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# Insights from Internal Fire PSA of UK ABWR in Generic Design Phase

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Internal Hazards PSA/PRA I

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# 1. Introduction – UK ABWR

- Hitachi-GE proposed to build Advanced Boiling Water Reactor (ABWR) plants in UK, based on enhanced Japanese ABWR design.
- Hitachi-GE has been honored to receive a Design Acceptance Confirmation (DAC) from ONR, and Statement of Design Acceptability (SoDA) from Environment Agency and Natural Resources Wales on 13 Dec. 2017.
- The generic Pre-Construction Safety Report (PCSR) is the main submission for Generic Design Assessment (GDA) supplied by Hitachi-GE. UK ABWR PSA is provided in the chapter 25 of PCSR.

More information found at:
*Hitachi-GE Nuclear Energy, Ltd., Generic PCSR Chapter 25: Probabilistic Safety Assessment,*
*http://www.hitachihgne-uk-abwr.co.uk/downloads/2017-12-14/UKABWRGA91-9101-0101-25000-RevC-PB.pdf*

# 1. Introduction – UK ABWR PSA for GDA

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| Source of Radioactivity | Internal events | Internal fire* | Internal flood* | Seismic events* | Other IHs* | Other EHs* |
|---|---|---|---|---|---|---|
| **Reactor (at Power)** | **Detailed PSA** | | | | Bounding assessment | |
| **Reactor (Shutdown)** | | Scoping Analysis | | | | |
| **SFP** | | | | **Detailed PSA** | | |
| **Fuel Route** | | Qualitative discussions to argue risk is insignificant | | | | |
| **Turbine system** | Simplified assessment | | | | | |
| **Radwaste system** | | | | | | |

**Covered by this presentation.**

* Retained after screening process in IAEA SSG-3

More information found in:
*N. Hirokawa, et al., OVERVIEW OF PSA FOR THE UK ABWR GENERIC DESIGN ASSESSMENT, ICONE26-82553*

# 1. Introduction – Background and Objective

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NUREG/CR-6850 and NUREG/CR-7114 are generally intended for application to an operating plant rather than a plant in design phase.

**Certain challenges: Absence of**

- Detailed circuit design
- Detailed locations of raceways and ignition sources
- Comprehensive Fire Hazard Analysis (FHA)
- Operating procedures

→ Simplified and conservative approaches developed*

**Objective of this presentation:**

- Further describes approaches to generic design FPSA (Section 2)
- Introduces approaches to model refinements (Section 3)
- Introduces results and insights (Section 5)
- Introduces peer reviews (Section 6)
- Introduces risk-informed activities (Sections 4 and 7)

*P. Guymer, et al., "DEVELOPMENT OF INTERNAL FIRE PSA FOR NEW BUILD UK GENERIC DESIGN ASSESSMENT", 15th Int. Post-Conf. Seminar on FIRE SAFETY IN NUCLEAR POWER PLANTS AND INSTALLATIONS, Bruges, Belgium, October, 2017.

# 2. Approaches to Generic Design FPSA

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| Guidance / Standard | Contents Used |
|---|---|
| NUREG/CR-6850 | Overall Guidance |
| NRC/NEI Frequently Asked Questions | Supplements to NUREG/CR-6850 |
| NUREG/CR- 7114 | Additional Guidance for Shutdown FPSA |
| NUREG-2169 | Generic Ignition Frequencies,<br>Generic Non-suppression Probabilities |
| NUREG/CR-7150 | Spurious Operation Probabilities, Failure Probabilities to Clear Spurious Operation |
| NUREG-1921 | Internal Fire HRA Guidance |
| NUREG-1855 / EPRI-1026511 | Approach to Uncertainties |
| NEI 00-01 | Multiple Spurious Operations Identification |
| NEI 07-12 | Peer Review Process |
| ASME/ANS RA-Sb-2013 | Requirements (for Peer Review) |
| Technical Assessment Guides (TAGs) | UK Regulatory Expectations |
| AESJ-SC-RK007:2014 | Additional Guidance |

