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Nuclear Safety Enhancements based on Daiichi Forensics Information

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DOE Forensics Effort Coordinated with TEPCO D&D Activities

Objectives:

- Develop consensus US input for *high priority time-sequenced examination tasks and supporting research* that can be completed with minimal disruption of TEPCO Decommissioning and Decontamination (D&D) activities
- Evaluate obtained information to:
  - Gain a better understanding of events that occurred in each unit at Daiichi
  - Gain insights to reduce uncertainties in predicting phenomena and equipment performance during severe accidents
  - Provide insights beneficial to TEPCO Phase 2 Fuel Debris Retrieval Evaluations
  - Confirm/improve guidance for severe accident prevention, mitigation, and emergency planning
  - Periodically update/refine original information requests
- Facilitate implementation of Japan-led international research efforts to support D&D

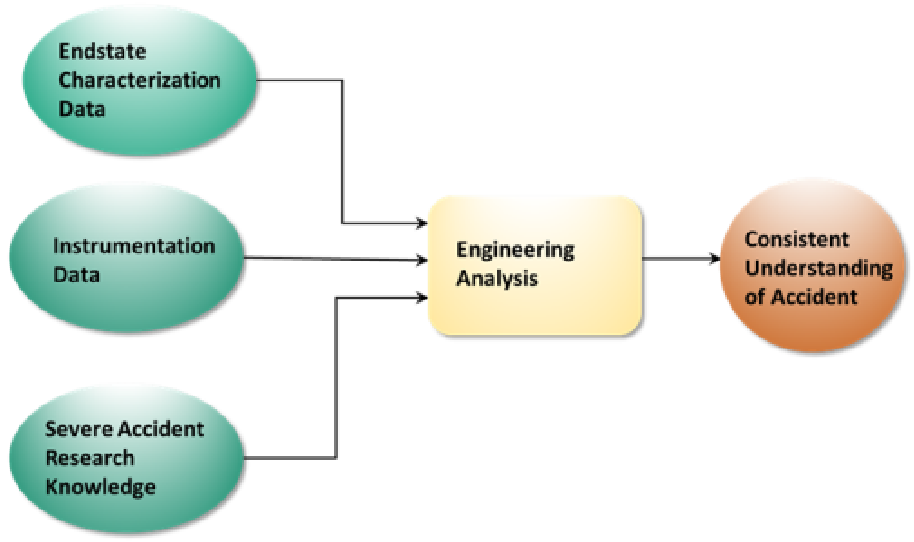
Motivations:

- Provides US access to prototypic data from BWR core melt events with distinct accident signatures
- Provides Japan access to US expertise in plant operations, severe accident modeling & testing, and defueling & cleanup

Image Courtesy of ANS

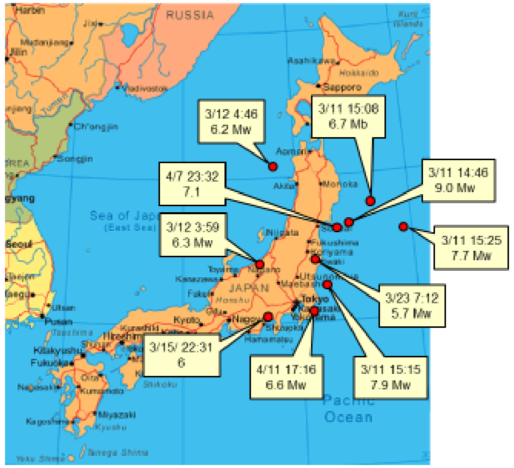
FY2021 report publicly available (https://doi.org/10.2172/1773089)
FY2022 report with updated information need requests (March 2022).

Similar Foreniscs Approach Applied for Post-Accident Investigations

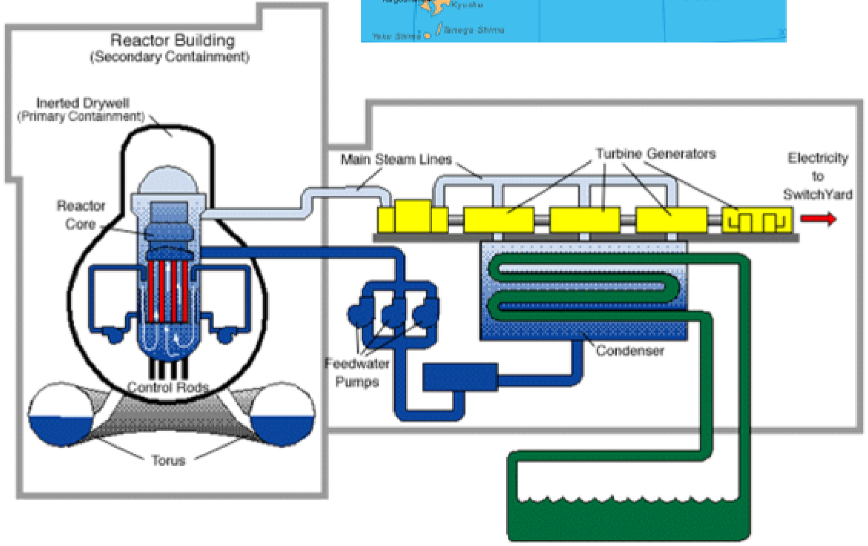


- **Process relies on instrumentation data, post-accident examinations, existing severe accident knowledge, and engineering analyses**
- **Efforts initially focused on stabilizing reactors and associated structures before focusing on cleanup**
- **Key to prioritize activities, emphasizing those that:**
  - **Minimize future radiation releases and site hazards,**
  - **Ensure safe and efficient D&D, and**
  - ***As resources allow*, reduce uncertainties related to accident progression and reactor safety enhancement.**
- **Most high priority information *desired* for reactor safety insights *required* for D&D**

