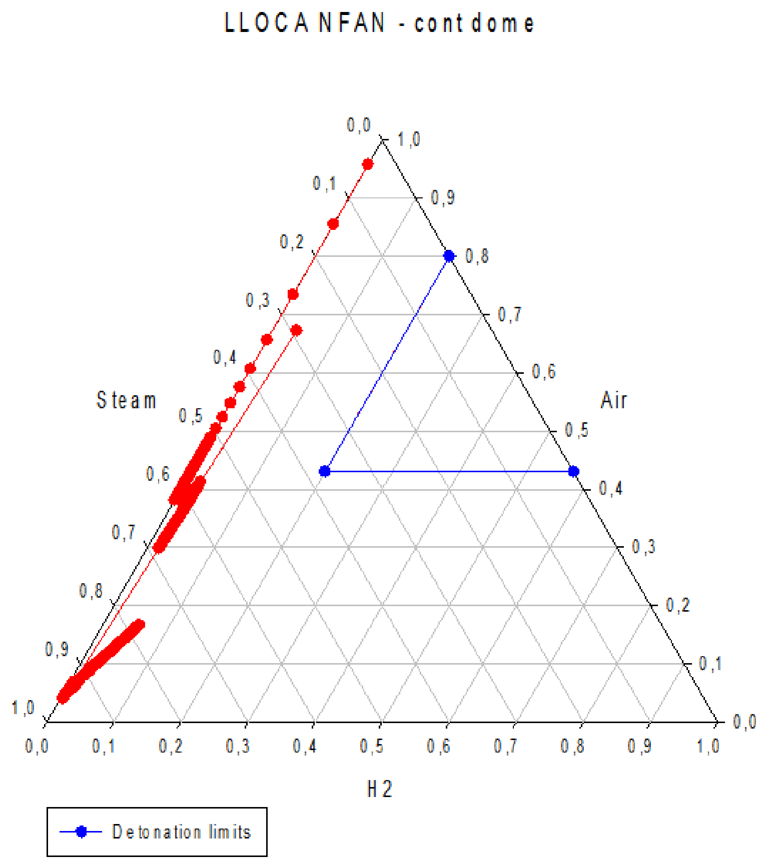


GOTHIC nodalization with subdivided containment dome

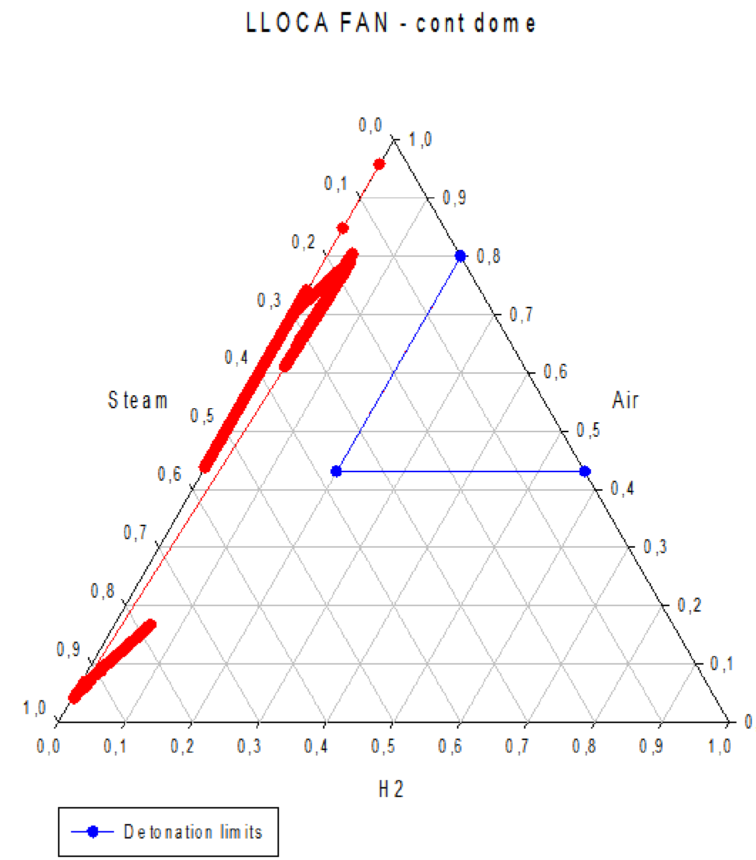


3D view of containment with SG1 compartment

Analysis Results



H2-air-steam diagram for containment dome, LLOCA NFAN, GOTHIC run



H2-air-steam diagram for containment dome, LLOCA FAN, GOTHIC run

Spent Fuel Pool Vulerability

Example: SFP States for Risk Significance Evaluation, Time Window to Recover SFP cooling