# 2. Approaches to Generic Design FPSA

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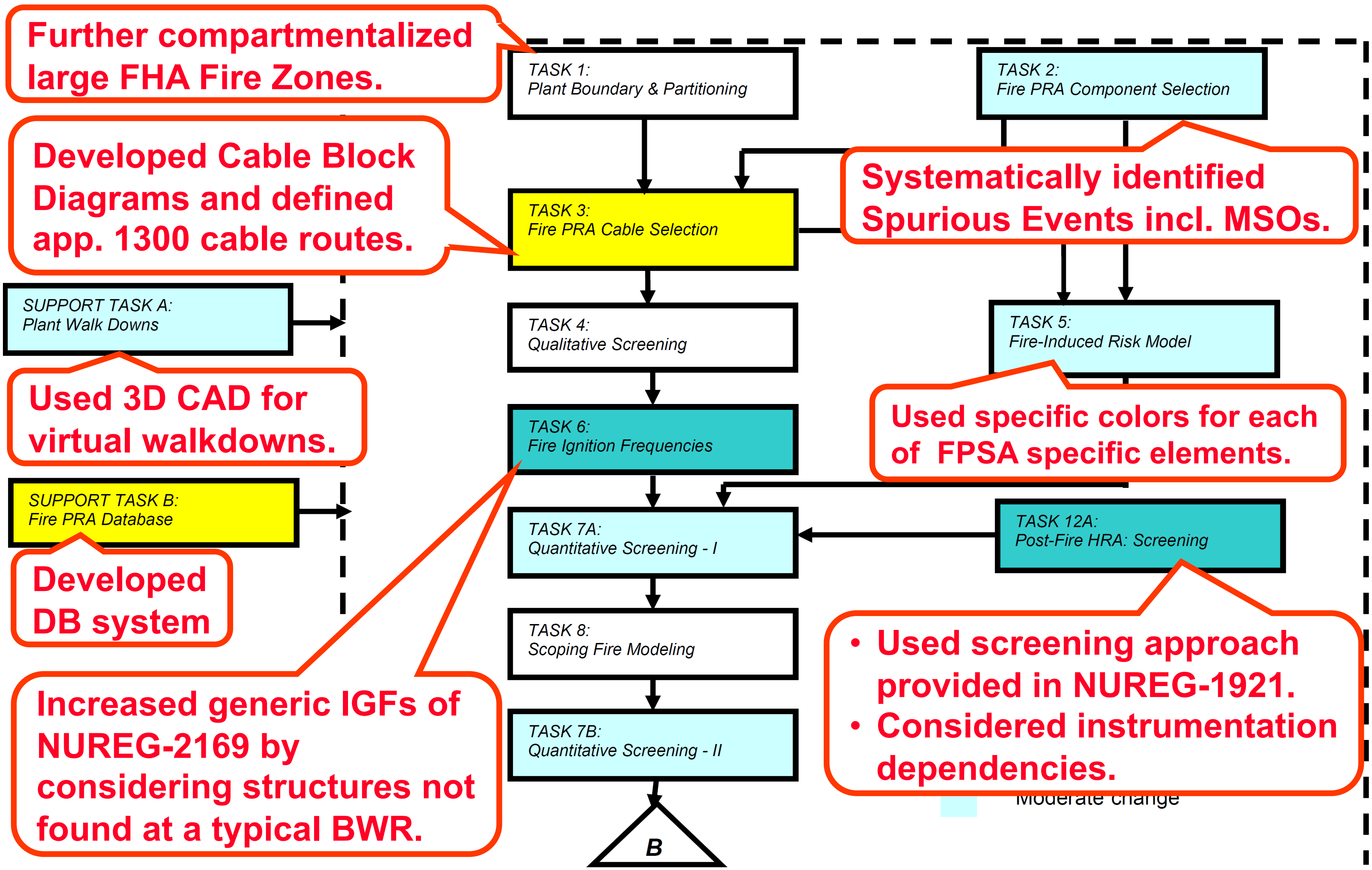