Fukushima Daiichi (1F) Accidents on March 11, 2011



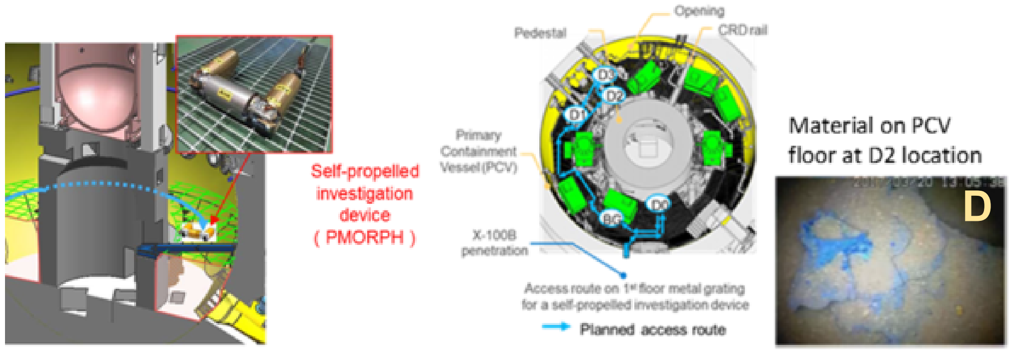
Graphics Courtesy of TEPCO



- **Design and Operating Considerations:**
  - Plant contained six GE BWRs (1F1, 1F2, and 1F3 operating; 1F4, 1F5, and 1F6 in outage)
  - International "Titanic" mentality about likelihood of multi-unit severe accidents and multiple external events

- **Event synopsis**
  - Multiple large seismic events with peak ground accelerations well above scram setpoints
  - Tsunami flooding nearly three times design basis flooding height
  - Loss of off-site power due to initial earthquake followed by loss of DC power from flooding and battery depletion
  - Progression differed in each unit; uncertainties in the timing and success of mitigating measures.
  - Major release of radioactive material: up to 14 E6 Ci (500 E15 Bq) ^{131}I; International Nuclear and Radiological Event Scale (INES) Level 7 event
  - Approximately 150,000 persons within 19 miles evacuated

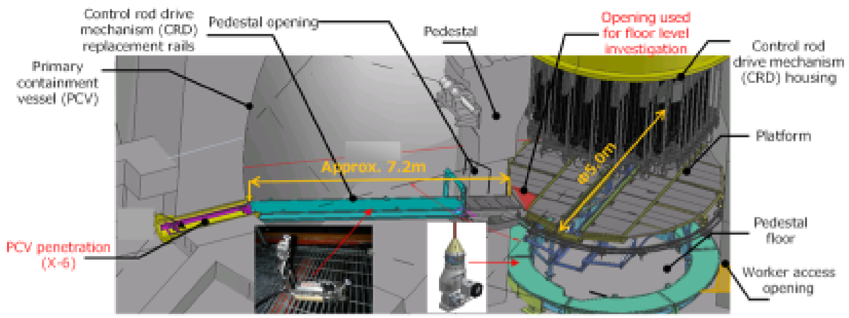
1F1 Accident Progression Insights

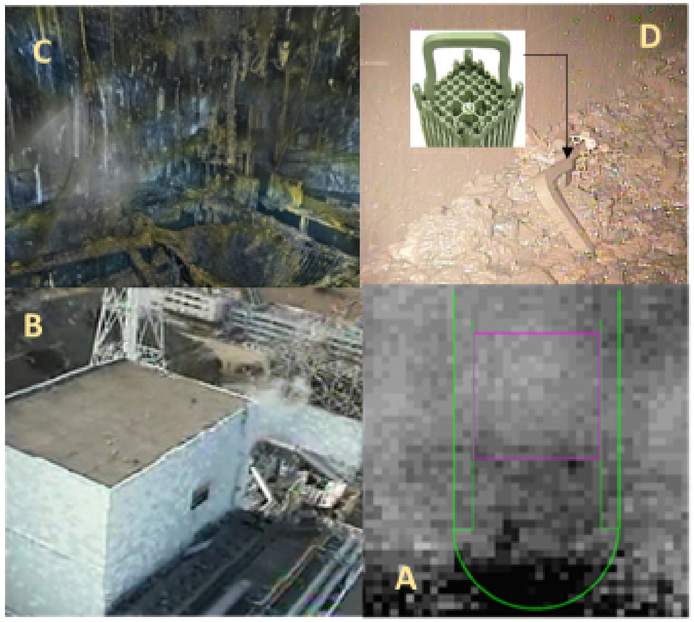


Graphics Courtesy of TEPCO and Fukushima Central Television (FCT)

- BWR/3 rated at 460 MWe in Mark I containment with Isolation Condensers (ICs) for decay heat removal
- Seismic accelerations, tsunami flooding, and station blackout hindered accident mitigation:
  - IC operation
  - Instrumentation availability and calibration
  - Venting
- Plant data, radiation surveys, calculations, and images from muon tomography and robotic inspections indicate:
  - Significant fuel heatup and relocation
  - High temperatures/pressures led to Primary Containment Vessel (PCV) leakage, hydrogen release to reactor building
  - Reactor building combustion led to significant damage to the reactor building and missile shield
  - Reactor Pressure Vessel (RPV) failure and significant relocation to PCV (initial examinations limited to locations external to pedestal)
  - Extent of core concrete interaction uncertain