State	Description	SFP Decay Heat (MW)	SFP Water Inventory	Time to Boil (hr) [1]	Time to Evaporate to FA+1m (hr) [2]	Duration (days)	Duration (%)
SFP1	Complete core from the previous cycle in the SFP [3]	6.40 – 4.39	C1	11.0 – 20.0	111.3 – 162.6	15.2	2.8%
SFP2	Partially burnt FAs from previous cycle returned to the core. Decay heat level higher than 1.5 MW.	2.37 – 1.50	C1 C2 (C3)	44.8 – 74.9 32.0 – 53.5 (32.0 – 53.5)	303.3 – 474.7 224.7 – 351.7 (174.1 – 272.6)	71.2	13.0%
SFP3	Decay heat level lower than 1.5 MW.	< 1.50	C2 (C3)	> 53.5 (> 53.5)	> 351.7 (> 272.6)	461.5	84.2%
					Total:	547.9	100%

Fukushima accident – SANDIA Evaluation

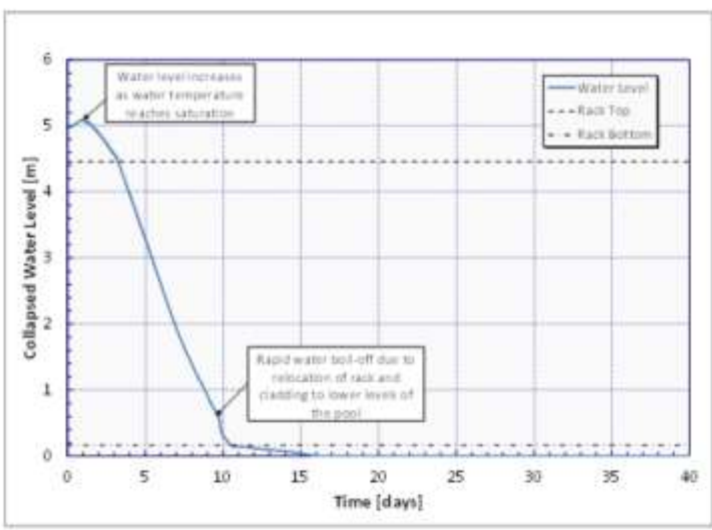


Figure 122. MELCOR Predicted Spent Fuel Pool Collapsed Water Level (0.5 m above Top of Racks Case).

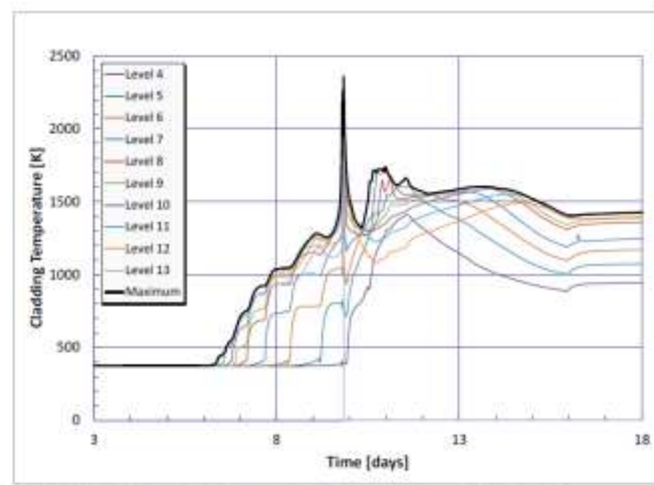


Figure 124. MELCOR Predicted Spent Fuel Pool Maximum Cladding Temperatures (0.5 m above Top of Racks Case, Detailed View).

Harsh Environment - equipment surveviability during DEC conditions?