NUREG/CR-6850 Vol.1, Figure 1

# 2. Approaches to Generic Design FPSA

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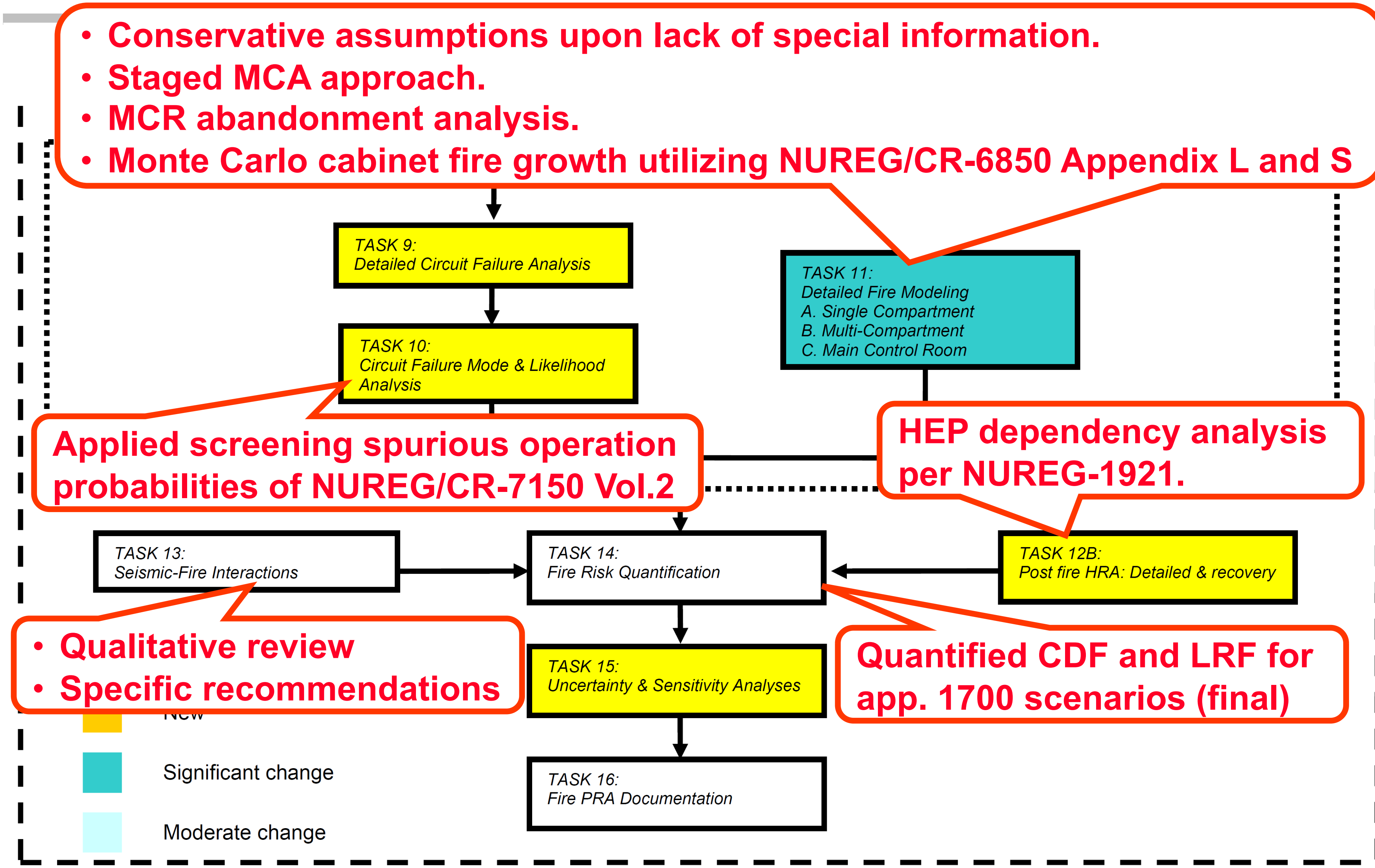


NUREG/CR-6850 Vol.1, Figure 2

# 3. Approaches to Reduce Conservatisms

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## Task 5 - Refinement of Plant Response Model

- Updated success criteria and event trees.
- Credited additional functions and interlocks.
- Credited more instrumentations supporting operator actions.

