ArevaEPRDCPEm Resource

From:	BRYAN Martin (EXTERNAL AREVA) [Martin.Bryan.ext@areva.com]
Sent:	Friday, August 13, 2010 11:53 AM
To:	Tesfaye, Getachew
Cc:	NOXON David (AREVA); STOUDT Roger (AREVA); ROMINE Judy (AREVA); RYAN Tom (AREVA); WILLIFORD Dennis (AREVA); SANDERS Harris (AREVA)
Subject:	DRAFT Response to U.S. EPR Design Certification Application RAI No. 382, FSAR Ch. 19 - PHASE 4 RAI, Supplement 1
Attachments:	RAI 382 Supplement 2 Response - US EPR DC (DRAFT).pdf; ANP-10314 OSSA Technical Report - US EPR DC (DRAFT).pdf

Getachew,

Attached is a draft response for RAI 382 in support of a final response date of September 6, 2010. Also attached is a draft of Operating Strategies for Severe Accidents Methodologies for the U.S. EPR Technical Report to support the review. Let me know if the staff has questions of if this response can be sent as final.

Thanks,

Martin (Marty) C. Bryan
U.S. EPR Design Certification Licensing Manager
AREVA NP Inc.
Tel: (434) 832-3016
702 561-3528 cell
Martin.Bryan.ext@areva.com

From: BRYAN Martin (EXT)
Sent: Tuesday, July 20, 2010 5:07 PM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen V (AREVA NP INC); ROMINE Judy (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); NOXON David B (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 382, FSAR Ch. 19 - PHASE 4 RAI, Supplement 1

Getachew,

AREVA NP Inc. (AREVA NP) provided a schedule for a technically correct and complete response to RAI No. 382 on June 1, 2010.

The schedule for RAI 382 Question 19-336 is being revised to allow more time for issuing the draft response and for NRC review of the draft response. The schedule for the technically correct and complete response to the remaining question has changed and is provided below:

Question #	Supplement Date (providing FSAR Markup)
RAI 382 — 19-336	September 6, 2010

Sincerely,

Martin (Marty) C. Bryan
U.S. EPR Design Certification Licensing Manager

AREVA NP Inc.
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From: BRYAN Martin (EXT)
Sent: Tuesday, June 01, 2010 7:58 AM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen V (AREVA NP INC); ROMINE Judy (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); NOXON David B (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 382, FSAR Ch. 19 - PHASE 4 RAI

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 382 Response US EPR DC.pdf," provides the schedule for a technically correct and complete response to the one question.

The following table indicates the respective pages in the response document, "RAI 382 Response US EPR DC.pdf," that contains AREVA NP's response to the subject question.

Question #	Start Page	End Page
RAI 382 — 19-336	2	4

The schedule for technically correct and complete responses to the one questions is provided below.

Question #	Response Date
RAI 382 — 19-336	July 21, 2010

Sincerely,

Martin (Marty) C. Bryan
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From: Tesfaye, Getachew [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Thursday, April 29, 2010 2:14 PM
To: ZZ-DL-A-USEPR-DL
Cc: Fuller, Edward; Phan, Hanh; Mrowca, Lynn; Chowdhury, Prosanta; Colaccino, Joseph; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 382 (4539), FSAR Ch. 19 - PHASE 4 RAI

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on March 22, 2010, and discussed with your staff on April 28, 2010. Drat RAI Questions 19-336 (a) was deleted and 19-336 (b) was modified as a result of that discussion. 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, 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.

Thanks,
Getachew Tesfaye
Sr. Project Manager
NRO/DNRL/NARP
(301) 415-3361

Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 1836

Mail Envelope Properties (BC417D9255991046A37DD56CF597DB710732F104)

Subject: DRAFT Response to U.S. EPR Design Certification Application RAI No. 382, FSAR Ch. 19 - PHASE 4 RAI, Supplement 1
Sent Date: 8/13/2010 11:52:41 AM
Received Date: 8/13/2010 11:53:01 AM
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MESSAGE	4109	8/13/2010 11:53:01 AM
RAI 382 Supplement 2 Response - US EPR DC (DRAFT).pdf		523107
ANP-10314 OSSA Technical Report - US EPR DC (DRAFT).pdf		3207839

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Response to

Request for Additional Information No. 382(4539), Supplement 2

3/29/2010

U.S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: Chapter 19

QUESTIONS for PRA Licensing, Operations Support and Maintenance Branch 2 (ESBWR/ABWR Projects) (SPLB)

Question 19-336:

Follow-up to RAI 133, Question 19-243 (OPEN ITEM)

The response to RAI 133, Question 19-243, includes as Appendix A the OSSA Methodology Technical Basis report. The staff has reviewed this document. In order to complete its review, however, the staff needs some additional information as follows:

a. [Intentionally deleted]

b. In Section 2.1, early containment failure is defined to be most consequential in terms of public dose. However, containment bypass states are of a greater potential risk to public dose than early containment failure. Please discuss any measures that are contemplated to manage steam generator tube rupture or other containment bypass states.

c. How are the other initiators (e.g., external flooding, fires, and seismic events) included in the current OSSA process?

d. Please discuss the reasons for not providing an additional indicator/measured parameter, besides core exit temperature, as a basis for entry into the ECHUR OSSA domain.

e. A correlation between primary system pressure, core outlet temperature, and maximum clad temperature, to determine entry into OSSA is mentioned in Section 4.3.1. Please provide details of the correlation, and define its limits of applicability.

f. In ECHUR, the main accident management action includes RCS depressurization by the opening of PDS valves. Please discuss the rationale for not using the steam generator depressurization system, especially if secondary heat sink is available. If both modes of depressurization were available, which one would be preferred and why?

g. The SAMG termination phase is stated to be based on following trends rather than monitoring a specific parameter. It is recognized that the instrumentation and their associated qualification and set point requirements are planned as part of the OSSA guidance development and implementation. Please discuss what sensors and/or measured parameters will be used to follow the trends/indicators of achieving a stable configuration, given that core exit temperature thermocouples are either not available or not useful.

h. The OSSA methodology addresses all plant operating states, including shutdown and refueling conditions. Section 2.4 outlines actions that would be considered for three categories of shutdown scenarios. Please clarify, for each scenario, the logic presented, particularly as pertinent to accumulator and LHSI injection.

i. Section 2.4.2 of the OSSA Methodology Technical Basis report states that a list of instrumentation required, and corresponding set points, will be documented during the OSSA development process. The staff needs to review this list to assure that the Technical Basis is truly established. Please either provide the list or propose a COL information item.

j. Please explain why heatup of hot legs, the surge line, and steam generator tubes would be addressed in the core melting phase (Section 3.1.3) and not in the core heatup phase (Section 3.1.2).

k. Please describe the accident management strategies that would be adopted to cope with possible relocation of core debris after vessel breach into steam generator compartments, pump rooms, and other containment compartments. What are the major issues associated with instrumentation and other equipment in these compartments, given the presence of relocated core debris?

l. Please discuss the provisions that exist to enable the operators to diagnose the potential for reduced effectiveness of Passive Autocatalytic Recombiners (PARs) due to coking, fission product aerosol poisoning, and/or removal of PARs for repairs (under shutdown/refueling modes). In Table 3-1, venting is listed as a potential mitigation strategy, and is again discussed in Section 3.4.6. Please elaborate on the strategies for using other hydrogen control measures under degraded PAR conditions to circumvent potential challenges due to hydrogen combustion. Furthermore, Table 3-1 also lists "Shut down heat sinks." Please explain what structures are being referred to, and show how they can be effective.

m. Please discuss the AM implications of any degradation in the behavior of the engineering systems (PDS, CGCS, CMSS, SAHRS) designed to mitigate the consequences of severe accidents in the U.S. EPR. For each system, explain how serious system degradation could influence planned OSSA strategies, including use of available instruments and other procedural alternatives.

n. Timely operation of the depressurization valves is part of the accident management strategy and is very important to avert possible induced creep ruptures of hot legs or damaged SG tubes. The response to RAI 133, Question 19-240 showed the amount of time available between when the core exit temperature reaches 1,200°F and when induced SGTR might be expected for varying degrees of tube damage. The results showed that 18 to 20 minutes would be available (assuming a hot leg would not fail first). These results establish the importance of prompt depressurization and the need for a good Human Reliability Assessment (HRA) assessment of the probability of failing to depressurize in time. Please describe how possible delays in primary system depressurization will be addressed in OSSA, and how HRA methods will be utilized in this regard.

o. Table 3-1 does not list ex-vessel steam explosions as a potential challenge. Please explain why this is not considered a challenge. If it is a significant challenge, what actions, if any, would be considered to mitigate the consequences.

p. Please discuss any downside associated with potential accident management strategies (e.g., shattering of a hot core due to flooding, enhanced oxidation beyond the capacity of PARs resulting in build-up of detonable mixtures in some containment regions, etc.), and how these may influence the implementation of SAMGs.

q. Regarding the information that the operators need to know (Section 3.4.3), please describe the reasons why the potential "downsides" of particular actions are not listed.

r. Please explain why the guidance to the TSC director does not provide, at every decision step, an explicit assessment of both the pluses and minuses of the various outcomes related to the situation as it is perceived to actually exist at the time to help the decision-making process.

Response to Question 19-336:

A description of the AREVA Operational Strategy for Severe Accidents (OSSA) Methodology and its application to the U.S. EPR was originally provided in the Response to RAI 133, Supplement 5, Question 19-243 as Appendix A. ANP-10314, Revision 0, “The Operating Strategies for Severe Accidents Methodology for the U.S. EPR Technical Report,” builds on the information provided in that response, restructuring and adding information to specifically address the evaluation of the U.S. EPR against the SECY-88-147 guidance on severe accident management. U.S. EPR FSAR Tier 2 Section 19.2.5 will be revised to add a reference to ANP-10314.

Part b:

ANP-10314 includes information to address this question. Specifically, Section 3.10, “Management of Radiological Releases”, acknowledges the potential fission product release pathway created by a steam generator tube rupture (SGTR) and the role of flooding as a mitigation response. Both the “induced SGTR” and “severe accident following SGTR initiating event” were identified in Table A-4, “Challenges and Potential Mitigation Strategies” as credible severe accident challenges for the U.S. EPR. Several mitigation strategies and actions are associated with these challenges. Of particular note is the statement “Fill steam generators, use emergency feedwater system EFW or MFW to maintain steam generator level high, continue RCS depressurization.” Positive and negative effects of injecting into the steam generators are addressed in Section 5.5 “High System-Level Action” .