1F2 Accident Progression Insights

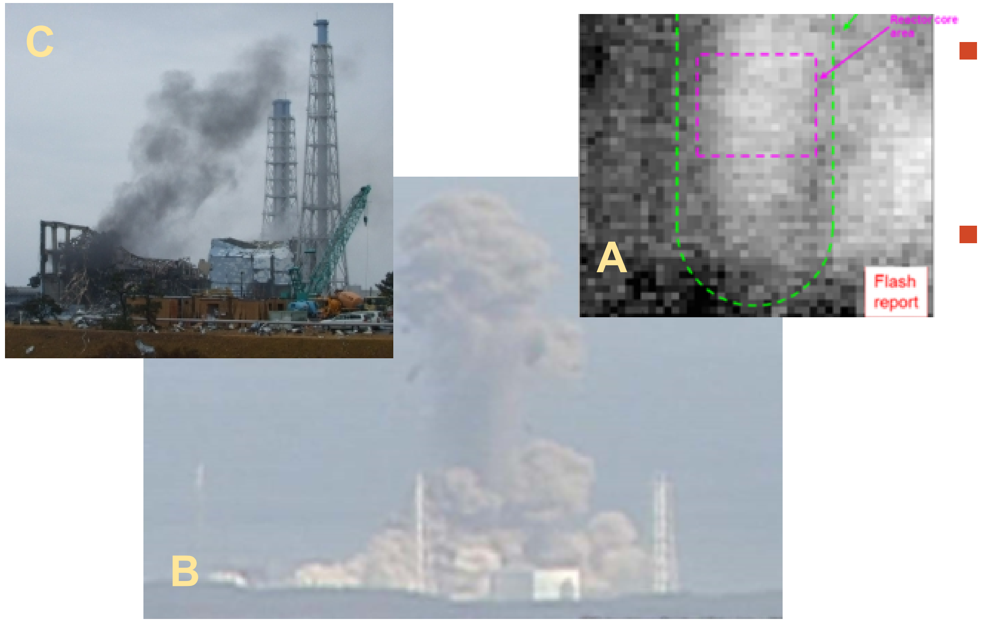


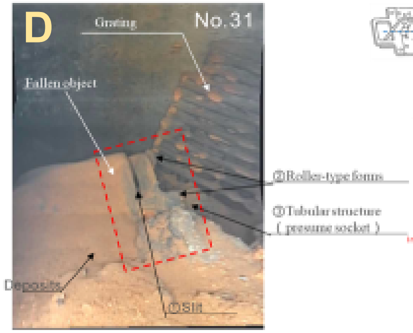


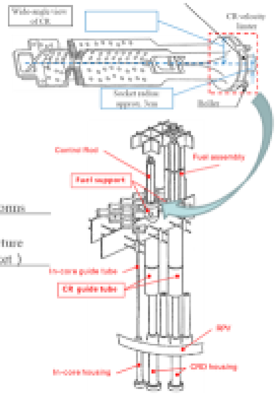
Graphics Courtesy of TEPCO

- BWR/4 rated at 784 MWe in Mark I containment with Reactor Core Isolation Cooling (RCIC) system for decay heat removal
- Available information indicates accident progression less advanced:
  - RCIC operation continued for nearly 3 days.
  - H_2 explosion precluded due to “unintentional” early Reactor Building (RB) venting
  - RPV failure with ex-vessel debris holdup on structures
  - Relocation and spreading into PCV; muon tomography indicates ~80% of fuel may remain in the RPV lower plenum
  - PCV water level lower than in 1F1
  - 1F1 and 1F3 explosions delayed recovery efforts