## Task 5/ Task 10 - Spurious Operation Duration Analysis

- Applied to SOVs for SRVs and MSIV etc.
- Determined key timings of clearing spurious operation from supporting thermal-hydraulic analyses
- Introduced conditional failure probabilities to clear spurious operations, based on NUREG/CR-7150 Volume 2.

# 3. Approaches to Reduce Conservatisms

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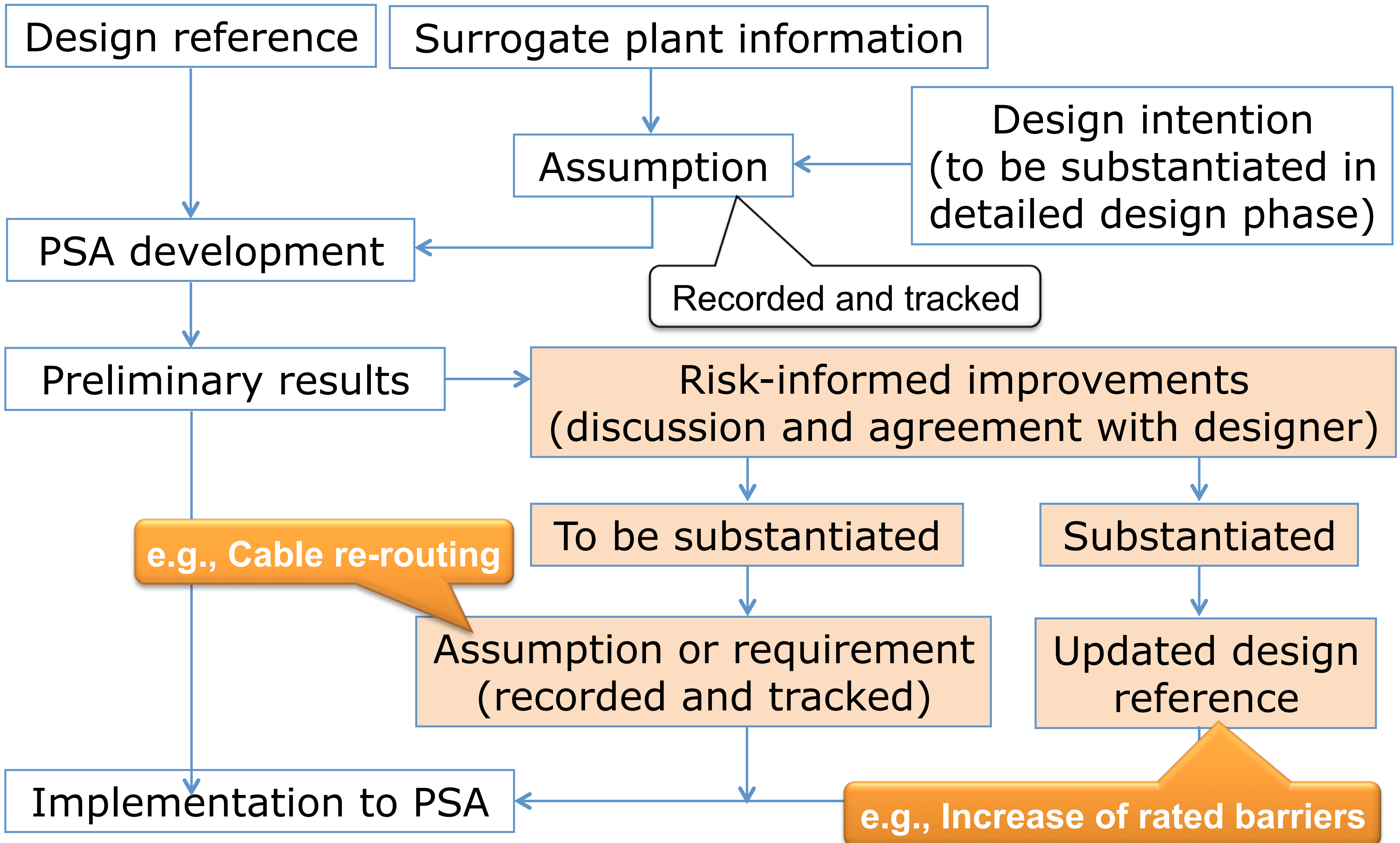
## Task 11 – Iterative Refinements of Detailed Fire Modelling