Part c:

ANP-10314 includes information to address this question. Section 2.0, “Decision-Making Process”, and Section 3.1, “Management of Guideline Development”, acknowledge the value of using symptom-based diagnostics and guidelines. This approach removes potential bias and uncertainty inherent in event-based approaches, including those from an external cause such as floods, fires and earthquakes.

Part d:

ANP-10314 includes information to address this question. As addressed in Section 4.4.8, instrumentation used for severe accident monitoring will be evaluated for their potential alternative roles in tracking severe accident challenges. Section 4.1.1 specifically addresses the OSSA entrance and exit criteria assessment and identifies containment radiation as a redundant measure.

Part e:

ANP-10314 includes information to address this question. Section 4.4.6, Entrance/Exit Criteria Assessment, describes an OSSA task that includes the development of an OSSA entrance criteria correlation. OSSA tasks support the COL applicants. OSSA customer end products are provided to the COL applicants, including the entrance criteria assessment, to aid them in their development of plant-specific SAMG.

Part f:

ANP-10314 includes information to address this question. Accident mitigation involving steam generator depressurization will be considered in certain situations as determined from the OSSA support studies task (note study 3.1 in Attachment B of the report). Section 5.4, “Depressurize Steam Generators”, presents some detail on depressurizing the steam generators in the context of a “High System-Level Action”, including both positive and negative aspects.

Part g:

ANP-10314 includes information to address this question. Ultimately, accident progression will be tracked by following the trends of the radiological releases, containment integrity, and core heat removal safety functions. Section 3.3 identifies a broad list of instruments that will be use to follow the trends, most of which are required for the emergency response data system (ERDS) (see Section 1.2.3). As stated in the response to Part d of this question, containment radiation is expected to be used as the backup signal indicating a severe accident. Sections 3.4 and 4.4.8 addresses the role of instrumentation survivability on OSSA development and the OSSA task for assessing survivability setpoints.

Part h:

ANP-10314 includes information to address this question. The operating modes are addressed in Section 4.2. As a symptom-based methodology, detailed specification of scenarios is not an objective. Rather, the development of OSSA end-products develop from the OSSA tasks described in Section 4.4 that separately considered the at-power, shutdown, and refueling plant states as well as the plant conditions that lead to challenges to the fission product barriers.

Part i:

ANP-10314 includes information to address this question. As described in Section 1.2.3, 10 CFR 50 Appendix E defines the regulatory requirements regarding emergency planning and specifies requirements for an ERDS. The U.S. EPR will incorporate an ERDS that complies with 10 CFR 50 Appendix E. Section 3.3 includes a list of instrumentation for monitoring and responding to severe accidents. Setpoints defining challenge states is a task of OSSA and is described in Section 4.4.8.

Part j:

ANP-10314 includes information to address this question. Attachment C provides the severe accident progression summary. During the early phase of an accident that leads to core damage (which can usually be characterized by a design-basis accident), heatup of the hot legs, surgeline, and steam generator tubes is not significant because they are maintained at the coolant saturated temperature. At the point of core degradation, significant steam superheat is possible. Considering that OSSA is developed to address recovery actions during a severe accident, the description provided associates hot leg, surgeline, and steam generator tubes challenges simultaneously with OSSA entry.

Part k:

ANP-10314 includes information to address this question. Section 5.0 presents several high system-level actions that address a variety of severe accident conditions. Among these is reactor coolant system (RCS) depressurization using the U.S. EPR primary depressurization system (PDS). PDS actuation satisfies the SECY-93-087 regulatory expectation for such a system to effectively eliminate the possibility of a high pressure melt ejection. In addition, the pathways between the reactor pit and the equipment rooms are torturous, presenting many opportunities for the shadowing of these rooms from an ejected melt. As such, relocated debris in these compartments is not viewed as a credible situation. Unlike instrumentation applied in design-basis analysis, instrumentation survivability is demonstrated for the more likely scenarios (see the Response to RAI 6, Question 19-78), not the most adverse condition imaginable. Nonetheless, the OSSA evaluation of instrumentation (Section 4.4.8) includes a study of alternative roles as challenge state indicators that could accommodate this low-frequency, high consequence challenge.

Part l:

ANP-10314 includes information to address this question. Section 3.4, “Management of Equipment Survivability and Recovery,” provides information on how system degradation is considered in the development of OSSA end-products. GDC 42 requires that the containment atmosphere cleanup system (which in particular includes PARs in the U.S. EPR containment) is designed to permit appropriate periodic inspection. Similarly, GDC 43 states that these systems must be designed to permit appropriate periodic pressure and functional testing. The design bases and testing programs associated with the PARs are described in U.S. EPR FSAR Tier 2, Section 6.2.5. The regular testing and maintenance required by GDC 42 and GDC 43 confirms availability should a severe accident occur. In addition, as was done for severe accident evaluations addressing Section 19.2 content in the U.S. EPR Final Safety Analysis Report, uncertainty in PAR performance will be explicitly addressed in OSSA support studies. The term “heat sinks” within Table 3-1 is not referring to physical structures but to any mechanism that would otherwise remove steam from the containment atmosphere, e.g., containment spray.

Part m:

ANP-10314 includes information to address this question. As stated in the response to Part i, Section 3.4, “Management of Equipment Survivability and Recovery,” provides information on how system degradation is considered in the development of OSSA end products.

Part n:

ANP-10314 includes information to address this question. The MAAP4 analysis prepared for RAI 133, Supplement 2, Question 19-240 was performed specifically to study induced SGTR. To do so, the scenario assumed that operators, rather than abiding to emergency procedures and severe accident management guidance, chose to depressurize the steam generators. As addressed in Section 5.2, depressuring the RCS/RPV by actuation of the PDS will be performed before other severe accident recovery actions. Section 5.4, “Depressurize Steam Generators,” specifically identifies the possible negative consequences of performing this action, including the potential to increase the transport of fission products to the environment. The analysis described in the response to RAI 133, Supplement 2, Question 19-240 demonstrated that actions to depressurize the steam generators must occur only after primary system depressurization to eliminate the creep rupture potential in the steam generator tubes.

Regarding Human Reliability Assessment (HRA), the uncertainty analyses supporting OSSA (described in Sections 4.4.4 and 4.4.10) will consider conclusions drawn from HRA by incorporating reaction time uncertainty among the treated uncertainty parameters.

Part o:

ANP-10314 includes information to address this question. As addressed in the response to RAI 349, Supplement 5, Question 19-334, an ex-vessel steam explosion analysis was performed and demonstrated that the containment structure would likely withstand the impact. Nonetheless, Table A-4 has been updated to include "Large containment failure due to ex-vessel steam explosion," As such, ex-vessel steam explosion will be considered among the severe accident challenges explicitly address in the OSSA methodology.

Part p:

ANP-10314 includes information to address this question. The assessment of downside or negative impacts from accident management actions is an integral part of OSSA, reflecting the emphasis appearing in NEI-91-04 and EPRI 101869 (see Section 1.2.4). The detailed assessment of both positive and negative aspects of candidate accident management strategies in the OSSA methodology is acknowledged in Section 4.4.9. Section 5.0 reviews several candidate high system-level actions, providing a qualitative review of the merits and disadvantages of these actions.

Part q:

ANP-10314 includes information to address this question. As addressed in Section 2.2.2, "Guidelines," and 3.2, "Management of Response Implementation and Personnel Training," the decision-makers will be expected to evaluate both positive and negative aspects of candidate actions and communicate this to the control room operators.

Part r:

ANP-10314 includes information to address this question. See the responses to Part p and Part q of this question, and Table 3-1, "Emergency Response Team Responsibilities" of ANP-10314.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 19.2.5 will be revised as described in the response and indicated on the enclosed markup.

U.S. EPR Final Safety Analysis Report Markups

the lessons learned to date in the field of severe accidents and incorporates a number of new features which simplify and streamline the guidance material while maintaining comprehensive guidance for response to any severe accident. The OSSA framework is described in ANP-10314, "The Operating Strategies for Severe Accidents Methodology for the U.S. EPR Technical Report" (Reference 23).

19.336

The purpose of this section is to describe the OSSA framework for the U.S. EPR SAMGs. The high-level actions that would need to be taken to mitigate severe accidents are described in the context of the unique severe accident design features of the U.S. EPR. The potential challenges that need to be addressed by the technical support center team and the OSSA diagnostic tool used to mitigate these challenges are described.

A COL applicant that references the U.S. EPR design certification will develop and implement severe accident management guidelines prior to fuel loading using the Operating Strategies for Severe Accidents (OSSA) methodology described in this section and in Reference 23. ~~Section 19.2.5.~~

As stated in Section 19.1.2.2, the COL applicant will review final plant-specific EOPs and SAMGs to confirm that the assumptions used in the PRA and severe accident analyses remain valid.

19.2.5.1 Accident Management through Design

Severe accident management in the U.S. EPR begins with several design elements specifically addressing the stated objectives of maintaining fuel, RPV, and containment integrity while minimizing radiological releases. These design elements have been described in Section 19.2.2 and Section 19.2.3.

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.

19.2.7.5.4 SFP Heat Removal Capability

With the SFP integrity maintained, SFP cooling is provided consistent with the PRA. The availability of the make-up systems is assured due to the integrity of the Fuel Building exterior walls. The fire protection system provides the capability to fill the Spent Fuel Pool.

19.2.7.6 Conclusions

The U.S. EPR has inherent protection to avoid or mitigate, to the extent practical and with reduced reliance on operator actions, the effects of an aircraft impact. The assessment confirmed that the U.S. EPR design meets the four acceptance criteria. The reactor remains cooled, AND the containment remains intact; AND spent fuel cooling is maintained, AND spent fuel pool integrity is maintained. Accordingly, the U.S. EPR design features and functional capabilities provide for adequate protection of public health and safety in the event of an impact of a large commercial aircraft as required by 10 CFR 10.150. In fact, by exceeding the minimum acceptance criteria, the U.S. EPR design maintains significant margin beyond the minimum requirements specified in 10 CFR 50.150.

19.2.8 References

1. ANP-10268P-A, Revision 0, "U.S. EPR Severe Accident Evaluation Topical Report," AREVA NP Inc, February, 2008.

2. Fauske and Associates, Inc., 1994a. MAAP4—Modular Accident Analysis Program for LWR Power Plants, vol. 2, Part 1: Code Structure and Theory, prepared for Electric Power Research Institute, May 1994.

3. SECY-90-016, "Evolutionary Light Water (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," U.S. Nuclear Regulatory Commission, issued January 12, 1990, and the corresponding SRM, issued June 26, 1990.

4. SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water (ALWR) Designs," U.S. Nuclear Regulatory Commission, issued April 2, 1993, and the corresponding SRM, issued July 21, 1993.