1F3 Accident Progression Insights





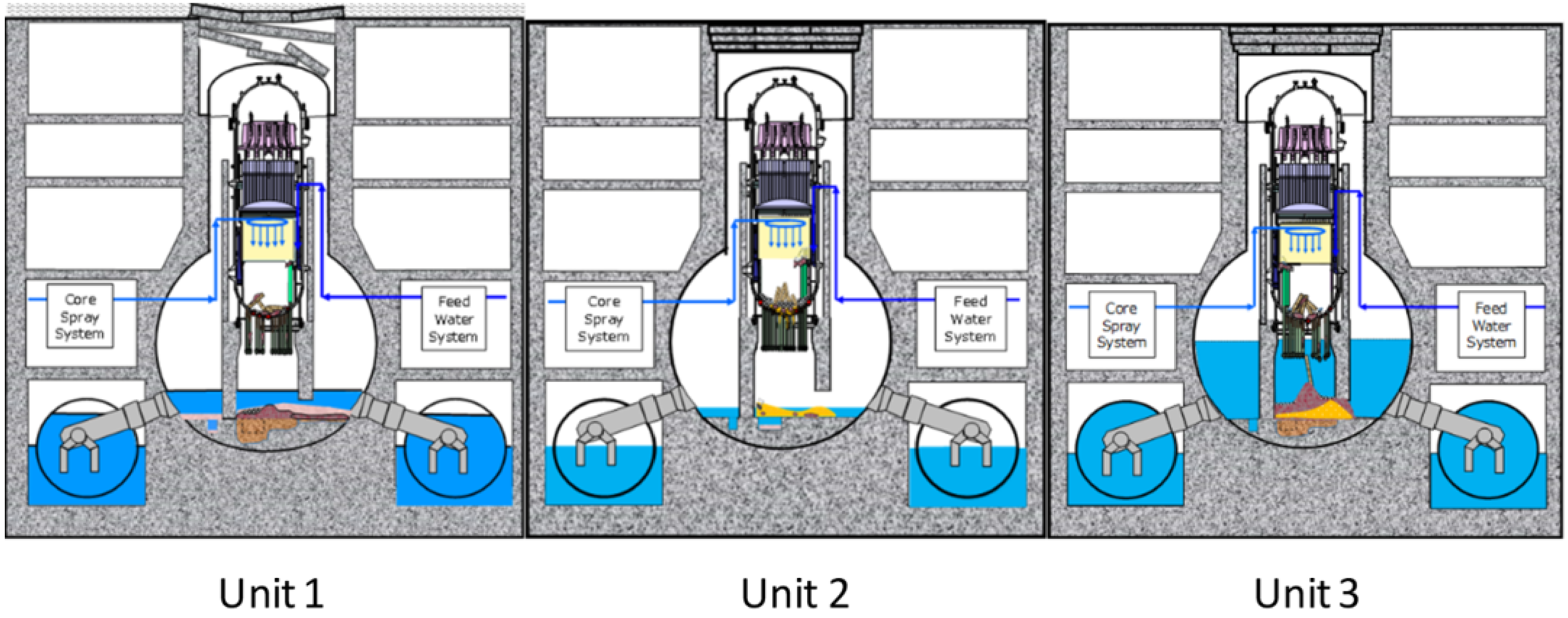


Graphics Courtesy of TEPCO, IRID, and FCT

- BWR/4 rated at 784 MWe in Mark I containment with RCIC system.
- Available information indicates accident progression more advanced than in Unit 2:
  - RCIC operation continued for less than 1 day
  - Significant fuel heatup and relocation
  - High temperatures/pressures led to PCV leakage and a multi-stage explosion
  - Venting through Standby Gas Treatment System (SGTS) led to Unit 4 explosion
  - RPV failure with significant relocation and spreading into PCV
  - PCV water level much higher than in Units 1 and 2

Important *Interim* Technology-Neutral Lessons Learned

- **Root cause(s) for losing control of cooling and radiation release**
  - Long-term station blackout due to earthquake and tsunami flooding
  - Inadequate resources, training, and procedures for extreme conditions
  - Inadequate safety culture and regulatory oversight
- **Perception of *known* challenges**
  - Concurrent external hazards
  - Multi-unit effects
  - Severe accidents
  - Long term loss of on-site and off-site recovery systems
  - External stakeholder communication



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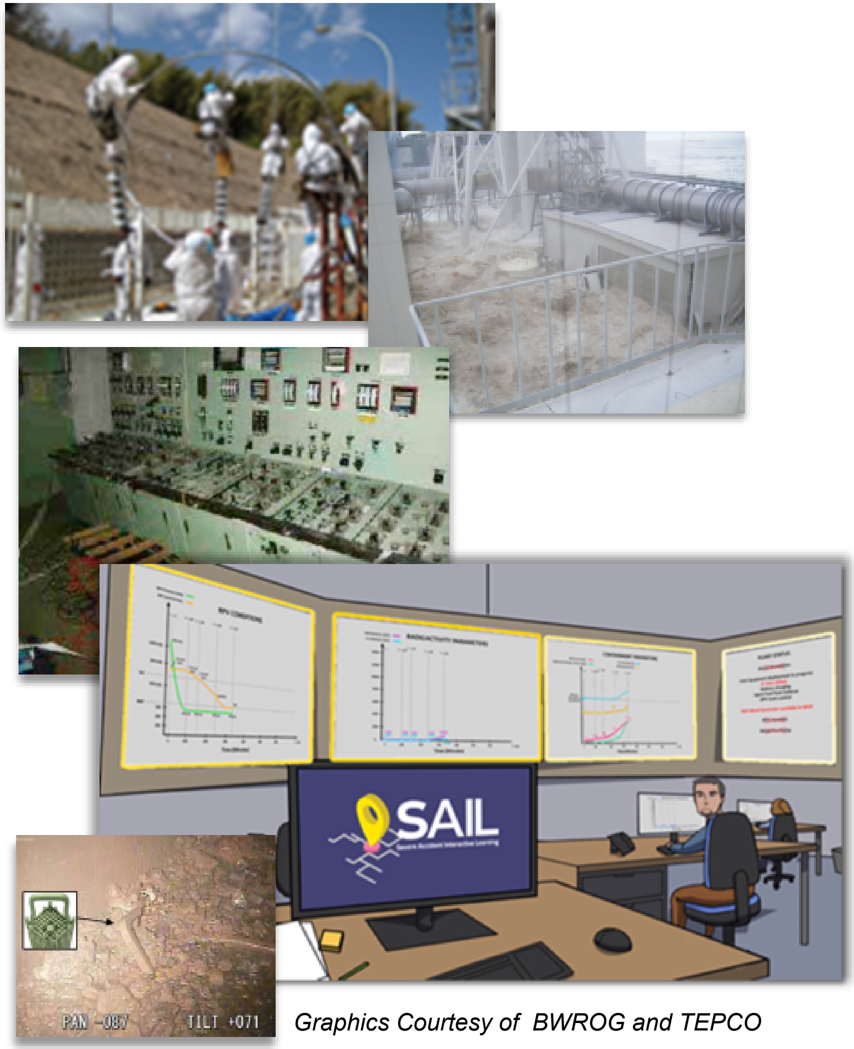
Actions to Address Interim Lessons Learned (continued)



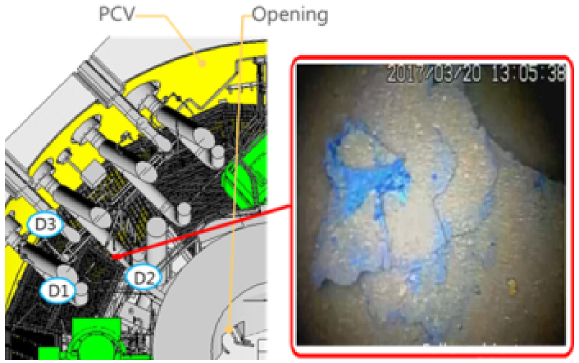
Graphics Courtesy of BWROG and TEPCO

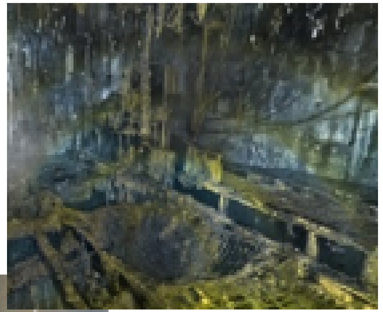
Actions to Prevent and/or Mitigate Fuel Damage for BDBEs

- U.S. Industry Diverse and Flexible Coping (FLEX) Program
  - Plant sites maintain additional equipment for water injection, power restoration, and debris removal
  - Similar equipment at two national response centers
- Improved spent fuel pool (SFP) level water level instrumentation and strategies to address challenges to SFP cooling
- Hardened containment wetwell vent (BWR I and II containments)
- Alternate venting and water addition strategies
- Revised procedures and guidance and updated training

Images, such as the fuel assembly handle observed in 1F2 PCV examinations, provide critical insights regarding RPV failure in new BWROG computer-based Severe Accident Interactive Learning (SAIL) training and guidance.