- Screened specific ignition sources for a POS if justified to be de-energized by interlocks and/or strict administrative control.
- Shifted center of ZOI if secondary combustible dominated HRR.
- Refined selection of critical cables vs. non-critical cables.
- Refined treatment of non-rated boundaries
- Used realistic flame propagation rate (Section R.4.1.2 of NUREG/CR-6850) for cables meeting appropriate standards.
- Applied CFAST modelling for risk significant MCA scenarios to consider oxygen depletion, in addition to FDTs (NUREG-1805).

## Task 12 – Refined FPSA specific HEP multiplier

- For risk significant HEPs
- Justified by Tabular Task Analysis and Human Error Analysis

# 4. Risk-informed Improvements during FPSA Development




More information found in:

*Y. Ishiwatari, et al., "Risk-Informed Design for UK ABWR Project", Int. Conf. on Topical Issues in Nuclear Installation Safety, IAEA CN-251, Vienna, Austria, June 2017.*

# 5. Results and Insights - Reactor at Power

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**Same order of CDF as the internal events by model refinements as much as practically feasible given generic design progress and project schedule, as well as risk informed improvements**

| | CDF (/y) | Contribution to Fire at Power CDF | LRF (/y) | Contribution to Fire at Power LRF |
|---|---|---|---|---|
| Task 11a Detailed Fire Models | 2.09E-07 | 42.2 % | 1.78E-07 | 67.2 % |
| Task 11b Main Control Room | 7.82E-09 | 1.6 % | 1.95E-09 | 0.7 % |
| Task 7 Whole Room Damage | 8.24E-08 | 16.6 % | 3.42E-08 | 12.9 % |
| ALL Single Compartment | 2.99E-07 | 60.4 % | 2.14E-07 | 80.8 % |
| Task 11c MCA Type 1* | 5.38E-09 | 1.1 % | 3.97E-09 | 1.5 % |
| Task 11c MCA Type 2* | 2.12E-08 | 4.3 % | 6.13E-09 | 2.3 % |
| Task 11c MCA Type 3* | 7.15E-08 | 14.4 % | 1.18E-08 | 4.4 % |
| Task 11c MCA Type 4* | 9.82E-08 | 19.8 % | 2.90E-08 | 10.9 % |
| ALL Multi Compartment | 1.96E-07 | 39.6 % | 5.08E-08 | 19.2 % |
| At Power Fire Total | 4.95E-07 | - | 2.65E-07 | - |
| *Internal Events at Power (for comparison)* | 2.27E-07 | - | 5.20E-08 | - |

*Type 1: Impacting temperature sensitive equipment in exposed PAUs by HGL.
*Type 2: Not producing a damaging HGL in the exposing PAU but potentially impacting the opposite side of a non-rated barrier by plume or radiant heat.
*Type 3: Producing a damaging HGL in the exposing compartment, and potentially impacting a fire zone surrounded by fire barriers.
*Type 4: Associated Type 3 scenario with further impact on adjacent fire zone with evaluated Barrier Failure Probability.

# 5. Results and Insights - Reactor at Power

- **Reactor Building electrical rooms** as the highest contributors, in large part due to existence of a large number of ignition sources (cabinets) and critical cables.
- Backup Building (B/B), Control Building (C/B) and Turbine Building (T/B) as relatively low contributors, which implied the effectiveness of fire rated boundaries additionally introduced for defense-in-depth and per **risk-informed recommendations.**
- **Low contribution from MCR** fire scenarios due to:
  - ✓ Class 1 digital C&I controller separated from the MCR fire zone
  - ✓ Two remote shutdown rooms and a B/B control panel room
- Accident Class AE (Large LOCA with loss of injection) as the highest contributor to CDF due to the high contribution from the MSO scenarios involving **spurious operations of more than 7 SRVs**.
- Relatively high LRF to CDF ratio (> 50 %), mainly due to the contribution from the Accident Classes involving containment failure / bypass.

