From: RYAN Tom (AREVA) [Tom.Ryan@areva.com]
Sent: Wednesday, January 29, 2014 3:00 PM
To: Wunder, George
Cc: HOTTLE Nathan (AREVA); GUCWA Len (EXTERNAL AREVA); UYEDA Graydon (AREVA); RANSOM Jim (AREVA); LEIGHLITER John (AREVA); WILLIFORD Dennis (AREVA); RYAN Tom (AREVA); ROMINE Judy (AREVA); DELANO Karen (AREVA); WILLS Tiffany (AREVA); BALLARD Bob (AREVA); NOXON David (AREVA); Eudy, Michael
Subject: Response to U.S. EPR Design Certification Application FINAL RAI 591, Chapter 19, Supplement 1
Attachments: RAI 591 Supplement 1 Response US EPR DC.pdf

George,

AREVA NP Inc. provided a response to the one question in RAI No. 591 on July 15, 2013.

The attached file, "RAI 591 Supplement 1 Response US EPR DC.pdf" provides a technically correct and complete final revised response to Question 19-370. The original response to RAI 591, with the added text from this Supplement 1, in addition to the changed pages in the U.S. EPR FSAR markups in this supplement, provide the full and complete Response to Question 19-370.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the revised response to RAI 591 Question 19-370.

The following table indicates the respective pages in the response document, "RAI 591 Supplement 1 Response US EPR DC.pdf," that contain AREVA NP's response to the subject question.

Question #	Start Page	End Page
RAI 591 — 19-370	2	2

This concludes the formal AREVA NP response to RAI 591, and there are no questions from this RAI for which AREVA NP has not provided responses.

Sincerely,

Tom Ryan
Manager, US EPR DCD
Regulatory Affairs
AREVA
7207 IBM Drive - CLT2B
Charlotte, NC 28262
Phone: 704-805-2643, Cell : 704-292-5627
Fax: 434-382-6657

From: WILLIFORD Dennis (RS/NB)
Sent: Monday, July 15, 2013 12:49 PM

To: Snyder, Amy
Cc: Ford, Tanya; phyllis.clark@nrc.gov; NOXON David (RS/NB); KOWALSKI David (RS/NB); ANDERSON Katherine (External AREVA NP INC.); DELANO Karen (RS/NB); LEIGHLITER John (RS/NB); ROMINE Judy (RS/NB); RYAN Tom (RS/NB); BUTLER Jen (EP/PE); GUCWA Len (External RS/NB)
Subject: Response to U.S. EPR Design Certification Application FINAL RAI 591, Chapter 19

Amy,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 591 Response US EPR DC.pdf," provides a technically correct and complete final response to Question 19-370.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the response to this question.

The following table indicates the respective pages in the response document, "RAI 591 Response US EPR DC.pdf," that contain AREVA NP's response to the subject question.

Question #	Start Page	End Page
RAI 591 — 19-370	2	2

This concludes the formal AREVA NP response to RAI 591, and there are no questions from this RAI for which AREVA NP has not provided responses.

Sincerely,

Dennis Williford, P.E.
U.S. EPR Design Certification Licensing Manager
AREVA NP Inc.
7207 IBM Drive, Mail Code CLT 2B
Charlotte, NC 28262
Phone: 704-805-2223
Email: Dennis.Williford@areva.com

From: Snyder, Amy [mailto:Amy.Snyder@nrc.gov]
Sent: Thursday, July 11, 2013 9:54 AM
To: ZZ-DL-A-USEPR-DL
Cc: Ford, Tanya; Segala, John; Mrowca, Lynn; Grady, Anne-Marie
Subject: U.S. EPR Design Certification Application FINAL RAI 591, Chapter 19

Attached please find the subject request for additional information (RAI). A draft RAI was provided to you on June 28, 2013. On July 2, 2013, you informed us that the draft RAI does not contain proprietary information and that the draft RAI is clear and no further clarification is needed. As result, the RAI was not changed.

The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs,. For any RAIs that cannot be answered **within 30 days or August 12, 2013**, it is expected that a date for receipt of this information will be provided to the staff within the 30-day period so that the staff can assess how this information will impact the published schedule.