5. WASH-1400 (NUREG-75/014), "Reactor Safety Study-An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S, Nuclear Regulatory Commission, October 1975.

6. NUREG-1116, "A Review of Current Understanding of the Potential for Containment Failure Arising from In-Vessel Steam Explosions," Steam Explosion Review Group (SERG), U.S. Nuclear Regulatory Commission, February 1985.

7. Theofanous, T. G., et al, DOE/ID-10489, "The Study of Steam Explosions in Nuclear Systems," Department of Energy, June 1996.

8. Deleted.

9. Wilks, S.S., "Determination of Sample Sizes for Setting Tolerance Limits," Ann. Math. Stat., Vol. 12, pp. 91-96, 1941.

10. EPRI TR-103413, "The MELTSPREAD-1 Computer Code for the Analysis of Transient Spreading and Cooling of High-Temperature Melts – Code Manual," Electric Power Research Institute, December 1993.

11. Breitung, W., et al, "Flame Acceleration and Deflagration-to-Detonation Transition in Nuclear Safety," NEA/CSNI-(2000) 7, October 2000.

12. Tutu, N.K., T. Ginsberg, and L. Fintrok (1988). "Low Pressure Cutoff for Melt Dispersal from Reactor Cavities," Fourth Proceedings of Nuclear Thermal Hydraulics, 29-37.

13. Pilch, M. M., et al, "Resolution of the Direct Containment Heating Issue for All Westinghouse Plants With Large Dry Containments or Subatmospheric Containments," NUREG/CR-6338, SAND95-2381, 1996.

14. Meyer, L., et al, "Melt Dispersion and Direct Containment Heating (DCH) Experiments in the DISCO-H Test Facility," FZK Report FZKA 6988, ISSN 0947-8620, May 2004.

15. ANP-10290, Revision 1, "Environmental Report Standard Design Certification," AREVA NP Inc, September 2009.

16. NEI 05-01 (Rev A), "Severe Accident Mitigation Alternatives (SAMA) Analysis, Guidance Document," Nuclear Energy Institute, November 2005.

17. NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," U.S, Nuclear Regulatory Commission, January 1997.

18. NEI 07-13, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs," Revision 7, May 2009.

19. Letter from D. Matthews, NRC to R. Ford, AREVA NP, "Approval of AREVA NP Inc. Safeguards Protection Program and Reviewing Official, and Transmittal of Beyond Design Basis, Large Commercial Aircraft Characteristics Specified by the Commission", December 21, 2007.

20. ANP-10296, Revision 0, "U.S. EPR Design Features that Enhance Security," AREVA NP Inc., December 2008.

21. Achenbach, J.A., Miller, R.B., Srinivas, V., "Large-Scale Hydrogen Burn Equipment Experiments," EPRI NP-4354, Electric Power Research Institute, 1985.

22. NUREG/CR-5334, "Severe Accident Testing of Electrical Penetration Assemblies," SAND89-0327, November 1989.

23. ANP-10314, Revision 0, "The Operating Strategies for Severe Accidents Methodoloy for the U.S. EPR Technical Report," AREVA NP Inc., July 2010.

19.336

ANP-10314
Revision 0

The Operating Strategies for Severe Accidents Methodology for the U.S. EPR™ Technical Report

August 2010

AREVA NP Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
		This is a new document

Contents

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List of Figures

Nomenclature

Acronym/Initialism	**Definition**
ALWR	Advanced Light Water Reactor
AVS	Annulus Ventilation System
CA	Computational Aid
CCWS	Component Cooling Water System
CDES	Core Damage End State
CET	Containment Event Tree
CFR	Code of Federal Regulations
CGCS	Combustible Gas Control System
CMSS	Core Melt Stabilization System
COL	Construction and Operation License
COT	Core Outlet Temperature
CVCS	Chemical Volume Control System
DBA	Design Basis Accident
DCR	Damaged Core Response
DCH	Direct Containment Heating
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
EBS	Extra Borating System
ECCS	Emergency Core Cooling System
ECHUR	Extended Core Heat Up Response
EDG	Emergency Diesel Generator
EFW	Emergency Feedwater
EFWS	Emergency Feedwater System
EOP	Emergency Operator Procedures
EPRI	Electric Power Research Institute
ERDS	Emergency Response Data System
FCI	Fuel-Coolant Interactions
FCM	Fuel Centerline Melt
FSAR	Final Safety Analysis Report
FWLB	Feedwater Line Break
HEPA	High Efficiency Particulate Air
HPME	High Pressure Melt Ejection

Acronym/Initialism	Definition
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IPE	Individual Plant Examination
IRWST	In-containment Refueling Water Storage Tank
LPD	Linear Power Density
LHSI	Low Head Safety Injection
LOCA	Loss Of Coolant Accident
LWR	Light Water Reactor
MAAP	Modular Accident Analysis Program
MCCI	Molten-Core Concrete Interactions
MFW	Main Feedwater
MHSI	Medium Head Safety Injection
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSRCV	Main Steam Relief Control Valve
MSRIV	Main Steam Relief Isolation Valve
MSRT	Main Steam Relief Train
NEI	Nuclear Energy Institute
OSSA	Operating Strategies for Severe Accidents
PRA	Probabilistic Risk Assessment
PAR	Passive Autocatalytic Recombiner
PDS	Primary Depressurization System
PICS	Process Information and Control System
PRT	Pressurizer Relief Tank
PWR	Pressurized Water Reactor
PZR	Pressurizer
RCCA	Rod Control Cluster Assemblies
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RTD	Resistance Temperature Detector
SACRG	Severe Accident Control Room Guide

Acronym/Initialism	Definition
SAEG	Severe Accident Exit Guide
SAG	Severe Accident Guide
SAHRS	Severe Accident Heat Removal System
SAMG	Severe Accident Management Guidelines
SCG	Severe Challenge Guide
SCST	Severe Challenge Status Tree
SI	Safety Injection
SICS	Safety Information and Control System
SIS	Safety Injection System
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SSS	Startup-Shutdown System
TMI	Three Mile Island
TSC	Technical Support Center
UPS	Uninterruptible Power Supplies
V&V	Verification and Validation

Executive Summary

In accordance with NRC policy statements on severe accidents and advanced and evolutionary reactors, new reactor designs should demonstrate improved severe accident characteristics compared with the current fleet of operating reactors. Accident management can improve plant performance during a severe accident by accelerating the recovery of the plant to a controlled, stable state. This technical report describes the operating strategies for severe accidents (OSSA) methodology for developing severe accident management guidelines (SAMG) for the U.S. EPR™. Implementation of OSSA is guided by specific safety goals and identified corresponding severe accident challenges.

Based on industry experience, the U.S. EPR™ includes design and operational features that specifically facilitate severe accident management. The objective of this technical report is to support the content on severe accident management guidance that appears in the U.S. EPR™ Final Safety Analysis Report. This report has been prepared specifically to address the evaluation of the U.S. EPR™ against the SECY-88-147 guidance on severe accident management. Section 1.0 provides a survey of regulatory and industry comment on functional requirements for a severe accident management program. The remainder of the report implements the methodology by:

- Describing the decision-making process (Section 2.0).
- Defining the OSSA mission requirements for the U.S. EPR™ (Section 3.0).
- Summarizing the OSSA design and development methodology (Section 4.0).
- Providing a compilation of high level actions for the U.S. EPR™ (Section 5.0).
- Discussing the documentation of user-end-products (Section 6.0).

1.0 INTRODUCTION

AREVA NP has developed an optimized approach to severe accident management in a project called operating strategies for severe accidents (OSSA). This optimized approach confirms that insights established through industry experience identify accident management mission requirements that direct activities related to the development of severe accident management guidelines (SAMG). Consistent with the U.S. EPR™ general design philosophy on severe accident response, OSSA-derived SAMG retain the goal of reducing or eliminating many of the uncertainties associated with severe accident progression. The dedicated severe accident response features are a key contributor towards this objective.

The OSSA approach for SAMG development follows a systematic methodology similar to that applied in the development of protection systems. This approach begins by defining mission requirements followed by a large set of support studies used to identify the plant condition and quantified thresholds for which actions are required to maintain the plant on a mitigation path.

While the major components of severe accident engineering are the credited test programs and corresponding analytical methods (see Reference [1]), the identification of the necessary analyses supporting SAMG development involves engineering insights that combine regulation, industry experience, fundamental understanding of thermal-hydraulic and severe accident phenomena, and risk/consequence factors. Safety goals are translated into analysis measures, uncertainties are characterized, and calculations are designed to demonstrate the completeness of the design based on the expected domain of possibilities.

A severe accident management program designed for a nuclear plant provides the plant staff with the capability of coping with the domain of credible severe accidents and securing maximum benefit from the margin of strength that enables containments to accommodate significantly greater loads than the design basis would suggest. The program requires that appropriate systems are available within the plant to enable plant staff the ability to diagnose the faults and implement appropriate response strategies.

The program must also provide the necessary guidance and training to confirm that appropriate corrective actions are implemented. While the principal goal of severe accident management is to prevent radiological releases, the fundamental objectives of accident management are:

1. To monitor the main characteristics of plant status.
2. To control core subcriticality.
3. To restore the heat removal from the core and maintain long-term core cooling.
4. To eliminate the possibility of high pressure melt ejection (HPME) through reliable primary system depressurization.
5. To protect the integrity of the containment by verifying heat and combustible gas removal.
6. To provide a long-term cooling solution in the event of a severe accident.
7. To regain control of the plant, if possible, and, if degradation cannot be stopped, delay further plant deterioration and implement on-site and off-site emergency response.

The objective of the OSSA methodology described herein is to describe the technical and analytical bases that satisfy the regulatory expectation for the development of strategies that lead plant personnel along an appropriate mitigation path following an event that results in the loss of core cooling and subsequent fuel rod damage. This technical guide builds upon the Defense in Depth in Nuclear Safety document (Reference [2]) and presents an expanded overview of the AREVA NP OSSA methodology.

1.1 *Role of SAMG Development within the Defense-in-Depth Framework*

Along with the physical barriers protecting the environment from the consequences of nuclear power plant radiological releases, accident management is a key component of an effective defense-in-depth strategy comprised of the following:

Level 1. A combination of conservative design, quality assurance, and surveillance activities to prevent departures from normal operation.

Level 2. Detection of deviations from normal operation, protection devices, and control systems to prevent escalation into accidents.

Level 3. Engineered safety features and protective systems that are provided to mitigate accidents and thus prevent their evolution into severe accidents.

Level 4. Measures to preserve the integrity of the containment and enable control of severe accidents.

Level 5. Mitigation of radiological consequences of significant external releases.