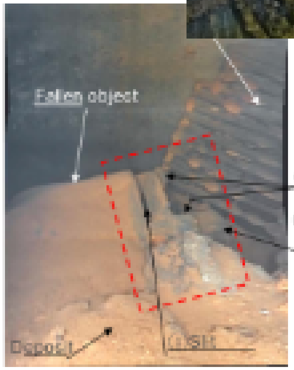
Actions to Address Interim Lessons Learned (continued)

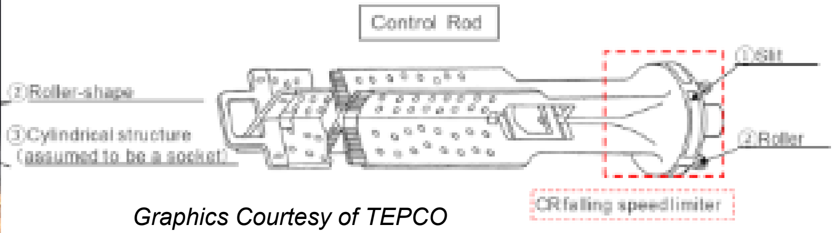




Updated Severe Accident Systems Analysis Code Models

- Improved detail in BWR primary system thermal hydraulic models
- A new containment fluid stratification model to allow simulation of suppression pool phenomena inferred from 1F3 PCV pressure data
- RCIC system Terry™ Turbine and HPCI system performance models
- Ex-vessel relocation and debris coolability modeling improvements to reflect images of holdup and relocation of debris and relocated components observed in 1F investigations
- New corium spreading and molten core concrete interaction erosion models inferred from images obtained during 1F1 examinations [OECD Reduction of Severe Accident Uncertainties (ROSAU) testing underway at ANL to support model development]
- Improved models used to optimize BWROG updated guidance efforts, identifying actions and decisions having the most risk impact (e.g., venting and water addition)



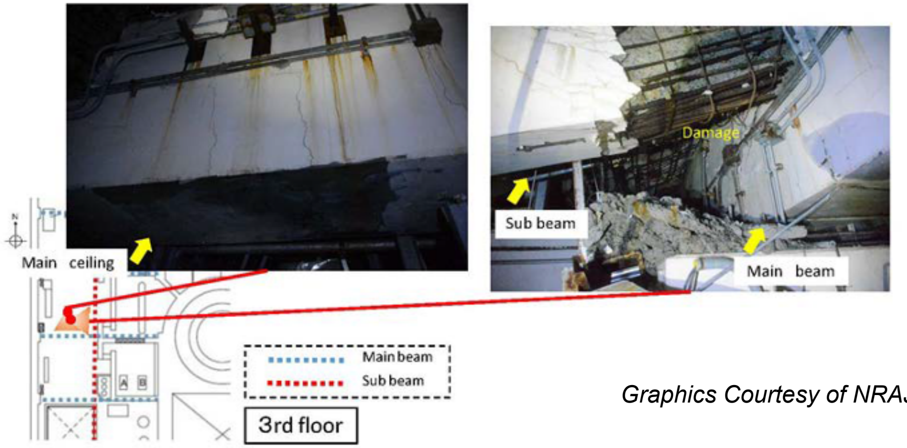


Graphics Courtesy of TEPCO

Many updates motivated by needs identified in 1F simulations and examinations

Current Areas of Emphasis to Gain New Insights

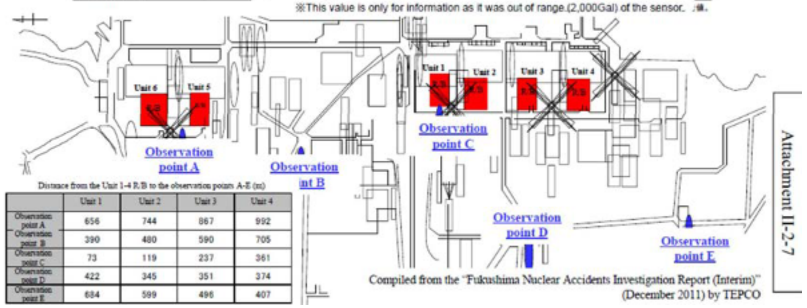
- **Combustible gas phenomena**
  - Seismometer data comparing energy exerted by 1F1 and 1F3 explosions
  - High resolution images of 1F1 and 1F3 explosions
  - Damage within 1F1, 1F3, and 1F4 buildings



Graphics Courtesy of NRAJ

	Unit 1 Explosion March 12 at 15:36			Unit 3 Explosion March 14 at 11:01			Unit 4 Explosion March 15 at 6:12		
	South-north direction	East-west direction	Vertical direction	South-north direction	East-west direction	Vertical direction	South-north direction	East-west direction	Vertical direction
Observation point A	81	52	120	14	15	21	2	2	2
Observation point B	284	129	138	45	18	28	4	3	3
Observation point C	2,320※	2,392※	1,956	115	158	490	11	9	6
Observation point D	102	91	231	36	51	173	6	7	11
Observation point E	39	22	26	26	24	30	5	5	11

※This value is only for information as it was out of range.(2,000Gal) of the sensor.