# 5. Results and Insights - Reactor at Power

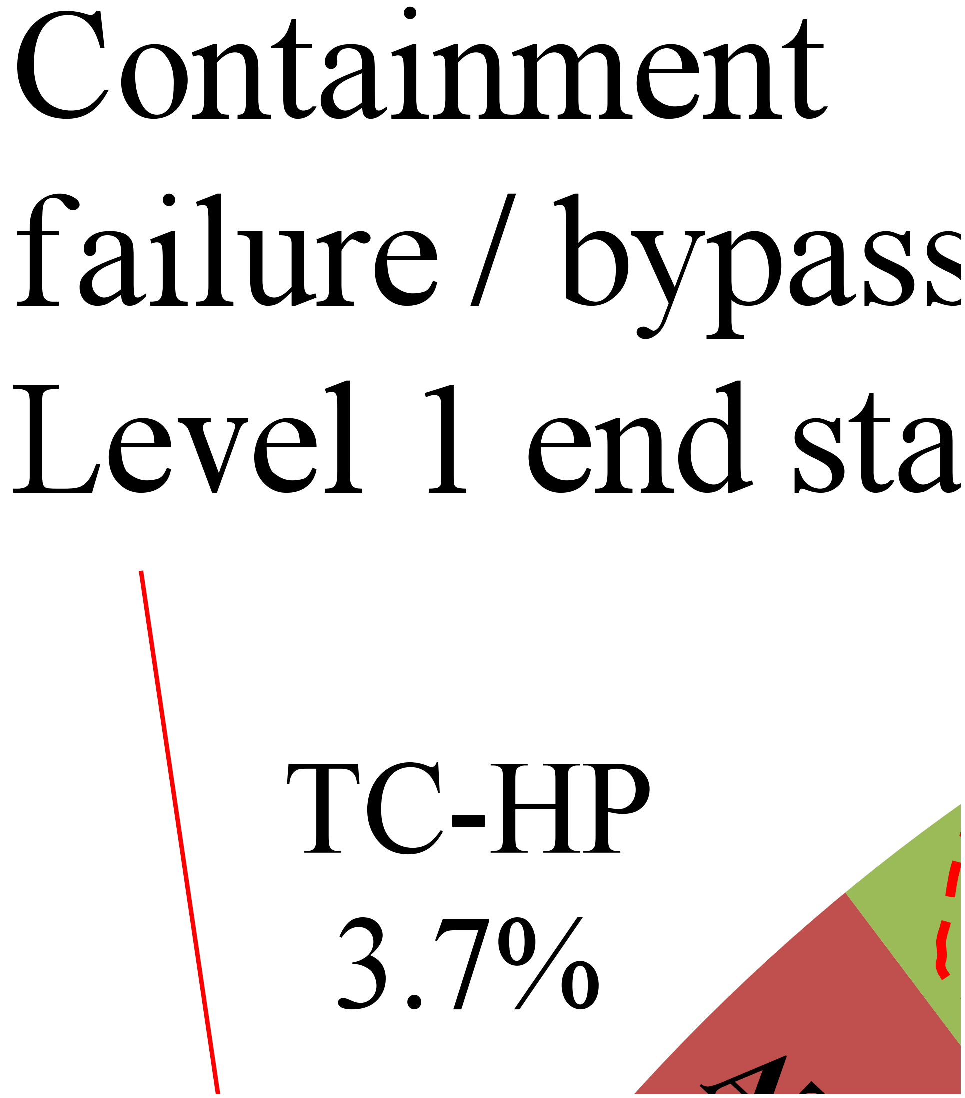


**Figure 2: Contribution of Accident Class to CDF**

# 5. Results and Insights - Scoping Analyses of Shutdown POSs

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**Conservative scoping analysis of representative POSs (selected from plant conditions and internal events PSA insights) showed insignificant fire risk compared to at Power.**