Thank You,

Amy

Amy Snyder, U.S. EPR Design Certification Lead Project Manager
Licensing Branch 1 (LB1)
Division of New Reactor Licensing
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Mail Envelope Properties (88F9B30A3139B1498DA89BEBA7B31B900B70E771)

Subject: Response to U.S. EPR Design Certification Application FINAL RAI 591, Chapter 19, Supplement 1
Sent Date: 1/29/2014 2:59:33 PM
Received Date: 1/29/2014 3:00:06 PM
From: RYAN Tom (AREVA)

Created By: Tom.Ryan@areva.com

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Options
Priority: Standard
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Reply Requested: No
Sensitivity: Normal

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Response to

Request for Additional Information No. 591, Supplement 1

7/11/2013
U.S. EPR Standard Design Certification
AREVA Inc.
Docket No. 52-020
SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: 19.2
SPRA Branch

Question 19-370:

NEW PHASE 4 RAI

The 2013 PRA update model reflects integrated effects of individual changes that occurred since the 2009 PRA model. Please submit for review the U.S. EPR FSAR Section 19.2, "Severe Accident Evaluations," reflecting all of the changes in Section 19.2, including those from the updated PRA model.

Response to Question 19-370:

The response to RAI 591, Question 19-370 has not changed from the original response to RAI 591, which was transmitted on July 15, 2013. For this supplement, U.S. EPR FSAR markups are added to address NRC comments from the PRA Level 2 audit conducted on July 15-19, 2013.

The original response to RAI 591, with the added text from this Supplement 1, in addition to the changed pages in the U.S. EPR FSAR markups in this supplement, provide the full and complete Response to Question 19-370.

U.S. EPR FSAR Tier 2 Section 19.2 will be revised to address the following comments from the Level 2 PRA audit:

- Item 17 – "Revise AICC pressure and temperature values from Chapter 19.2 for equipment survivability for clarity so the licensee will know which values they need to use for equipment survivability (Pages 19.2-51 and 6.2.5-303)"
- Item 18 – "Determine if MOVs in the SAHRS (30JMQ42AA004 and 30JMQ42AA006) need to be included in the Table 19.2-3"
- Item 20 – "Page 19.2-52 Need clarification on whether it is still crediting the leak tightness of containment penetrations using the EPRI report"
- Item 21– "Page 19.2-53 Update wording of mixing damper position sensors to state they have no mission time"
- Item 22– "Table 19.2-3 Sheet 5 of 5 Note 1, and Title of Table 19.2-4 – Change SAMGs to OSSA."

FSAR Impact:

U.S. EPR FSAR Tier 2, Sections 19.2 and 6.2.5 will be revised as described in the response and indicated on the enclosed markup.

U.S. EPR Final Safety Analysis Report Markups

has no effect on the PARs and rupture and convection foils, and will cause the fail-safe mixing dampers to fail-safe open.

The CGCS is designed to operate in DBAs with elevated temperature, pressure and radiation. The mixing dampers, rupture and convection foils open early in the accident progression on pressure differential or temperature differential. If the DBA transitions into a Severe Accident, these components have to maintain integrity only. Their operation is not affected by localized pressure and temperature increase due to hydrogen combustion.

The PARs are designed for DBA, as well as SA conditions. For the SA analysis, 100% PAR efficiency is assumed.

The PARs are not pressure retaining components and are open at the bottom and the top, therefore, are unaffected by localized pressure increase. The design covers hydrogen combustion temperature peaks and SA radiation.

The CGCS operates effectively in a steam-saturated atmosphere (steam concentration greater than 55 percent by volume. CGCS equipment required to function during severe accident environmental conditions will demonstrate equipment survivability, including loads resulting from a hydrogen burn as discussed in Section 6.2.5.3.2. Several methodologies to demonstrate equipment survivability are presented in Section 19.2.4.4.5. Based on these two Sections, (6.2.5.3.2 and 19.2.4.4.5.2), the maximum AICC pressure and temperature are 100.8 psia and 1366°F (from 19.2.4.4.5.2). The AICC pressure is a theoretical value that cannot be reached because actual combustion is not adiabatic, isochoric, or complete. The best estimate, limiting pressure, and temperature resulting from a forced hydrogen combustion for CGCS equipment survivability are 76 psia and 796°F. Equipment survivability analyses consider hydrogen concentrations equivalent to that generated from a fuel clad-coolant reaction involving 100 percent of the fuel cladding surrounding the active fuel region using the methodology provided in Section 6.2.5.3.2. The low range and high range HMS systems are capable of operating during design basis accidents and severe accidents, respectively.