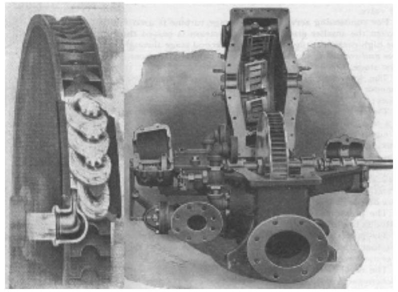


Distance from the Unit 1-4 R/B to the observation points A-E (m)

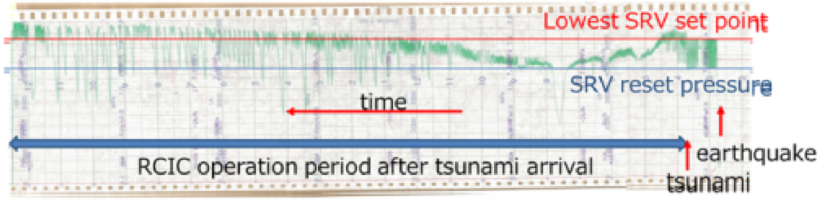
	Unit 1	Unit 2	Unit 3	Unit 4
Observation point A	656	744	867	992
Observation point B	390	480	590	705
Observation point C	73	119	237	361
Observation point D	422	345	351	374
Observation point E	684	599	496	407

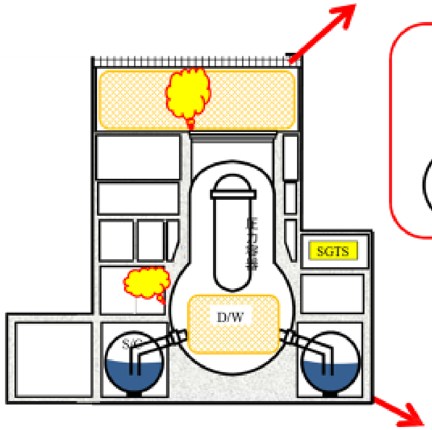
Compiled from the "Fukushima Nuclear Accidents Investigation Report (Interim)" (December 2011) by TEPCO

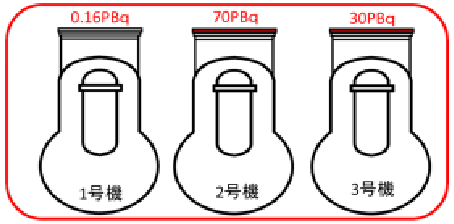
Current Areas of Emphasis to Gain New Insights (continued)

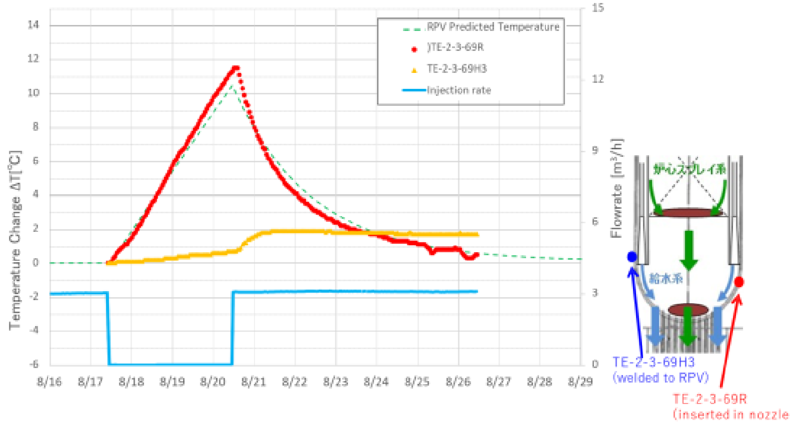


- **System and component performance**
  - Combustible gas (and fission product) transport during venting based on 1F1/1F2 SGTS contamination levels
  - 1F1, 1F2, and 1F3 contamination levels to infer fission product transport and debris location
  - 1F2 and 1F3 SRV and rupture disk operation based on available plant instrumentation data and contamination levels
  - 1F3 RCIC operation based on available plant instrumentation data and insights gained from bi-lateral Japan/U.S. Terry Turbine™ testing program
  - 1F1, 1F2, and 1F3 coolant suspension tests to detect PCV leakage locations and potential to reduce coolant water injection (reducing contaminated water generation and potential for seismic-induced structural failures)