Emergency operating procedures (EOP) and SAMG are applied when the defense-in-depth layers Level 1 – 3 are challenged. The SAMG primarily serve to maintain the Level 4 defense-in-depth objective as the last barrier before radioactive material releases into the environment. In addition, on containment failure or bypass, SAMG continue to provide guidance for actions that limit or otherwise reduce radiological releases to the environment.

In light water reactors (LWR), several severe accident challenge areas are emphasized such as core coolability/melt stabilization, containment heat removal, and isolation of the source of radiological releases. The ultimate objective of severe accident management is to bring the plant back to a controlled, safe, and stable state that can be maintained long-term. This is defined as the quasi-steady-state situation which should exist after the early and intermediate efforts are completed (i.e., within a few days). The time-span from long-term control is anticipated to range from a few days to several months.

In order for a plant to function in conditions well beyond the design-basis, a margin of safety should be exploited to maintain control over events and minimize the consequences to the public. The most effective approach is to make accident initiation less likely (accident prevention), as well as to reduce the probability of it propagating at every subsequent stage. Accident management is highly important at all stages of accident development, from initiation to long-term control.

1.2 *Applicable Documents*

1.2.1 *Background Information for Severe Accident in the U.S. EPR™ Design*

As described in the AREVA NP Request for Review and Approval of ANP-10268P letter (Reference [1]), the goal of the severe accident mitigation concept of the U.S. EPR™ design is to verify the function of the containment even in the event of a severe accident. To meet this design goal, specific design features have been incorporated to retain and stabilize the molten core inside containment as well as to mitigate environmental effects that can compromise its fission product retention capability. The dedicated features addressing severe accident challenges incorporated in the U.S. EPR™ design include:

- Primary depressurization system (PDS) valves for rapid depressurization of the RCS.
- Multiple passive autocatalytic recombiners (PAR) to reduce in-containment hydrogen concentration; thus, minimizing the risk of hydrogen detonation.
- Engineered features (e.g., containment sprays and PARs) incorporated into the containment design to promote atmospheric mixing and to withstand the loads produced by hydrogen deflagration.
- A compartment to spread and cool molten core debris for long-term stabilization
- A SAHRS.
- Electrical and instrumentation and controls (I&C) systems dedicated to support severe accident mitigation features.
- A double-shell containment structure (i.e., Reactor Building and Shield Building) with a sub-atmospheric annulus.

These features verify that the U.S. EPR™ design has the ability to mitigate a broad spectrum of severe accident challenges and are consistent with advanced LWR expectations regarding severe accidents.

Probabilistic risk assessment (PRA) plays an important role in the development of U.S. EPR™ severe accident management strategies, including the task of identifying plant-specific severe accident challenges. PRA is also used in preparatory decision-making, defining symptoms and associated plant process parameters that must be monitored, selection of suitable strategies, development of severe accident management guidance, and in training. In addition, it can serve the needs of the onsite and offsite emergency organizations by giving an indication of the potential releases caused by severe accidents.

1.2.2 *U.S. Regulatory Insights*

There has been progressive development of both regulatory and industry guidance related to the development of SAMG since the original U.S. NRC policy statement on severe accidents in nuclear power plants in 1985 (Reference [3]). It became clear to the NRC and the industry, through the experience of the Three Mile Island Unit 2 (TMI-2) accident, the examination of PRA studies, and an increased understanding of severe accident phenomena (Reference [4]), that the remaining residual risk associated with severe accidents could be further reduced by the use of SAMG.

SECY-88-147 (Reference [5]) addresses the NRC integration plan for closing severe accident issues. In Section 6 of SECY-88-147, the NRC staff concluded that accident management can result in substantial reduction in risk from severe accidents and gives general guidance on a proposed accident management program plan. Accident management is defined as follows:

"... the measures taken by the plant operating and technical staff to (1) prevent core damage, (2) terminate core damage if it occurs and retain the core within the reactor vessel, (3) failing that, maintain containment integrity as long as possible, and finally (4) to minimize the consequences of offsite releases."

In the proposed outline of an accident management plan, the NRC staff refers to the following key elements: (1) prevention of core damage, (2) in-vessel accident management, (3) ex-vessel accident management, and (4) related activities including operator training and transition from EOP to SAMG. The staff expresses the NRC expectation that licensees make ample use of insights obtained during their individual plant examinations (IPE) in developing severe accident management guidance.

SECY-89-012 (Reference [6]) describes the major goals, framework, and elements of the NRC accident management program and the recommended approach for implementation of SECY-88-147. Accident management is referred to as an extension of the defense-in-depth principle as follows:

"Accident management, in effect, extends the defense-in-depth principle to plant operating staff by extending the operating procedures well beyond the plant design basis into severe fuel damage regimes, with the goal of taking advantage of existing plant equipment and operator skills and creativity to find ways to terminate accidents beyond the design basis or to limit offsite releases."

In SECY-89-012 the staff concludes that SAMG can reduce the risks associated with severe accidents by improvements in the following areas:

- Accident management procedures (taking advantage of PRA insights).
- Training in severe accidents.
- Accident management guidance (diagnosing progress of severe accidents and planning response).
- Instrumentation.
- Decision-making responsibilities.

GL-88-20 (Reference [7]) was released to instruct licensees on how to develop SAMG as part of the IPE process. In GL-88-20, the NRC identified three categories of severe accident management strategies:

- Conserving and/or replenishing limited resources during the course of an accident.
- Using plant systems and components for innovative applications during an accident.
- Defeating appropriate interlocks and overriding component protective trips in

emergency situations.

Table 1 of GL-88-20, Supplement 2, shows a list of example strategies derived from PRA studies divided into the categories identified above.

The NRC has not provided guidance on the development of SAMG documentation to the degree of which it has for the development of EOP (References [8] – [11]). Because SAMG are expected to smoothly interface with EOP, retaining analogous elements, nomenclature, and overall format is expected.

1.2.3 *Emergency Response Data System*

SAMG require an acceptable list of plant parameters used to track action time and monitor progress. 10 CFR 50, Appendix E defines the regulatory requirements regarding emergency planning and preparedness. Among the requirements specified in that section of the regulations, an emergency response data system (ERDS) is described. The ERDS serves as a "direct near real-time electronic data link between the licensee's onsite computer system and the U.S. NRC Operations Center that provides for the automated transmission of a limited data set of selected parameters." The ERDS will provide data for selected types of plant conditions. The information required for pressurized water reactors (PWR) is presented in Table 1-1. This list represents the minimum list of parameters to consider for the development of SAMG.

1.2.4 *Insights and Precedence from NEI 91-04 and EPRI TR-101869*

In the early 1990s, the Nuclear Energy Institute (NEI) and the Electric Power Research Institute (EPRI) prepared separate reports (NEI 91-04 (Reference [11]) and EPRI TR-101869 (References [12] and [13]), respectively) addressing severe accident issue closure guidelines. Based on these reports, the severe accident management goal is to enhance the capabilities of the emergency response organization to mitigate severe accidents and prevent or minimize any offsite releases. The severe accident management objective is to establish core cooling and verify that any current or immediate threats to the fission product barriers are managed. The severe accident management strategies should make maximum use of existing plant equipment and capabilities, including equipment and alignments that may not be part of the typical system.

Of particular importance to accident management is the information needed to respond to a broad spectrum of severe accidents supplemented by effective computational aids. To obtain information on plant conditions during a severe accident, instrumentation must be available. Therefore, during the development of SAMG, the availability and survivability of instrumentation needs to be evaluated for the domain of severe accidents. Computational aids (CA) should be developed as part of the severe accident management guidelines. The emergency response personnel will use the aids to evaluate key plant parameters and plant response relative to the accident management decisions. The aids are not required to be computer-based, but should be easy to use.

Collectively, NEI-91-04 and EPRI 101869 state that SAMG should provide the comprehensive guidance necessary to:

- Diagnose plant conditions – a symptom-based approach for evaluating plant conditions and challenges to plant safety functions (see Section 4.1.2).
- Prioritize response – relevant plant parameters and operating strategies reserved for accident management are prioritized based on the expected effectiveness of the action and time available for response.
- Assess equipment availability – availability of equipment for response is determined (a key item in this part of the process is prioritizing the recovery of equipment when it is not available).
- Identify and assess negative impacts – this part of the process includes the identification of additional actions that can mitigate the negative impacts.
- Determine whether to implement available equipment – based on a comparison of the negative impacts to the consequences of taking no action, the decision to implement a given strategy can be reached.
- Determine whether implemented actions take effect – after the strategy is implemented, it is necessary to know if the actions are effective and if the negative impacts are still acceptable.

- Identify long-term concerns for implemented strategies – after the strategy is implemented, there may be additional long-term actions required to maintain the strategy (e.g., refilling tanks).
- Provide a clear delineation of the flow of information, identification of the decisions that have to be made, and up front consideration of the viability of implementing alternate strategies.

1.2.5 *Consideration of International Regulatory Guides*

To confirm the thoroughness and broad acceptance of a set of SAMG, it is necessary to verify that the guidelines address requirements set forth by the numerous international nuclear safety organizations. Of particular relevance to the deployment of the EPR design world-wide are the Radiation and Nuclear Safety Authority of Finland (STUK), the International Atomic Energy Agency (IAEA), the French Nuclear Safety Authority (ASN), and the European Union. The respective documents for these organizations are as follows: STUK – YVL Requirements, IAEA – IAEA Safety Standards No: NS-R-1, "Safety of Nuclear Power Plants: Design" (Reference [14]), French Safety Authorities and EdF – Technical Guidelines (Reference [15]), European Union – (European Utility Requirements) EUR, Volume 3 of (Reference [16]). International collaboration organized by the Organization for Economic Cooperation and Development (OECD) compiles much of the state-of-the-art regarding SAMG development in References [17] and [18].

1.3 *Definitions and Acronyms*

Several technical terms are used in the development of severe accident related analysis and documentation. Those that frequently occur have been compiled in this section along with appropriate definitions and explanations.

Accident Prevention –All measures to prevent severe core damage, including: reducing the frequency or severity of challenging events; improving the reliability of plant equipment needed to respond to challenges; and the use of instrumentation and automatic or operator action to control events before severe core damage occurs.

Accident Management (AM) –The totality of measures, both short-term and long-term, taken by the plant operating staff to prevent accidents, to control the course of an accident in progress, and to mitigate the consequences of an accident during its occurrence.

Challenge – A condition that can lead to one of the containment failure modes. The phenomena during the severe accident scenarios may lead to such a situation.