The low range hydrogen sensors are located inside the containment and meet the single failure criterion. These sensors are located in physically separated areas of the containment. Additionally, the signal processing is carried out by separate channel cards installed within the signal processing units that are located outside containment. The sensors and cables located inside containment are designed to remain operable during DBAs. The failure of one sensor or cable does not influence the reliability or accuracy of the other sensors.

The high range monitor for the HMS utilizes measuring modules and associated equipment of each independent train. The trains meet the single failure criterion by

Severe Accident Management Guidelines, and thus, their continued operation is no longer required.

Because of this unique design feature, the dampers can be powered from either the standard or essential buses. Power from the standard bus would be lost during a loss of offsite power, allowing the dampers to open. For power from the essential bus, sufficient time exists between loss of power and loading from the emergency diesel generators that would allow the dampers to open. In summary, during a transient, the safety function for these dampers occurs on a loss of power.

Pressure, Temperature, and Humidity within Containment

The pressure, temperature, and humidity time developments during a severe accident were evaluated from the MAAP4-based uncertainty analysis. The maximum "global" containment pressure and temperature that equipment and instrumentation may be exposed to during the progression of a severe accident are ~~76.9~~59.4 psia and approximately ~~410~~381°F, respectively. The maximum humidity inside the containment experienced by the equipment can conservatively can be assumed to be 100 percent after the commencement of spraying. However, due to the existence of other gases inside containment, the steam concentration approaches a conservative value of ~~80~~70 percent. Finally, the IRWST reaches a maximum temperature of ~~257~~230°F. The SAHRS is conservatively designed for a maximum IRWST water temperature of 320°F.

For the relevant scenarios, which form the basis of the uncertainty analysis cases, local~~ized~~ and global hydrogen detonations ~~and deflagration~~ can be reliably excluded. The highest AICC pressure and temperature resulting from the uncertainty analysis cases~~representative and bounding scenarios~~ is ~~105~~100.8 psia and ~~1634~~1366°F, respectively. The AICC pressure is a theoretical value that cannot be reached because actual combustion is not adiabatic, isochoric, or complete. The best estimate, limiting pressure, and temperature resulting from forced hydrogen combustion for equipment survivability are 76 psia and 796°F (Section 6.2.5.3.2). These are based on a forced hydrogen combustion for the uncertainty analysis case which yielded the highest AICC pressure. It is important to assess the equipment and instrumentation capabilities within these extended operational ranges. ~~Because best estimate, limiting pressure, and temperature resulting from hydrogen combustion during representative and bounding scenarios is 90 psia and 550°F, respectively, it is necessary to assess the equipment and instrumentation capabilities within this extended operational range.~~

While the equipment and instrumentation inside containment may be exposed to such pressure and temperature spikes, only equipment relied upon to establish and maintain safe shutdown and containment structural integrity must be capable of performing their functions during and after the exposure to the environmental conditions created by the burning of hydrogen per 10 CFR 50.44(c)(3).

For the mitigation of severe accidents, the Severe Accident Management Guidelines rely on safety-related and non-safety-related systems, structures, and components (SSC). Although electrical and mechanical safety-related equipment must perform their safety-related function during design bases events, the equipment necessary for mitigating the severe accident consequences is required to provide a reasonable level of confidence that it will function in a severe accident environment for the necessary time span. This requirement is referred to as "equipment survivability" and is different from "equipment qualification," which is terminology used for the level of assurance provided for equipment necessary for design basis accidents. This implies that the equipment survivability assessment may be performed with a practical engineering approach accompanying best-estimate accident environments.

Different methodologies such as analysis, calculations or data and test results obtained from type testing, industry experience, or test facilities can justify the operability and survivability of selected equipment and instrumentation. One example of performance expectations of similar equipment in severe accident environments is EPRI NP-4354 (Reference 21). This information is supplemented by NUREG/CR-5334 (Reference 22). The chamber conditions during these tests envelop those calculated for the U.S. EPR severe accident scenarios. Results of the Industry Degraded Core Rulemaking program (IDCOR) on equipment survivability provide further reassurances that selected equipment has the ability to survive and to perform its function.