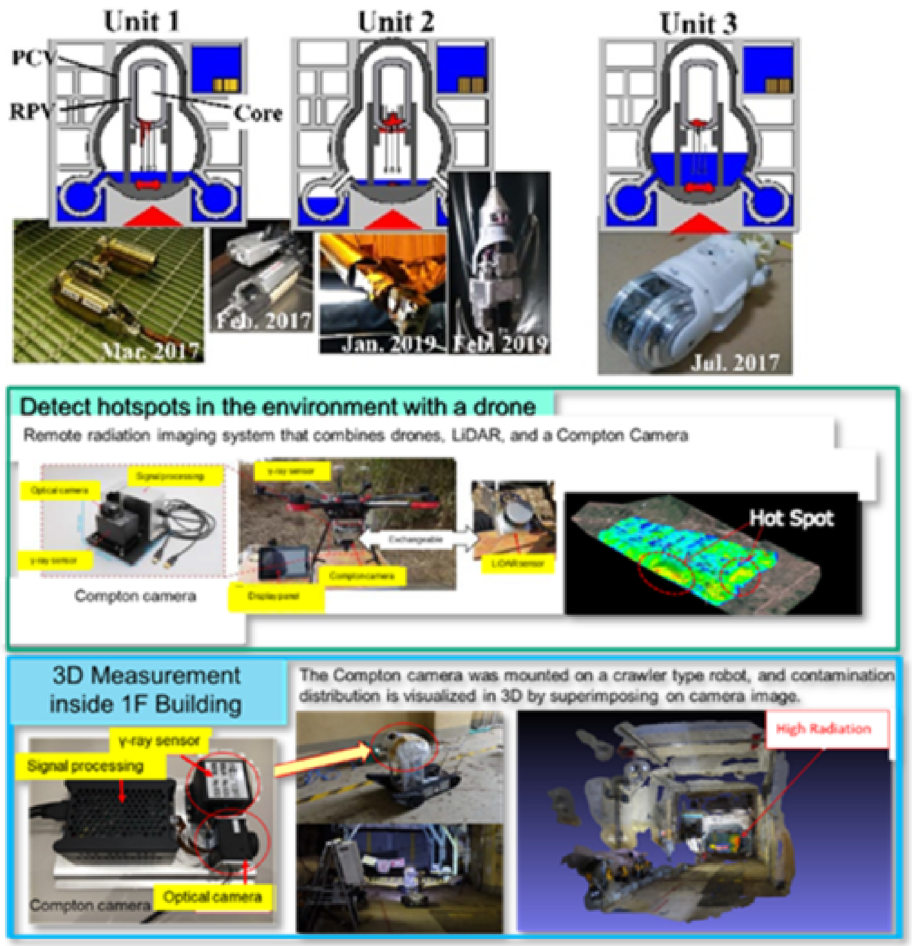






Graphics Courtesy of TEPCO , SNL, and TAMU

Current Areas of Emphasis to Gain New Insights (continued)



Graphics Courtesy of TEPCO, IRID, and JAEA

- **New technologies facilitate 1F D&D**
  - Muon tomography
  - Special-purpose robots, drones, and Unmanned Aerial Vehicles (UAVs)
  - Portable gamma-ray imaging camera
  - Infrared thermography
  - Real-time monitoring with 2D or 3D visualization of radiation levels and temperatures
  - Plastic scintillation fiber monitors
  - Centralized data system to optimize worker exposure
- **These new technologies offer the potential to improve plant operations and maintenance**

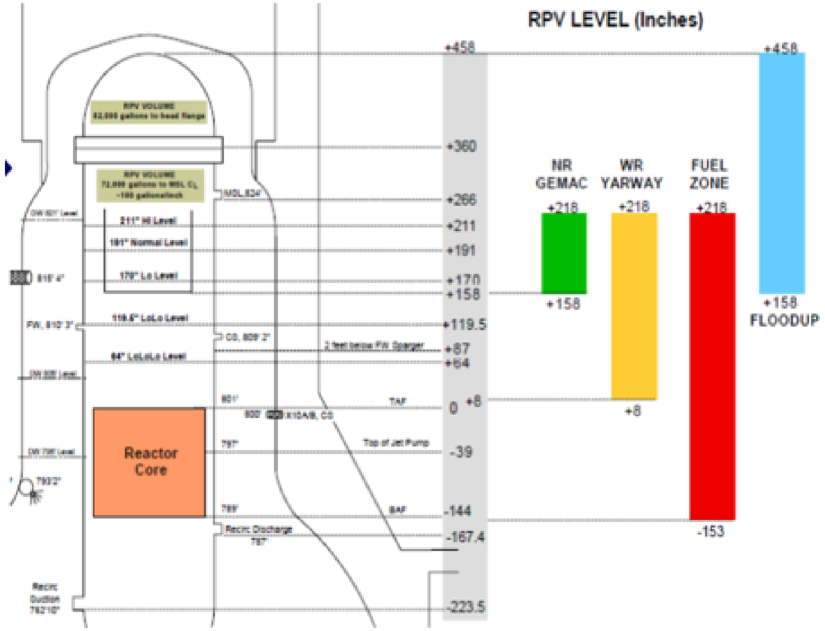
Implemented Actions Enhanced Recovery during 2020 DAEC LOOP

Graphic Courtesy of BWROG

Plant Status on August 10, 2020

- DAEC was operating at ~80% power; coasting down to end of cycle (EOC). This power limited the cycling of a turbine control valve that occurs around ~84% power
- A dry cask storage campaign under way in the spent fuel pool (SFP); estimated time for the SFP to boil was 64 hours
- Some essential/non-essential equipment status:
  - Diesel driven fire pump inoperable due to maintenance; drywell cooler degraded
  - Low pressure coolant injection "B" train inoperable due to testing prior to the event (not being tested during the event and available for use if needed)
  - Two control rods fully inserted to suppress a fuel leaker

Implemented Actions Enhanced Recovery during 2020 DAEC LOOP (continued)



Graphic Courtesy of BWROG

Response to LOOP Event

- Derecho approached DAEC with peak windspeeds between 100 and 130 mph; Loss of Offsite Power (LOOP) at 12:49 on August 10, 2020.
  - A generator load reject occurred, tripping the turbine and causing reactor scram
  - ‘A’ and ‘B’ emergency diesel generators automatically started; supplying power to safety related busses
  - Recirculation pumps tripped (LOOP prevented restart)
  - RPV water level initially lowered rapidly to Level 2 due to loss of feedwater.
  - RCIC and HPCI systems automatically initiated and restored RPV water level until Level 8 reached, causing both systems to trip.
- Operators placed both systems in manual control and intentionally increased water levels to promote natural circulation cooling.
  - Using revised guidance, operators quickly established pressure control using main steam line drains and RCIC.
  - Revised Level 8 trip inhibit values allowed operators to maintain RCIC and HPCI system operation and reduce SRV cycling
- Shutdown cooling started at 22:30 on August 10, 2020; cold shutdown established at 02:30 on August 11, 2020.