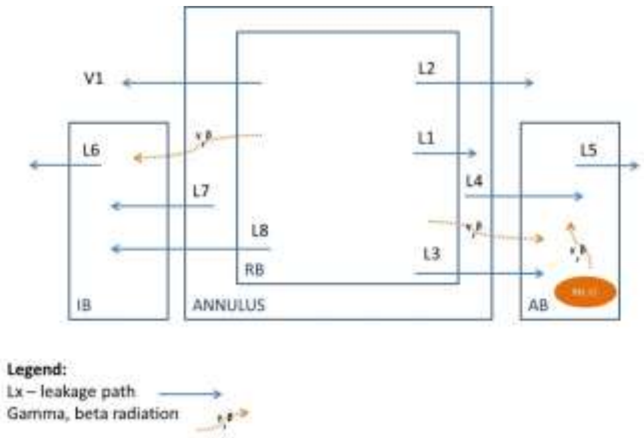


Figure 1 Principle leakage scheme for dose and TH evaluation

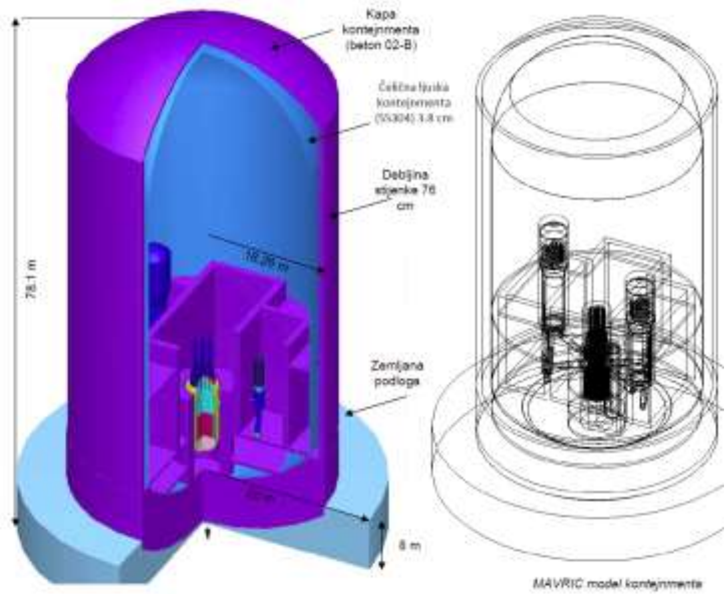


Figure 1 3D Containment model for gamma calculation in AB/IB

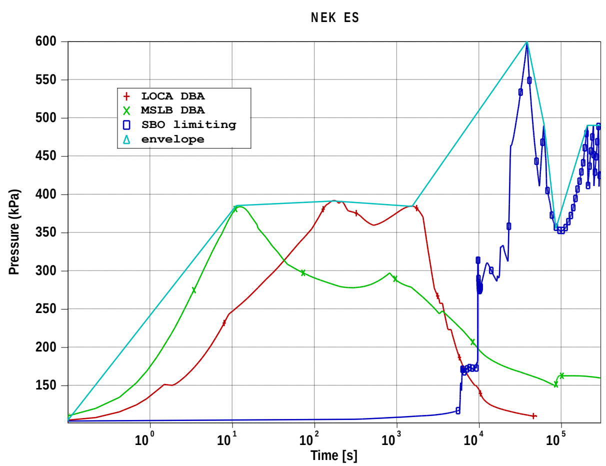


Figure 1 NEK ES DBA and DEC RB pressure envelopes (log time scale)

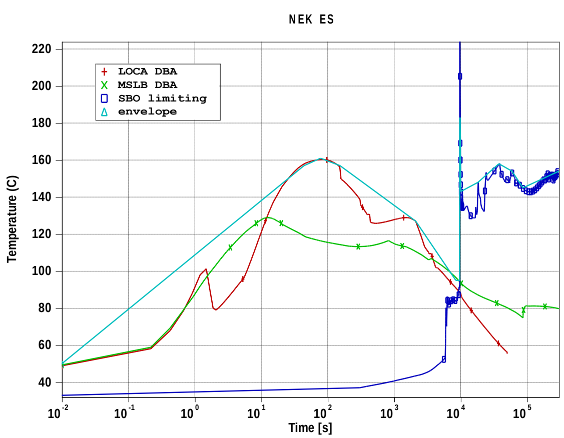
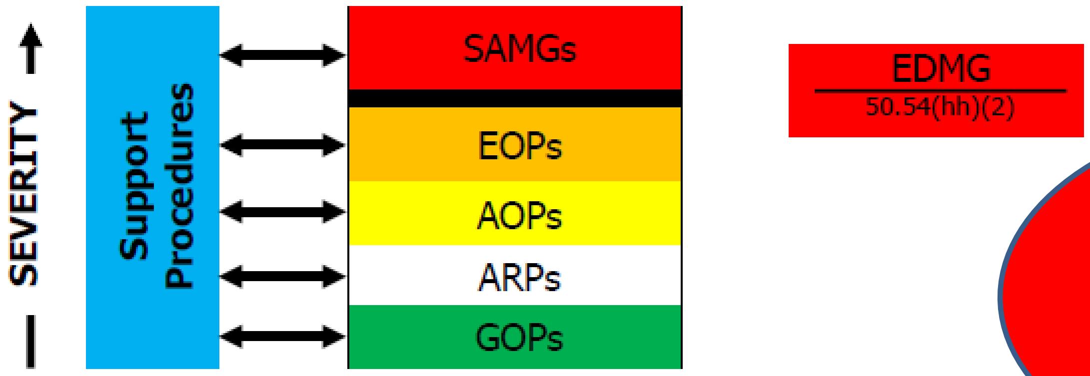