Design Basis Accident (DBA) – In the event of a nuclear reactor accident, the principal concern is that the engineered safety systems will fail, resulting in a large release of radioactive material. A nuclear plant is, therefore, designed according to basic specifications that verify the capability of the plant to undergo a specified range of operational events, accidents, and external hazards within strictly limited radiological protection requirements. This design basis usually includes the specification of challenging events, important assumptions, and in some cases particular methods of analysis. A DBA is essentially a design tool to help make an engineering judgment on the appropriate safety margins for different component parts and systems of a nuclear plant. Therefore, the scenarios associated with a DBA should not be used to assess accident consequences because of the extreme conservatism placed on the basic assumptions.

Mitigation –All measures taken to limit the radiological consequences of an accident, including: limiting release into containment; limiting release from the facility; reducing public radiation exposure (e.g., by sheltering, evacuation, offsite cleanup). Release mitigation refers only to measures taken to limit the release of radioactive material from the facility.

Mitigation Path – A severe accident sequence in which the dedicated severe accident measures perform as designed.

Phenomenon – Used to describe physical characteristics of events (such as the formation of an oxide layer during the flooding of the corium).

Scenario – Accident scenarios started from the reactor trip and evolving to an accident sequence or even to the melting of the core.

Severe Accident – A severe accident is a category of beyond DBAs which result in catastrophic fuel rod failure, degradation of the structural integrity of the reactor core, and release of radioactive fission products into the reactor coolant system (RCS). Such an event can only occur as a result of a sustained loss of adequate core cooling, which leads to a build up of fission product decay heat and elevated core temperatures. The resulting consequence of melting the reactor core (and internals) may lead to the breaching of the reactor pressure vessel (RPV) and, through the relocation of molten core material into the containment, may potentially compromise the ability of the containment to perform its radionuclide retention function.

Severe Accident Management –Actions that are taken by the plant staff during the course of an accident to prevent core damage, terminate progress of core damage and retain the core within the vessel, maintain containment integrity, and minimize offsite releases. Severe accident management also involves pre-planning and preparatory measures for severe accident management guidance and procedures, equipment modifications to facilitate procedure implementation, and severe accident training. The overall objective is to further reduce the risks of large releases. It is the responsibility of the licensees to develop and implement a severe accident management program.

Strategies – Management practices aimed to mitigate the progression or consequences of an accident.

Table 1-1—ERDS Parameter List

Plant System	ERDS Parameter
Reactor Coolant System	RCS pressure
	RCS temperatures (hot leg, cold leg, and core exit thermocouples)
	Subcooling margin
	Pressurizer level
	Reactor coolant charging/makeup flow
	Reactor vessel level
	Reactor coolant flow
	Reactor power
Secondary Coolant System	Steam generator levels
	Steam generator pressures
	Main feedwater flows
	Emergency feedwater flows
Safety Injection System	Medium- and low-pressure safety injection flows
	In-containment refueling water storage tank level
Reactor Containment	Pressure
	Temperatures
	Hydrogen concentration
	Sump levels
Radiation Monitoring System	Reactor coolant radioactivity
	Containment radiation level
	Condenser air removal radiation level
	Effluent radiation monitors
	Process radiation monitor levels
Meteorological Survey System	Wind speed
	Wind direction
	Atmospheric stability

2.0 DECISION-MAKING PROCESS

Accident management is the implementation of actions for returning a damaged plant to a controlled, stable state. Fundamental to any decision-making process is the expectation of robustness, flexibility, and ease-of-use. OSSA end-products are built from analytical support studies used to develop a thorough understanding of plant behavior and responsiveness to operator action. The OSSA activity is guided by statements of specific safety goals and the identification of corresponding severe accident challenges to the plants safety functions (i.e., radiological transport, containment integrity, and heat removal) while anticipating the human factors needs of plant personnel.

Severe accident management actions require an informed assessment of the plant conditions, allowing for effective decision-making and prioritization. Lessons learned from the TMI Unit 2 accident and industry experience gained from the application and training with accident management programs lead to the conclusion that severe accident management diagnostics should be symptom based. Symptom-based guidelines and procedures remove the potential bias and uncertainty inherent in event-based approaches with a focus on recovery and reinforcement of safety functions. This approach provides the greatest degree of robustness and flexibility into the development of U.S. EPR™ severe accident management guidance, allowing plant personnel to efficiently transition between priorities.

The reliability of the decision-making process requires a degree of structure and ease-of-use. Human factor considerations must be addressed to verify that the guideline documentation is understandable and ergonomically functional. Instructions must be clear, concise, and well-organized. Hierarchical diagramming and flow charting are presentation methods that have been well-received by end-users and will be incorporated into the final products.

2.1 *Role of the Plant Personnel*

SAMG end-products serve to improve the ability of plant personnel to monitor, diagnose and influence the course of a severe accident. Ultimately, the responsibility for developing a staffing arrangement that meets all U.S. regulatory requirements rests with each individual combined license (COL) application. COL applicants referencing the U.S. EPR™ design will determine staffing levels and qualifications of plant personnel based on corporate staffing philosophy, existing site operations, fleet operations, and plant design. Nonetheless, the role of the plant personnel during a severe accident is to monitor the plant's principal safety functions and establish actions that maintain or recover the plant's defenses against radiological releases.

In case of a severe accident, the technical support team provides plant management and technical support to plant operations personnel in accordance with the emergency plan. The main control room operators will discontinue using the ongoing EOP and work with the TSC to develop and implement accident mitigation actions. A separate severe accident management guide will be used by the technical support team to help assess the accident conditions and determine which coping strategies need to be implemented. Such strategies should be implemented by the main control room operators either using appropriate procedures (or parts thereof) from the set of symptom-based emergency operating procedures or per 10 CFR 50.54(x) and (y), allowing departure from licensing basis (e.g., license condition or technical specification) without predefined operating procedures, according to the instructions of the technical support team.

When severe accident conditions are recognized, control room operators should begin monitoring and comparing process information and control system (PICS) values of critical variables with those on the safety information and control system (SICS) displays. As long as the values remain consistent and operator actions can be performed with consistent feedback, main control room operations may be conducted from the PICS using any other systems available. Should the PICS and SICS data be inconsistent and not within the established credibility criteria, PICS should be abandoned and operations continued from the SICS.

SAMG developed from OSSA are to be formatted to support division of responsibility between the shift manager and control room supervisor. The procedure strategy (i.e., the goal and overall direction of the function implementation) is flowcharted in a paper-based format. This is the document used by the shift manager. Individual tasks and subtasks are provided for the main control room staff in a computer-based format linked to the distributed control system process.

The TSC provides a location separate from the main control room where an independent plant technical support team can provide management and technical support to plant operation personnel during emergency, severe accident, and post accident conditions. TSC also provides a place where auxiliary personnel can relieve the main control room operators of peripheral duties and communications not directly related to reactor system manipulations. The TSC is housed with the necessary information and control system displays that are used for reviewing the accident sequence, determining appropriate mitigating actions, and evaluating the extent of any damage.

The shift manager will assess the plant symptoms to determine its state, and then evaluate the potential strategies that may be used to mitigate the event. With input from the TSC, the shift manager assesses and selects the mitigation strategies to be implemented. The plant operators are responsible for performing the steps necessary to accomplish the objectives of the strategies, such as hands-on control of valves, breaks, controllers, and special equipment.

The responsibility of various roles or actions is to be assigned based on an individual's position in the emergency organization and the person's ability to perform the required function. OSSA guidance for operators and the TSC are envisioned as follows:

Operator guidelines:

- Plant status monitoring.
- Monitoring for severe challenges (if applicable).
- Performing systematic actions (actions that should be done anyway, and do not require evaluation).

- Implementing appropriate mitigating actions.
- Verifying actions have been properly implemented.
- Monitoring the effects of the actions taken and feeding back to the TSC.
- Verifying that ongoing strategies can be maintained (e.g., refilling water sources).

TSC guidelines:

- Evaluating plant status, determining potentially applicable strategies (only where evaluation needed).
- Evaluating the positive and negative impacts of different potential strategies.
- Recommending strategies to be applied.
- Monitoring for achieving controlled stable condition/exit condition.

2.2 *Structure of U.S. EPR™ Guidance*

Severe accident management guidance is expected to have an organized structure to facilitate effective decision-making. For the U.S. EPR™, the form of this structure is based on customer SAMG methodology; however, the application of OSSA provides a model set of SAMG end-products. The U.S. EPR™ guidance for severe accident management will include overall diagnostic tools that control the flow of the decision-making process, as well as detailed guidelines. The following sections provide a summary of the expected decision-making flow charts, as well as further information on the content of the detailed guidelines. OSSA end-products support the U.S. NRC expectation for SAMG as identified in Regulatory Issue Summary 2006-21 which include:

- Diagnosis CAs.
- Severe Challenge Status Trees (SCST).
- Severe Accident Control Room Response Guides (SACRG).
- Severe Accident Guides (SAG).

- Severe Challenge Guides (SCG).
- Severe Accident Exit Guides (SAEG).

2.2.1 *OSSA Diagnostic*

It is generally accepted that the severe accident approaches shall not be based on a detailed progression of a severe accident. First, in most cases a severe accident is not based on a single simple failure and/or initiating event; second, the evolution of the sequence is complicated. However, it can be reliably monitored using available instrumentation.

One of the key aspects defining the structure of the severe accident management approach is the means used to monitor and assess plant conditions and identify potential actions for evaluation. This process is referred to as “diagnosis.”

In most of severe accident management approaches, when entry conditions are reached, an initial phase involves monitoring and assessing plant conditions. Following this, potential actions are identified and evaluated. A decision is made whether the actions should be taken, and if the decision is to act, the implementation of actions is performed and monitored.

The OSSA diagnostic covers the entire process from entry to exit of the OSSA guidelines. The diagnosis is based on three safety functions evaluated based on graphical computational aids (e.g., flow charts). The diagnosis process allows for different sets of actions to be considered, and prioritizes the evaluation process so that the different action sets are evaluated in an appropriate order (i.e., most important first).

OSSA diagnostics framework consists of easy-to-use end-products, such as:

- Diagrams and flow charts for diagnosis of the plant status in relation to a controlled stable condition.
- Challenge–system matrix for linking safety function challenges to systems capable of providing mitigative response.

Diagrams and flow charts for diagnosis specify key parameters to be monitored and controlled during a severe accident. They provide continuous monitoring of each key parameter until all parameters are in a state that the plant can be declared to be in a controlled stable state. Parameters reflecting severe accident phenomena that may challenge the fission product boundaries, such as those required for the plants ERDS, are to be included. If parameter values exceed setpoints specified for a controlled stable state, the shift manager and TSC evaluate the need to implement strategies to bring the parameter to a controlled, stable condition.