Equipment specifications will define the applicable mission time and expected containment conditions during a severe accident for the equipment based on equipment location. Hydrogen burns will be considered in the equipment specifications as appropriate.

The containment integrity is maintained by the containment isolation valves, equipment hatch and airlocks, containment penetrations, and the containment structure itself. A structural analysis of the reactor containment dead-weight loads, prestressing loads, and the internal pressure load due to hydrogen burning was conducted and showed that the liner is within allowable levels of strain. Containment isolation valves are closed early during the accident progression prior to RPV failure when the threat of hydrogen burning is greatest. Therefore, their function of closing is unaffected by the conditions created from burning. As discussed, different methodologies such as analysis, calculations or data and test results obtained from type testing, industry experience, or test facilities can justify the performance of leak-tightness by the containment penetrations during combustion of hydrogen. One example of performance expectations of similar equipment in severe accident environments is EPRI NP-4354 (Reference 21). This information is supplemented by NUREG/CR-5334. ~~The performance of leak-tightness by the containment penetrations is not affected by the combustion of hydrogen. Containment isolation valves and penetrations were tested during the EPRI NP-4354 (Reference 21) and~~

~~NUREG/CR-5334 (Reference 22) tests. Neither test resulted in mechanical degradation that would impact the capability of the components to maintain leak-tightness.~~ Equipment hatch and airlocks are designed to be pressure resistant and airtight (see Section 3.8.2). The seals are embedded in grooves on the flanges and are protected from temperature spikes. The additional pressure load resulting from a spike will compress the seals and enhance contact between mating surfaces. The large metal structure will act as a heat sink over the long run. The components that provide containment isolation are classified as safety grade and are part of the U.S. EPR equipment qualification program (see Sections 3.10, 3.11, and Appendix 3D).

The main components identified in Table 19.2-2 and Table 19.2-3 are the containment spray nozzles, passive outflow restrictors, and the primary depressurization system (PDS) valves. The operability of these components is not impacted by hydrogen burning during a severe accident. The containment spray nozzles are tangential-flow hollow cone types and are unaffected by external pressure. The temperature spike predicted because of hydrogen burning will not affect the performance of these nozzles. The passive outflow restrictors are located in the valve compartments adjacent to the core spreading room in the bottom of the containment. The restrictors are steel pipe components and require no moving parts to perform their function. Pressure and temperature increases attributed to hydrogen burning will not impact their performance. The steel acts as a heat sink to the elevated temperature, and the pressure will not affect the structural integrity of the component. The PDS valves will perform their function of depressurizing the RCS early in the accident sequence. Their function will not be impacted by hydrogen burning because the release of hydrogen from the degraded core is not a factor until after the depressurization. Following the depressurization, the actuator will not be required to function and the valves can fail as is.

Instrumentation identified in Table 19.2-3 is mostly located in either the Safeguard Buildings or the Fuel Building, and is not exposed to the elevated temperature and pressure attributed to hydrogen burning. Instrumentation having a mission time early in the severe accident sequence (e.g., core outlet temperature, RCS pressure, PDS position indication, thermocouples for the monitoring of corium relocation in-vessel and ex-vessel, hydrogen mixing damper position indication, and delta pressure sensors) will not be affected by the resulting temperature and pressure spikes from hydrogen burning. Their function is performed before RPV failure and is acceptable to fail once their mission is completed. There is reasonable assurance that instrumentation that is required to function during a severe accident will survive the environments created by hydrogen burning. This instrumentation will be provided with additional protection and shielding, be protected from the environment by their installed location, or a combination of the two.

~~In the case of the hydrogen mixing damper position indicators, a pseudo mission time is associated with them~~ As previously discussed for the hydrogen mixing damper position indicators the mission time is effectively zero in that, for an SBO, the position indicators are not operable, meaning that power is also lost to the dampers, forcing them to open. Therefore, by power being removed, the position indicators provide the function that a pseudo open signal has been initiated. For non-SBO events, the hydrogen mixing damper position indicators will function at the entrance of the severe accident management function, at which time the operators will open the dampers, rendering the need of the position indicators unnecessary and emphasizing an effectively zero mission time.