Figure 1 NEK ES DBA and DEC RB temperature envelopes (log time scale)

Implementation of NEI 12-06 (FLEX)

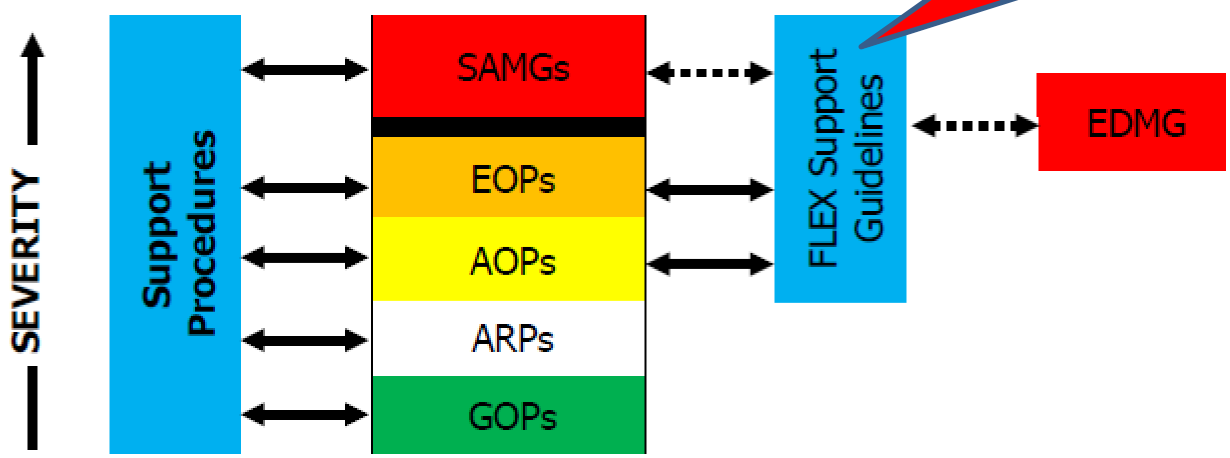
(a) Existing View of Typical Operating Procedure Hierarchy



Added as EOPs Attachments (37 !!!) which are referenced to SAMGs if needed

Revision of SAMGs

(b) Future View of Typical Operating Procedure Hierarchy



Major Plant Safety Upgrade: 2000 – nowdays

Safety upgrade program:

- BB1 (3rd EDG)
- PARs (apssive autocatalytic recombiners)
- PCFV (passive containment filter venting)
- Mobile MHX
- ECR (emergency control room)

SAME (Severe Accident Measurement Equipment) modification performed to extend existed mobile equipment and satisfy NRC B.5.b (NEI 06-12) measures and requirements for all NPPs:

- ensure equipment and personnel to manage serious fires and
- ensure mobile equipment for:
  - Core cooling and Containment cooling,
  - Spent Fuel Pit (SFP) cooling.

In such manner the emergency such as a commercial aircraft crash on the plant can be managed.

Preliminary Post Fukushima Actions

- Response on STORE (**S**afety **T**erms **of** **Re**ference) including NRC Bulletin 2011-01, and WENRA stress report:
  - May 2011, preparation phase of DMP
  - June 15th, 2011, presentation of results and proposed changes to the KSC (including SES 10CFR50.59, UCP and DMP)
  - July 1st , 2011 presentation to the SNSA, approval
  - July – August 2011, implementation (OL25)
  - September 30th, 2011, testing and notification of new configuration

- Mitigative actions need to take into account the following scenarious:
  - Loss of SBO and UHS without any off-site support 72 hours,
  - Time windows > 7days, core damage postulated,
  - Extreme external events (seismic, flooding, storms, etc.),….