**POS C: Transition to closed condition of Containment/RPV heads with Divisions 1 and 3 in maintenance**

**POS B-2: Full water level in reactor well and gate open with Divisions 1 and 3 in maintenance**

| | FDF (/y) of POS C | Contribution to FDF of POS C | FDF (/y) of POS B-2 | Contribution to FDF of POS B-2 |
|---|---|---|---|---|
| ALL Single Compartment | 1.68E-08 | 38.2 % | 6.58E-09 | 33.8 % |
| ALL Multi Compartment | 2.72E-08 | 61.8 % | 1.29E-08 | 66.2 % |
| Fire Total for each POS | 4.40E-08 | - | 1.95E-08 | - |
| Internal Events Shutdown (for Comparison) | 5.38E-08 | - | 1.46E-08 | - |

**Insights:**

- Small contribution of fires in **containment**.
- Need attention to **fire-induced spurious overfill** of reactor well and SFP and subsequent internal flooding.
- Larger contribution of T/B fires than at Power due to higher transient fire frequencies during shutdown per NUREG-2169.

# 5. Results and Insights - Scoping Analyses of SFP at Power

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**Conservative scoping analysis of SFP and sensitivity analyses showed insignificant fire risk compared to reactor at Power.**

| | FDF (/y) at Power | Contribution to Fire FDF of SFP at Power |
|---|---|---|
| ALL Single Compartment | 1.95E-07 | 32.7 % |
| ALL Multi Compartment | 4.03E-07 | 67.3 % |
| Fire Total for SFP at Power POS | 5.98E-07 | - |
| Internal Events SFP PSA (for comparison) | 3.17E-08 | - |

**Further reduction expected per sensitivity analyses, application of refinements performed for at Power FPSA, and the nature of SFP faults (e.g., large time margin to fuel damage).**

**Sensitivity Analyses:**

- Consider realistic time available for terminating fire-induced flooding – 15% reduction
- Consider realistic consequence from containment failure without core damage (Level 1 PSA success sequences) – 20% reduction
- Remove FDF which is double-counting the LRF of reactor – 30% reduction

# 6. Peer Reviews

- Independent Peer Review team organized by US industry experts
- NEI 07-12 process
- Multiple in-process reviews and follow-on reviews
- Part 4 of ASME/ANS RA-Sb-2013
- UK Technical Assessment Guides (TAGs) for PSA/HRA
- Met large part of requirements for Capability Category II or III of ASME/ANS RA-Sb-2013

More information found at
*D. Henneke, et al., The Use of Comprehensive In-Process Peer Reviews in Support of the UK ABWR PSA Generic Design Assessment Process, PSA2017, Pittsburgh, PA, September 24-27, 2017.*

## 7. Additional Risk-informed Activities

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**Risk-informed improvements recommended from FPSA**

1. Systematic identification of risk-reduction options by
   - PSA result / sensitivity analyses / assumptions / uncertainties
   - Involvement of designers/operators
2. Examination of risk-reduction options
   - Reasonably practical or grossly disproportionate?
   - Phase of implementation?
3. Disposition: either of
   - No further options reasonably practicable
   - Implemented into design
   - To be assessed in later design phase

**e.g., use closed duct for risk-significant raceway, de-power specific valve**

**Use of FPSA to support various decision making**

**e.g., change in layout (tray, panel), ignition source quantity**

More information found in:
*Y. Ishiwatari, et al., "Risk-Informed Design for UK ABWR Project", Int. Conf. on Topical Issues in Nuclear Installation Safety, IAEA CN-251, Vienna, Austria, June 2017.*

# 8. Conclusions

- FPSA of UK ABWR generic design encountered challenges unique to applying the existing guidance.
- Challenges were overcome by **initial simplified approaches** followed by **model refinements** as well as **risk-informed improvements**.
- It was possible to demonstrate that the internal fire risk of UK ABWR was reduced **as low as reasonably practicable at the generic design phase**.
- Adequacy of FPSA was examined by Peer Review.
- It is intended to demonstrate further risk reduction in the later project phase based on more detailed design / operational information and continued risk-informed improvements.

# END Thank you for your attention.

Acknowledgements

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