The uncertainties in the equipment survivability are primarily attributed to uncertainties in the severe accident environments used for evaluation which may be beyond those conditions used in prior survivability tests and experiments. Some uncertainties may exist due to a lack of validation of analytical results by experimental data. The uncertainties do not alter the conclusions reached in the evaluation. Most of the equipment evaluated has inherent capabilities to function under conditions that exceed the severe environment requirements. The inherent capability of equipment to withstand severe environmental conditions, margins used in qualification tests, and prior hydrogen burn survivability experimental results provide assurance that equipment will survive degraded core accidents.

Radiation within Containment

A deterministic analysis of the direct dose radiation environment in the U.S. EPR buildings, as well as the submersion dose for the Reactor Building accident conditions, was performed for the U.S. EPR. For DBA conditions, the analysis is based on a large break loss of coolant accident (LBLOCA) which provided the bounding dose rates within the primary containment. A similar analysis for the post severe accident condition area dose rates within the Safeguard Building housing the SAHRS equipment was performed. The calculation addressed a LOCA with an end of cycle (EOC) equilibrium source term, and fractional releases from the core under severe accident conditions. Consideration was given to gamma doses inside the primary containment, in the annulus, above the IRWST, adjacent SAHRS pipes, and above a post- LOCA spill from SAHRS pipe leakage. In addition, beta doses were calculated for the primary containment and annulus.

For the doses from airborne radioactivity within the primary containment, plateout effects on the airborne halogens and particulates were ignored because they are strongly dependent on the post-LOCA conditions within the containment. The radiation fields produced by radioactivity plating out on internal surfaces was assumed to lead to the same overall doses when plateout is ignored. Note, however, that plateout effects within the primary containment were accounted for in definition of the airborne source within the annulus. This is because the annulus source term is a

19.2.5.2 OSSA Directed Actions

The ultimate goal for the OSSA is to provide mitigation strategies to cover all potential events that lead to core melt and to stop or reduce the releases of fission products to the environment.

Considering containment challenges rather than accident scenarios promotes protection of the containment as priority in every case regardless of the accident sequence. The OSSA considers a broad range of sequences, even if not analyzed or quantified through the PRA Level 2 or through the supporting safety studies. For the severe accident sequences occurring in the Fuel Building, building failure is not a concern due to the leakage rate and high degree of permeability of the structure. In this case, the building-defined challenges are the phenomena that can lead directly to large radioactive releases.

The OSSA diagnostic is developed based on a list of possible challenges to severe accident mitigation and the corresponding instrumentation used to assess safety margins. Subsequently, appropriate actions can be derived, predicated on which challenges may be present at a given moment. The technical bases for such actions can be developed in most instances through deterministic process studies using the U.S. EPR MAAP4.0.7 model. By using the framework defined in this section, the ~~SAMG~~OSSAs are developed which include specific U.S. EPR diagnostics, definition of necessary associated instrumentation, and the development of high level and in-depth mitigation strategies to be used by the technical support center during a postulated severe accident. Table 19.2-4—OSSA SAMG Technical Basis - Mapping Challenge Mechanism to Operator Action provides several examples mapping challenge mechanisms to operator action.

19.2.5.3 Interface with Emergency Procedures

The prevention of core damage is considered to be within the domain of EOPs. The review of existing approaches, together with the consideration of advanced reactor design measures, led to the choice of a 'standalone' severe accident management guidance. Once a specified set of plant conditions is met, the operators will implement guidance based on the OSSA mitigation strategies. For the U.S. EPR, it has also led to the decision to treat primary system depressurization as an immediate action. However, since primary system depressurization performs both preventive and mitigative functions, a consideration is needed which ensures that if the preventive function of depressurization is successful, the EOP guidance remains in effect. In addition to primary system depressurization, other immediate actions upon entering OSSA include manual initiation of containment isolation, manual initiation of low head safety injection (LHSI) for in-vessel recovery, and opening of the motor-operated valves in the SAHRS passive cooling lines. These immediate actions provide the starting point for the OSSA severe accident mitigation path.