Preliminary Post Fukushima Actions

Covering:

- **Design Modification** (establishing a new system "AE – Severe Accident Management Equipment" covering the hardware connections to available systems (AFW, MFW, CI, CS, IA, VA, etc.), local control of SG PORVs and purchase the mobile DGs, mobile injection and flood pumps, etc.)
- **Software Changes** (Associated SEOPs/SAMGs and Emergency Program changes and purchase the personal protection equipment)
- **Safety Function establishing**:
  – Alternative Residual Heat Removal through SGs (alternative feed/bleed means)
    - E.g Rosenbauer pump FOX III ((60m3/hr, 15 bar), OR HS60 (60m3/hr, 11 bar)), fix connection downstream AFW pumps/upstream FW cont. isolation

  – Alternative Residual Heat Removal through RCS (PORVs with alternative feed/bleed means)

  – Alternative SFP makeup and cooling

Krsko NPP Response to NEI 06-12

EOP SAME/FLEX Attachment Example:

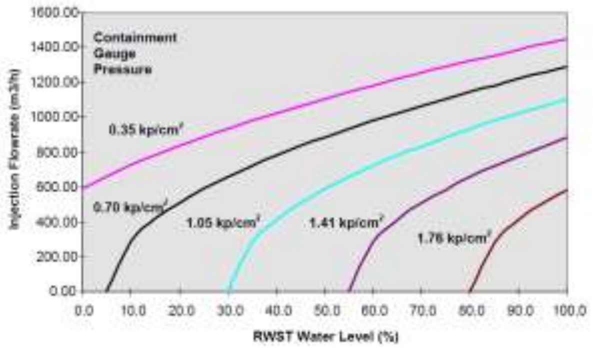
- Format of standard EOP is used
  - Detailed instruction for local operators and firemen
- System flow diagrams were corrected to clearly evidence possible SAME fact connections of alternate equipment
- SAME equipment on-site is tested periodically
  - Pumps head/flowrate
  - DGs capacity and initiating
  - Pipe hose status, spare nozzles and connections, etc.
- Training and drill is annually performed taking into account realistic scenario driven on simulator and including MCR, TSC, OPC and locall fire brigade

Number: EOP-3.5	Title: APPENDIX 34 RWST GRAVITY DRAIN TO CONTAINMENT	Rev./Date: Rev. 0 Jun 2011

CAUTION: This Appendix can be used in procedure ECA-0.0 based on decision of Technical Support Center if secondary heat sink is lost and can not be established in timely manner.

NOTE: Gravity drain flow from RWST to containment sump depends on the containment pressure as shown in the next Figure 1. Minimum water level in RWST is shown in Figure 2.

Figure 1
Flowrate from RWST Gravity Drain



Notes:

(a) Curves are calculated for 2 meters of water in containment.
(b) When level in containment is higher than 2 meters, the injection flow is smaller than showed in Figure 1
(c) Backflow to RWST is possible if containment pressure is high enough.

Availability of important support functions as well as possibility of their restoration

- AC/DC capability for essential SSCs and critical safety function should be assesed together with possible alternatives (existing alternative sources + portable devices + FLEX connection)
  - Special attention to diagnostic instrumentation

- Water sources for makeup of SG and RCS should be evaluated togetger with alternative paths and sources for prolonged severe time window (4h, 24h, 72h...)
  - Special attention for long term cooling of RCS and containment

- Compressed Air for essential valves necessary for establishment of critical safety function
  - Special attention for containment isolation valve or PRZR PORV and SG PORVs

Conclusions

Development of KRŠKO Specific SAMG covered:

- The current worldwide state of the art in severe accident research including experimental and analytical efforts;
- Plant specific capabilities (structures, systems, components) and strategies assessment including FLEX capability NEI 06-12;
- Generic and specific PSA insights assessment;
- However, **that certain changes and revision of SAMGs and SEOPs were introduced by post Fukushima WENRA stress tests evaluations and safety upgrade**
  - **PARs, PCFV, new ECR, additional LP SIS pump, mobile RHR HX (MHX), etc**

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Questions?
Comments?

Thanks for your attention!