Implemented Actions Enhanced Recovery during 2020 DAEC LOOP (continued)

Graphic Courtesy of TAMU

Guidance Informed by 1F2 Response and TTEXOB Testing

- 1F2 RCIC system operation continued for nearly 3 days
  - Prior to this, it was generally assumed that loss of DC power would flood the steam line and disable the turbine
- The goal of BWROG-led Terry^{TM} Turbine Expanded Operating Band Project (TTEXOB) Project is to expand and define actual operating limitations of Terry^{TM} turbine systems (i.e., RCIC/TDAFW)
  - US Nuclear Industry, USDOE, and IAE (Japan) are major stakeholders of international consortium supporting project
  - System and component testing conducted at Texas A&M university
  - Testing supported by modeling (MELCOR, MAAP, and SAMPSON)
  - New RCIC models benchmarked using Tennessee Valley Authority data in which RCIC system ran on April 27, 2011 after a tornado

Implemented Actions Enhanced Recovery during 2020 DAEC LOOP (continued)

Graphic Courtesy of BWROG

Lessons Learned

- The post Fukushima EOP changes to Level 8 trip inhibits important in maintaining RCIC and HPCI system performance and in reducing SRV cycling
- The DAEC event re-emphasizes need for symptom-based procedures for EPG/SAGs and FLEX
- Procedures and proficiency important to restoring systems out of service for testing or maintenance and for returning failed systems to operation during a LOOP
- Modeling of event assumptions needs to be consistent with actual plant operations or conditions
- RCIC testing provides specifics about turbine and pump operation that improve modeling
- Plant transient response was as expected and agreed with simulator training for LOOP response

Closing Remarks

- **Insights, which continue to be obtained from Japan's 1F D&D efforts, offer the international community the opportunity to enhance global nuclear safety**
- **The DOE-sponsored U.S. Forensic Effort has worked to provide input to Japan regarding future D&D activities, to better understand the progression of the accidents, and (where possible) to reduce uncertainties in our understanding and modeling of severe accident progression**
- **The U.S. nuclear enterprise has used insights gained from the Forensics Effort and improved severe accident models to update guidance and training for severe accident prevention, mitigation, and emergency planning**
- **More insights and reactor technology-neutral lessons are expected as 1F D&D progresses**

Actions Implemented to Prevent Fuel Damage based on Examinations and Lessons Learned

Action	Main Elements	U.S. NRC Regulations	Regulatory / Industry Guidance	Objective	Daiichi Lessons Learned Addressed
FLEX Strategies	• Installed Hardware • Portable Hardware • Off-site Response • Analysis • Staffing • Communications • Procedures • Training	Order EA-12-049 10 CFR 50.155(b)(1)	Regulatory Guide 1.226 NEI 12-06 Rev 4	Primary and alternate strategies to maintain core cooling, spent fuel cooling and containment integrity	• Diverse equipment • Diverse strategies • Off-site response equipment and capability • Standard connections to plant systems • Beyond Design Basis hazard protection
MSA Hazard Re-evaluations	Present day methods for: • Flooding • Seismic	Request for information pursuant to 10 CFR 50.54(f)	NEI 12-06 Rev 4	FLEX strategies remain viable under beyond design basis conditions	• Beyond Design Basis hazard protection for equipment storage • Beyond Design Basis hazard capability for strategies
SFP Level Indication	• Wide range level instrument	Order EA-12-051 10 CFR 50.155(e)	Regulatory Guide 1.227 NEI 12-02 Rev 1	Provide SFP level indication from normal water level to top of fuel racks	• Prioritization of event response
Hardened Containment Wetwell Vent	• Ease of operation • Functionality • Quality • Maintenance Procedures • Training	Order EA-13-109 BWR Mark I and II containments only	NEI 13-02 Rev 1	Minimize reliance on operator action, 1% rated steam flow capability, preservation of RCIC or HPCI system operation	• Containment Integrity - overpressure protection • Maximize steam driven injection capability

Actions Implemented to Mitigate Fuel Damage based on Examination Information and Lessons Learned

Action	Main Elements	US NRC Regulations	Industry Guidance	Objective	Daiichi Lessons Learned Addressed
FLEX Strategies include Enhancements to both BWR SAGs and PWR SAMGs on use of Alternative Water Sources and Use of Portable Equipment	• Installed Hardware • Portable Hardware • Off-site Response • Analysis • Staffing • Communications • Procedures • Training	Regulations do not extend to severe accident conditions; however, SAMG commitments have been added to ROP inspection criteria.	NEI 12-06 does not extend to severe accident conditions. NEI 14-01 Section 3.2 addresses SAMG maintenance.	Provide injection capability under severe accident conditions to stabilize core debris (SAWA) SAMGs will utilize any available equipment	• Severe accident conditions may impede plant access needed to mitigat core damage • Mitigation actions within the secondary containment need to be prioritized for completion before radiation levels become prohibitive fo secondary containment entry .
MSA for Hazard Re-evaluations	Present day methods for: • Flooding • Seismic	Regulations do not extend to SA conditions.	NEI 12-06 does not extend to SA conditions.	Injection capability remains viable under beyond design basis conditions	• Beyond design basis hazards may fail design basis equipment used to mitigate the effects of the beyond design basis hazard
SFP Level Indication	Wide range level instrument	Regulations do not extend to SA conditions.	NEI 12-02 does not extend to SA conditions.	Allow proper prioritization of response actions	• Design basis equipment may be inadequate to allow prioritization of equipment used for event response
SFP Strategies to Address Postulated Challenges to SFP Cooling	Same as FLEX but may be under severe accident conditions	Regulations do not extend to SA conditions.	Extension of NEI 12-06 and B.5.b vent capabilities	Continued cooling if irradiated fuel in the SFP	• SA conditions may impede plant access needed to mitigate fuel damage • Considerable time exists before SFP makeup required; however, mitigation actions within the secondary containment need to be prioritized for completion before radiation levels become prohibitive for secondary containment entry
Hardened Containment Vent	• Ease of operation • Functionality • Quality • Maintenance • Procedures • Training	Order EA-13-109 BWR Mark I and II containments only	NEI 13-02 Rev 1	Minimize reliance on operator action, 1% rated steam flow capability, withstand or prevent combustible gas detonation	• Containment Integrity - overpressure protection • Combustible gas control - controlled venting to atmosphere • Minimize off-site release