Challenge/system matrix follows key plant parameters (reflecting safety function status), which must be monitored on a regular basis to determine if their value exceeds a setpoint which indicates that a more serious condition exists. The challenge/system matrix is monitored in conjunction with the flow chart-based diagnostics for the evaluation of strategies. If a setpoint value in the challenge/system matrix is exceeded, a system-level severe accident management strategy is implemented to deal with the more serious condition.

Priority is expected to be established by the shift manager and TSC among the severe challenges in the challenge/system matrix. The diagnosis includes a high level monitoring scheme which allows a change of direction if inappropriate actions were taken, negative impacts of actions become unacceptable, or if a misdiagnosis occurs. This scheme will also include a check for success (i.e., controlled and stable conditions).

2.2.2 *Guidelines*

While the OSSA diagnosis framework is used to establish the organizational structure of severe accident management guidance, the details and the majority of the technical content are contained within guidelines. OSSA supports guideline development in the common severe accident management categories, including SACRGs, SAGs, SCGs, and SAEGs. Guidelines are referenced directly from the OSSA diagnostic when safety function status changes occur. As advised by NEI-91-04 and EPRI TR-101869, comprehensive guidance provides:

- Interface to the OSSA diagnostic – The OSSA diagnostic contains cross references to guidelines to efficiently direct emergency responders to accident management activities.
- Prioritization response considerations – OSSA guidance emphasizes the prevention and mitigation of potential radiological releases. Activities addressing containment integrity and heat removal that have an immediate effect on this principal goal of OSSA take precedence over actions that do not.
- Equipment availability assessment and actions - The guidelines identify the possible equipment that may be used to implement an action. If no equipment is available, instructions will include the consideration of restoring the non-functioning equipment.
- Identification and assessment of negative impacts - The benefits of candidate actions are weighed against the potential for negative impacts. If the negative impacts are judged to be acceptable, then methods to minimize the negative impacts are considered. If the impacts differ based on the choice of methods or equipment, this distinction will be made.
- Determination of action plan and effectiveness - If the decision is made to implement a strategy, implementation instructions will be provided that include any limitations identified during the evaluation. The implementation instructions will also identify the expected response of the plant as a basis to compare the actual response. The option to abort the action, or to implement additional actions, will also be considered.
- Identification of long-term concerns for implemented strategies - When a severe accident management strategy is implemented, there may be one or more additional plant parameters that require periodic surveillance to verify that the strategy implemented continues to be effective. These generally include support functions such as an adequate water supply and continued equipment cooling. The identification of long-term concerns associated with the implementation of any severe accident management strategy should also include a brief description

of the actions that can be taken to address long-term concerns when they become critical to the continuation of the selected strategy.

2.3 *Severe Accident Management Goals*

The U.S. EPR™ design has both systems and instrumentation for the mitigation and monitoring of severe accidents. A severe accident sequence in which the severe accident systems perform as designed is described as following the “mitigation path.” The “OSSA controlled and stable area” concept defines the targeted plant conditions, with heat removal from the core debris and from containment, and fission product releases terminated or reduced below an acceptable value. While the accident remains in the controlled area there is no challenge to the ultimate fission product barrier, and only a relatively few manual actions that must be taken. Parameters used to monitor for this condition should be stable or decreasing (i.e., at least trending in the right direction). When no setpoints can be used, trends are to be used.

The OSSA diagnostic is used to verify plant conditions and determine whether the accident is on a mitigation path and in the controlled area. Severe accident management strategies are implemented only when the plant state, as measured through the plants safety functions (heat removal, containment integrity, and radiological transport), deviates from the mitigation path or the controlled area. In these situations, OSSA guidance recommends strategies to bring the plant conditions back to the controlled and stable state.

2.3.1 *Core Heat Removal Safety Function*

The core heat removal safety function addresses potential plant severe accident challenges resulting from the failure of the core cooling or corium quenching. This safety function relates to the broader OSSA mission of managing a degraded core addressed in Section 3.6. The OSSA diagnostic for the core heat removal safety function is separated into two main parts:

- Core in-vessel: The time window starts at the entry of severe accident and stops if a vessel failure occurs. it aims to cover all situations of core degradation in-vessel.

- Core ex-vessel (i.e., relocated in spreading area): The time window starts at gate failure after temporary melt retention in the reactor pit, and stops when a stable state is reached and exit from OSSA is decided.

Core heat removal during the period of temporary melt retention is to be evaluated in an OSSA task to determine whether in-vessel core cooling strategies are more beneficial than allowing melt to relocate passively to the spreading area.

2.3.1.1 *Characteristic Challenged Core State Condition: In-Vessel*

Without adequate heat removal from the core, fuel elements will heat up. If the core is uncovered, fuel temperatures will rise rapidly; unmitigated melting of the fuel occurs. The generated melt spreads axially and radially within the core. Several solidification and remelting processes lead to the formation of a molten pool on top of the lower core support plate, which is enclosed by a crust. In case of a global failure of the crust and lower core support plate or molten pool penetration of the heavy reflector and the core barrel, the melt relocates to the lower plenum.

2.3.1.2 *OSSA Controlled, Stable Area for Core State: In-vessel Considerations*

As long as the configuration of fuel, at any time during the in-vessel phase, remains in a coolable geometry, the cooling of the core can be accomplished via several methods. If the core is to remain within the reactor pressure vessel, not only must the core initially be cooled, but a long-term heat removal process must be established. The first possibility to be considered is heat transfer to the steam generators. For this option to be feasible, there must be water inventory in the secondary side of the steam generators, the reactor coolant system (RCS) should be relatively intact to allow natural circulation, and there must be some water inventory within the RCS. However, it is not necessary to have a complete RCS water inventory because condensation of steam is also an effective heat transfer mechanism.

If the RCS is at high pressure and the core outlet temperature exceeds 1200°F, the primary depressurization system valves are opened by the operators. Assuming that-power is available to the safety injection (SI) and residual heat removal (RHR) system, long-term heat removal could come from the emergency core cooling system (ECCS) designed to respond to design-basis accidents (DBA). These systems actively circulate water from the in-containment refueling water storage tank (IRWST) through heat exchangers that transfer energy to the ultimate heat sink.

2.3.1.3 *Characteristic Challenged Core State Condition: Ex-Vessel*

If the severe accident is not mitigated before the RPV lower head fails and the core debris is transported ex-vessel, the only dedicated long-term heat sink is through the cooling chain provided by the SAHRS.

2.3.1.4 *OSSA Controlled, Stable Area for Core State: Ex-vessel Considerations*

Vessel failure can occur from a combination of thermal attack, elevated pressures, and dead weight of the corium. Upon failure of the RPV, corium in the lower head flows into the reactor pit. The U.S. EPR™ reactor pit is lined with a layer of sacrificial concrete on top of a zirconia brick layer. At the center-bottom of the reactor pit, a melt plug consisting of the same sacrificial concrete and backed by an aluminum and steel gate (no zirconia) acts as a check valve. The molten-core-concrete interaction (MCCI) on the predefined thickness of sacrificial concrete provides a temporary phase of melt retention in the reactor pit of which all remaining melt from the vessel is collected. Eventually, sufficient energy is imparted onto the melt plug and gate, and corium is allowed to flow freely through a corium transfer channel to a large spreading room. The arrival of corium in the spreading room passively initiates gravity-driven overflow of water from the IRWST which cools and quenches the spread melt from all sides.

Steam generated from corium cooling lifts and migrates throughout the containment, condensing on the cool surface of the large steel and concrete Reactor Building structures. Coalescing condensate drains into the IRWST, which is cooled by the dedicated SAHRS cooling chain.

2.3.2 *Containment Integrity Safety Function*

The containment integrity safety function addresses potential plant severe accident challenges resulting from failure to prevent large-scale core melting and breach of the RPV. This safety function relates to the broader OSSA mission of managing reactor debris, combustible gas, containment pressure, and temperature addressed in Sections

3.7, 3.8, and 3.9, respectively. Containment integrity can be verified when both immediate and long-term challenges, including containment bypass, overpressure, combustion, basemat ablation, steam explosion, and HPME, are resolved.

2.3.2.1 *Characteristic Challenged Containment Condition*

Among the several unique challenges to containment integrity from a severe accident in a U.S. EPR™ plant, overpressure, combustion, and basemat ablation are considered more credible. The natural, physical, and chemical processes occurring during a severe accident are expected to release heat in the form of saturated steam and hydrogen from in-vessel, metal-water reactions, and MCCI. Left unmitigated containment pressure, temperature, and hydrogen concentration could rise to levels approaching design limits.

2.3.2.2 *OSSA Controlled, Stable Containment State*

The mechanism for preserving containment integrity is the systems in-place or realignment for a controlled transference of energy bypassed to the containment, then delivered to the long-term heat sink. Regarding the issue of containment bypass during an accident, normal containment isolation is expected prior to the realization of severe accident challenges through the use of safety-related isolation valves qualified to full power RCS conditions.

The SAHRS is the main means available to operators for challenges of containment overpressure and basemat ablation. The SAHRS is designed with the capability to remove residual heat from the spread melt and control the containment atmosphere during a severe accident so that the containment pressure remains below the applicable load limits. The SAHRS performs its function in the short-term (from 12 hrs to several days) via the containment spray functionality. Following the short-term phase, the SAHRS can be operated in two modes. The first mode consists of a melt by direct cooling through a connection to the passive flooding line, providing forced water to the spreading area. The overflow rises up the steam chimney and returns back to the IRWST. The second mode involves spraying in the containment for atmospheric heat removal. After the spreading room is completely flooded with water, the molten corium forms a solid mass within days. Long-term stabilization can be accelerated through the recovery of any or all low head safety trains (i.e., increase containment heat removal).

The generation of hydrogen can occur in the U.S. EPR™ during a severe accident due to oxidation on fuel rod surfaces, MCCI, and oxidation of core support material. Hydrogen reduction in the U.S. EPR™ is achieved via 47 PARs, which are used to reduce hydrogen concentration in the containment atmosphere during a severe accident to minimize the risk of hydrogen deflagration and detonation. This is described further in Section 5.8.

While generally considered less probable, prevention of HPME and steam explosion scenarios is expected. If the RPV fails while the reactor coolant system is at a high pressure, several severe accident phenomena occur that can impact containment pressure, temperature, and concentration of fission products. To eliminate this possibility, the U.S. EPR™ PDS is manually actuated to rapidly decrease RCS pressures below the level of concern. Likewise, the design of the U.S. EPR™ plant eliminates credible scenarios involving water and corium coming together in configurations resulting in steam explosions.