Table 19.2-3—Severe Accident Instrumentation and Equipment
Sheet 1 of 6

Function	Measured quantity/ Activated device	Type of device	Location	12h Batt.	SBO
RCS Depress. & Monitoring					
	Core outlet temperature (COT)	Thermocouple	Incore fingers	x	x
	RCS pressure wide range	PDE	inlet RHR to RCS	x	x
	RCS pressure narrow range	PDE	inlet RHR to RCS	x	x
	PDS Actuation	Valve Actuator	PDS/PRZ	x	x
	PDS position	Position sensor	PDS/PRZ	x	x
Corium Position					
Threat of RPV melt_through	RPV outside-wall temperature	Thermocouple	RPV insulation	x	x
Corium arrival in core catcher	Temperature in chimney above spreading area	Thermocouple	"chimney" above spreading area	x	x
Corium arrival in core catcher	Position of flooding valve	Position sensor	flooding valve - SAHRS dedicated valve compartment	x	x
Corium arrival in core catcher	Passive flooding valves	Spring loaded valves	flooding valves - SAHRS dedicated valve compartment		
Entry into OSSA	Passive Flooding Isolation Valves	Motor Operated Valves	flooding valves - SAHRS dedicated valve compartment		
Corium arrival in core catcher	Flow downstream of Passive flooding valve	Flow rate meter	SAHRS dedicated valve compartments	x	x
Threat of basemat penetration	Temperature in basemat main cooling channel, or failure of thermocouple	Thermocouple	main core catcher cooling channel	x	x

Table 19.2-3—Severe Accident Instrumentation and Equipment
Sheet 6 of 6

Function	Measured quantity/ Activated device	Type of device	Location	12h Batt.	SBO
Activity Release					
	Position of MSRIV	Position sensor	LAB MSRIV compart.	x	x
	Dose rate in containment	Gamma sensitive detect.	inside containment and outside equipment hatch	x	
	Liquid and Containment Atmosphere samples	KUL Modules	Safeguard Building 4		x
	Dose rate downstream of filters	Gamma sensitive detect.	stack	x	
	Volume flowrate stack	Flow rate meter	stack		
	Dose rate in safeguard building	Gamma sensitive detect.	downstream of ventilation filter	x	
	Volume flowrate safeguard building ventilation	Flow rate meter	downstream of ventilation filter		
	Dose rate in the annulus	Gamma sensitive detect.	downstream of ventilation filter	x	
	Volume flowrate annulus ventilation	Flow rate meter	downstream of ventilation filter		
Annulus Ventilation Monitoring					
	Subatmospheric pressure	PDE	annulus		

Notes:

1. The mission time of the hydrogen dampers and the position sensors is effectively zero in that the hydrogen dampers either open at the initiation of the accident from loss of power (prior to the loading from the emergency diesel generators), open as a result of a pressure differential, or by the operators during the declaration phase of entering the OSSA. Therefore, their operation during the severe accident is not required; however, their open state must be maintained.

Table 19.2-4—OSSA ~~SAMG~~ Technical Basis - Mapping Challenge Mechanism to Operator Action

Challenge Mechanism	Applicable Safety Function	Plant Parameters / Instrumentation Needed for Diagnostic	Applicable Immediate or Systematic Actions
Interfacing LOCA Containment Isolation Failure	Releases	Site dose Auxiliary / safeguards buildings radiation Containment isolation valve position	Containment isolation (immediate action)
Hydrogen deflagration in phase 2 (SG available) Hydrogen deflagration in phase 2 (no SG available) Ex-Vessel Hydrogen Deflagration	Containment	Containment pressure Containment hydrogen concentration	N/A
Induced SGTR	Releases	Site dose Steam system radiation Primary system pressure	Primary depressurization (systematic action, in EOPs prior to entry)
Basemat Penetration	Heat removal	Containment heat removal system temperatures and flows Core catcher thermocouples	N/A
SA Following Initiated SGTR (Containment Bypass)	Releases	Site dose Steam system radiation	N/A
Quenching	Containment	Containment pressure	N/A
DCH RPV Rocket	Releases	Site dose Annulus radiation Primary system pressure	Primary depressurization (systematic action, in EOPs prior to entry)