2.3.3 *Radiological Releases Safety Function*

The radiological releases safety function addresses potential plant severe accident challenges resulting from the failure to contain releases from the plant radiological defenses. This safety function relates to the broader OSSA mission of managing radiological releases addressed in Section 3.10. Instrumentation monitors the radiological condition along possible release paths, from the source of the release (either airborne or waterborne) to the environment. Release paths are breaches in the containment boundary, allowing a direct interface with the potential fission product sources. The containment boundary includes the containment structure, the containment penetrations, the steam generators tubes, and the piping of systems connected to the RCS or containment up to the first (operable) isolation valve. Releases may be categorized based on the unique containment challenge mechanism as evaluated from PRA Level 2 analysis (see Section A.2.5). Accidents involving the spent fuel pool within the Fuel Building are also considered.

2.3.3.1 *Characteristic Challenged Containment Condition*

Sustained challenges to core heat removal and containment integrity increase the likelihood of elevated fission product inventory and breach of the last fission product barrier. Depending on the mechanisms that lead to this situation, fission product transport could occur across the physical barriers where the Reactor Building, Safeguards/Auxiliary Buildings, steam system, or Fuel Building interfaces with the outside environment. The environmental conditions at the source of the release can influence the rate of release. The most challenging situation is one in which the conditions from an airborne fission product source are at elevated pressure and temperature with minimal obstruction to the environment.

2.3.3.2 *Controlled, Stable Radiological State*

Control and stabilization of fission product releases require that the containment boundary is secured or that the leakage rate is eliminated. The primary objective towards this goal is to maintain existing containment boundaries while reestablishing barriers to fission product transport using valves, doors, or other means to block the flow of fission products. Actions supporting the isolation of containment, reduction of fission product inventory, and/or the reduction of fission product driving force are broadly considered. Stabilization of fission product releases occurs when either the fission product inventory is immobilized or from their isolation by securing the leak paths allowing environmental release.

During the early phase of a severe accident in a U.S. EPR™ plant, the fission product inventory airborne in the containment can be prevented or reduced by maintaining the RCS integrity, thereby, retaining a large fraction of fission products in the RCS. RCS depressurization relieves structural loads on the steam generator tubes and other interfacing systems, possibly preventing eventual escape through those pathways. In addition, flooding the steam generators to submerge the U-tubes provides a cool surface for fission product deposition and retention. Alternatively, potential leakage between the primary and secondary systems can be terminated by keeping the secondary system pressure slightly above the RCS pressure.

Within the containment natural processes passively remove airborne fission products by deposition onto cool surfaces and absorption into steam condensation. Steam released in containment condenses on the steel and concrete structures. This process transfers these now waterborne fission products to coalescing pools of condensate that eventually drains to the IRWST. Airborne fission products may also be scrubbed from the containment atmosphere using the SAHRS in containment spray mode.

Containment integrity can be recovered by resolving unintended leak paths. While low levels of leakage from these sources are permitted within the plant design basis, these are based on offsite dose limits and, as such, all leakage must be terminated during a severe accident. Such leakage can be terminated by closing all valves in the piping, closing doors, and other seals. In addition, water seals can be created by flooding select piping. As with the RCS, leakage from the containment can be terminated by reducing containment pressure to near atmospheric pressure. For waterborne contamination, using systems that keep all radioactive water within the containment IRWST are preferred.

3.0 OSSA MISSION REQUIREMENTS FOR THE U.S. EPR™ DESIGN

Managing a severe accident requires action on several fronts, sometimes simultaneously. Such action is triggered by an exceedance of specific instrumented setpoints or calculated measures. The following is a list of those challenges applicable to severe accident management of the U.S. EPR™ design. This list of challenges may apply during post reactor trip from full power, while at shutdown, or in both cases.

1. Response implementation and personnel training.
2. Control room/TSC plant state instrumentation.
3. Equipment survivability and recovery.
4. Containment isolation.
5. Degraded core.
6. Reactor debris (including fuel-coolant interactions).
7. Combustible gases.
8. Containment pressure and temperature.
9. Radioactivity releases.

By using this list of challenges while respecting the requirements set forth by both U.S. and various international safety authorities, OSSA is to be developed considering the unique U.S. EPR™ accident prevention and mitigation features and its specialized plant diagnostics and associated instrumentation.

3.1 *Management of Guideline Development*

Severe accident management provides further protection through documented symptom-based guidance or explicit procedures to the responsible emergency response teams. Documented guidance and/or procedures include the necessary instructions for the responsible emergency response teams so that the plant is set upon a mitigation path with the initial objective of establishing a controlled plant state leading to an eventual safe plant condition. The combined effect of each of these systems, a robust and leak-tight containment, and the OSSA-derived SAMG verifies that for the U.S. EPR™ design the offsite dose following a severe accident is acceptable.

Human factor considerations must be addressed to confirm that the guideline documentation is understandable and ergonomically functional. Instructions must be presented in a clear, concise, and well-organized manner. Hierarchical diagramming and flow charting are presentation methods that have been well-received by end-users.

3.2 *Management of Response Implementation and Personnel Training*

Devising procedures for accident management actions are directions received by the operating staff that are specific and in a familiar format. The essential starting point includes an overall structure that clearly delineates responsibilities and any transfer of responsibilities during the development of an accident. As such, an important step in the implementation of OSSA is identifying the roles of the emergency response team members, defining functions – actions which need to be taken to respond to a severe accident situation, and assigning these functions to the emergency response team members. Ultimately, this is the responsibility of the COL applicant. The onsite organizational responsibilities of a typical emergency response team may resemble the example shown in Table 3-1.

In transitioning from EOP to the OSSA domain, the shift manager must promptly perform their responsibilities (i.e., assess and prioritize challenges and authorize strategy implementation). This necessitates reliable plant diagnostics and a clearly proceduralized entry process. It requires that the entry symptom(s) is well-defined and that entry to the guidance is independent of other actions occurring prior to and following reaching the entry criteria.

Given the separation of authority between emergency response teams, the transition between EOP and OSSA requires an orderly exchange of responsibility. It should not be assumed that plant staff familiarity with the SAMG is limited; but, it is very important that the guidance is easy to use and that appropriate training is provided.

The ability of plant personnel to monitor, diagnose, and take action during the course of a severe accident should be periodically assessed and continuously improved. Operator training programs must take into account accident diagnosis and management beyond normal operating transients and anticipated operational occurrences, from the earliest precursors to the eventual recovery of radiological protection measures.

Emergency organizations shall train and practice to verify the correct usage of the OSSA end-user products. Training will emphasize a selection of more likely sequences as quantified by PRA results, supplemented by a subset of low frequency, high risk sequences. Simulator training and table-top exercises can play an important role meeting this objective. Practice with these tools not only qualifies the trainee; but, also, provides supplemental verification and validation (V&V) of the actual plant-specific, OSSA-derived guidance.

3.3 *Management of Control Room/TSC Plant State Instrumentation*

To return the reactor to a safe state from the control room or the TSC, instrumentation must be available during a severe accident to provide adequate information on plant status. When implementing a strategy in a given plant condition, operators need to know:

1. When to initiate a procedure for that strategy.
2. That the procedure has been initiated.
3. That the procedure is effective.
4. If the procedure is ineffective, when to abandon it and what to do next.

Instrumentation and indicators that can relay plant information and the level of severity of an accident include those identified in this section and computer aids providing data such as reactor coolant subcooling margin, the threat of reactor vessel melt-through or hydrogen combustion, and location of reactor debris (i.e., in-vessel and ex-vessel).

These measures are the key constituents in the OSSA diagnostic. Among the more critical measures are those on which the emergency response teams rely on to transition between EOP and OSSA procedures (e.g., such as core outlet temperature and containment radiation).

The U.S. EPR™ design includes I&C that are part of the overall severe accident management concept. These I&C functions can be categorized as (1) those necessary to perform operator action, and (2) those necessary to closely monitor the progression of a severe accident. Specific I&C can be further identified by association with those severe accident features used to mitigate the effect of a severe accident as follows:

1. Monitoring core heat removal.
2. Supporting RCS depressurization.
3. Monitoring melt progression.
4. Monitoring hydrogen mitigation.
5. Monitoring containment heat removal.
6. Monitoring radiation levels/releases.

While other I&C may be necessary to support the ultimate strategies for severe accident management, the functionality of the primary plant I&C facilitating severe accident management is described in the following sections.

3.3.1 *Monitoring of Core Heat Removal*

Recovery of core heat removal is the primary objective during the early phase of an accident. The U.S. EPR™ design includes the following provision to support plant operators:

Measurement of Cold and Hot Leg Temperatures. Allows the operator to assess core heat transfer.

Measurement of RCS Pressure. Collectively with cold and hot leg temperatures, provides the ability to calculate subcooling margin, which allows the operator to anticipate core uncovery.

Measurement of Charging/MakeUp Flow. Provides the ability to assess RCS liquid inventory.

Measurement of Reactor Vessel Level. Provides the ability to assess liquid inventory in the RPV.

Measurement of Reactor Coolant Flow. Provides the ability to assess coolant delivery.

Measurement of Reactor Power. Provides the ability to assess core heat load.

Measurement of Steam Generator Liquid Levels. Provides the ability to assess primary-to-secondary heat removal capability.

Measurement of Steam Generator Pressure. Provides the ability to assess primary-to-secondary heat removal capability.

Measurement of Main Feedwater Flow. Provides the ability to assess primary-to-secondary heat removal capability.

Measurement of Emergency Feedwater Flow. Provides the ability to assess primary-to-secondary heat removal capability.

Monitoring of Safety Injection System Flows. Provides the ability to assess coolant delivery to the RCS.

3.3.2 *Support of RCS Depressurization*

The U.S. EPR™ design includes depressurization valves as part of the PDS to verify that a core melt does not progress ex-vessel under high pressure conditions. The system is actuated manually based on a core outlet temperature setpoint. The U.S. EPR™ design includes the following provisions to support reliable RCS depressurization:

Measurement of Core Outlet Temperature. Provided to allow the operator to anticipate the onset of core damage. Used in the U.S. EPR™ design to signal the transition from EOP to SAMG.

Manual Actuation of PDS Valves. Provides the ability to depressurize the RCS during a severe accident.

Position Indication for PDS Valves. Provides the ability to monitor the status of the PDS valves.

Measurement of RCS Pressure. Provides the ability to monitor the effectiveness of RCS depressurization prior to failure of the RPV.

3.3.3 *Monitoring of Melt Progression*

The U.S. EPR™ design uses a dedicated core melt stabilization system (CMSS) to bring molten core debris released from the RPV into a safe, stable condition. Measurements are provided within the plant to monitor the progression of the core melt, including:

Monitoring RPV Failure. Thermocouples in the RPV insulation are used to measure the outside temperature of the RPV lower head. The temperature evolution of the RPV lower head allows the operator to predict the onset of RPV failure. Failure of the thermocouples in the RPV insulation provides the operator indication that the RPV has failed.

Monitoring Corium in the Spreading Compartment. The arrival of molten core debris within the spreading compartment triggers the actuation of the SAHRS passive flooding valves. Position indication of these valves allows the operator to determine that the conditioned core melt has flowed into the spreading compartment. IRWST level indication provides redundant information relative to passive flooding initiated by molten core debris in the spreading compartment.

Monitoring Basemat Failure Threat. Thermocouples located in the central cooling water supply duct of the CMSS cooling structure allows the operator to determine if molten core debris has entered the cooling channels either through increasing temperature readings or a loss of function.

3.3.4 *Support of Hydrogen Mitigation*

The U.S. EPR™ design uses a combustible gas control system (CGCS) to control post-accident hydrogen within the containment. While this hydrogen mitigation process is entirely passive, dedicated measurements are provided within the plant to monitor the progression of its effectiveness.

Measurement of Hydrogen Concentration. Hydrogen concentration is monitored in various parts of the containment including the upper dome and steam generator, pressurizer, and pressurizer valve compartments. Hydrogen concentration measurements allow the effectiveness of recombination to be monitored as well as the potential for combustion within the containment.

Actuation of Hydrogen Mixing Dampers. Provides the ability to open the mixing dampers either automatically on measured containment pressure or manually from the control room.

Position Indication of Hydrogen Mixing Dampers. Provides the ability to monitor the state of the mixing dampers.

3.3.5 *Monitoring of Containment Heat Removal*

The U.S. EPR™ design uses the SAHRS, IRWST, and the component cooling water system (CCWS) to control the long-term, post-accident, environmental conditions within the containment. To control containment pressure, the SAHRS is operated in an active mode with either a containment spray or long-term recirculation. This system is manually started on a defined containment pressure or approximately12 hours after declaration of a severe accident, and supported by the dedicated cooling chain. The U.S. EPR™ design includes the following provisions to monitor containment heat removal:

Measurement of Containment Pressure. Provided to identify the need for active containment cooling.

Measurement of IRWST Temperature. Provided to monitor IRWST temperature and measure system performance.

Measurement of IRWST Water Level. Provided to monitor remaining water level available and measure system performance.

Measurement of SAHRS Heat Exchanger Inlet Temperature. Provided to monitor SAHRS performance.

Measurement of SAHRS Heat Exchanger Outlet Temperature. Provided to monitor SAHRS performance.

Measurement of SAHRS Flow Rate. Provided to monitor SAHRS performance.

Measurement of SAHRS Sump Level. Provided to identify fluid leakage from the SAHRS train.

Measurement of SAHRS Pump Inlet Pressure. Provided to identify sump strainer clogging and the need to align the system for operation in back flush mode.

Measurement of CCWS Heat Exchanger Inlet Temperature. Provided to monitor CCWS performance.

Measurement of CCWS Heat Exchanger Outlet Temperature. Provided to monitor CCWS performance.

Measurement of CCWS Flow Rate. Provided to monitor CCWS performance.

3.3.6 *Monitoring of Radiation Levels/Releases*

The primary mission of any accident recovery initiative is to minimize radiological consequences. The U.S. EPR™ design includes the following provision to support plant operators:

Measurement of reactor coolant radioactivity. Provided to monitor radiological release potential of the RCS.

Measurement of containment radiation level. Provided to monitor radiological release potential of the containment atmosphere.

Measurement of condenser air removal radiation level. Provided to monitor radiological release potential in the condenser air removal system.

Measurement of local plant wind speed and direction. Provided to monitor potential radiological release dispersion.

3.4 *Management of Equipment Survivability and Recovery*

The availability and survivability of the equipment and information sources is necessary for effective accident management following a severe accident. Regarding information sources, the reliability of these sources to provide indication of sufficient accuracy for their intended use is also needed. A key activity during a severe accident will be to maximize equipment and monitoring capabilities.

OSSA benefits from safety-related requirements associated with DBA response. The in-vessel conditions are accurately represented in the main control room. Because most of these instruments and controls support design basis functions, they are designed to meet the applicable code or standard defining equipment qualification. For the longer term in which accident management is addressed solely within the OSSA response domains, the instrumentation required for severe accident mitigation are designed to withstand severe accident environments they would experience in postulated accident scenarios, for the duration in which they are needed, including the effects of pressure, temperature, and radiation.

Similarly, the systems relied upon by emergency response teams to bring the plant to a safe state must also address survivability criteria for environmental conditions anticipated during a severe accident within the RCS and containment. Those specifically dedicated for severe accident response include PDS valves, CGCS, CMSS, and SAHRS.

The PDS, CMSS, and CGCS components are located inside the containment and are assessed for survivability to local ambient conditions (e.g., pressure, temperature, humidity, radiation). While the SAHRS is used to limit the pressure and temperature inside the containment, its main components (e.g., heat exchanger and pump) are not located inside the containment. These components only need to address survivability to elevated temperature and radiation doses inside the compartments in the Safeguard Building where they are located. Containment isolation valves, containment penetrations, air locks, hatches, and gaskets are required to maintain their leak-tightness during a severe accident. This equipment is assessed for survivability during elevated pressure and temperature.

Although not required for systems addressing beyond-design-basis accidents, the consequences of degraded performance of these systems have been addressed through design. PDS and CMSS flooding from the SAHRS are designed with two piping trains, providing operational redundancy. For example, if a valve fails to open, the valve line in the redundant piping is sufficient to accomplish its mission. Regarding the CGCS, there are 47 passive PARs. The regular testing and maintenance required by GDC 42 and 43 verify availability should a severe accident occur. A degree of degradation appears as an uncertainty parameter in U.S. EPR™ severe accident analyses. The active features of the SAHRS are located in a Safeguards Building, thus accessible to plant staff at all times. Staff could be assigned to perform maintenance required for the SAHRS pump or valve alignment.

The capability to repair and maintain equipment following the onset of a severe accident is also important. The onset of a severe accident involves the failure of plant equipment protecting the initial fission product barriers. The challenging local conditions occurring during a severe accident may contribute to additional malfunctions in equipment useful during the recovery. In addition, the severe accident progression or actions taken to recover from severe accident conditions may compromise the habitability, particularly due to high radiation levels, of certain plant areas. Plant operators and emergency responders need to quickly assess the situation and identify and prioritize opportunities for equipment recovery and maintenance actions. As in the case of environmental conditions and power supplies for equipment operability, severe accident management decisions should take into account the habitability of plant areas in which alignment, maintenance, or repair of equipment enhance the recovery capabilities.

Steam explosion and HPME are two low-probability, high-consequence events that could seriously challenge the survivability of equipment near the reactor cavity. Design features such as maintaining a dry reactor cavity and effective shadowing resulting from a tortuous path from a possible high energy transport of steam or corium minimize the probability of such occurrences. An effective SAMG program must consider the impact of such events on equipment survivability – in particular, instrumentation. As such, a task of OSSA is to identify alternative signals correlated to safety function performance to accommodate the possibility that the primary signal is lost or becomes unreliable.

3.5 *Management of Containment Isolation*

Containment isolation systems are considered reliable for use during a severe accident. The containment isolation is verified and performed either prior to or shortly following transition into OSSA.

The following specific safety provisions are provided for the power supplied to containment isolation valves:

1. Electric motor-operated containment isolation valves inside containment are supplied from (IEEE Standard 384) Class 1E 480V busses and are backed up by batteries and emergency diesel generators (EDG).

2. Electrical motor operated valves (MOV) outside containment are supplied from Class 1E 480V buses normally backed up by the EDGs, and can also be supplied from a severe accident uninterruptible power supply (UPS) (12-hour battery) with manual operator action. The severe accident power supply UPS (12-hour battery) is backed up by the station blackout diesel generators.

3. The success criterion for the containment isolation function is the closure of at least one valve in each containment release path. Common-cause failures are considered for MOVs and check valves that are identical and fulfill similar functions under similar operational and environmental conditions.

3.6 *Management of a Degraded Core*

Management of a degraded core begins with EOP-defined actions to revive reliable core heat removal. In the U.S. EPR™ design, heat removal requirements depend on the progression of the severe accident. Early response emphasizes either recovery of secondary cooling or primary feed and bleed.

Secondary cooling with the steam generators is sufficient for transients or events where RCS integrity is maintained (no loss of coolant accident (LOCA) condition). This can be satisfied with one main feedwater (MFW) pump, or feedwater startup-shutdown system (SSS) pump, or one emergency feedwater pump supplying one steam generator with steam relief to the main condenser through the main steam bypass, or to the atmosphere through a main steam relief valve or main steam safety valve (two per steam generator). If secondary cooling is unsuccessful, the operators initiate primary feed and bleed cooling.

Primary bleed is initiated through pressurizer safety valves or PDS valves, and feed is provided by non-safety-related chemical volume control system (CVCS) or an SI train. The heat transferred to primary containment is removed by IRWST cooling. Low head safety injection (LHSI) trains with heat exchangers or the SAHRS provide the IRWST heat removal function. Inventory make-up to the reactor vessel can be provided by the safety-related accumulator and SI functions and also from the CVCS and extra borating system (EBS).

If these actions are unsuccessful and core temperatures continue to rise above 1200°F (650 °C), operators permanently deviate from EOP and initiate OSSA-derived procedures. The immediate action following this transition is primary system depressurization. Primary system depressurization is a keystone severe accident management action in the U.S. EPR™ design performed during the transition from EOP to OSSA. While RCS depressurization may allow net SI, allowing preservation of the RPV, it also effectively eliminates the possibility of HPME and subsequent direct containment heating (DCH). When conditions requiring entry to OSSA are reached, a final attempt for primary depressurization is performed. It is appropriate to verify the performance of the primary depressurization system as a systematic immediate action. At this transition the accident management strategy is redirected from operating strategies supporting the return of core heat removal to operating strategies focused on arresting further core degradation, possibly leaving the reactor pressure vessel intact.

Core damage and melt progression can be arrested if injection to the RPV can be re-established. If the capability to inject into the reactor system is recovered, but with limited flow capacity, it is important to provide clear and unambiguous guidance on what action to take if the recovered flow capacity is less than the necessary flow rate expected to quench the core.

There is a general agreement that the hazards posed by increased hydrogen generation, possible recriticality, and increased steam production do not outweigh the benefits of retaining the degraded core inside the vessel. The criteria generally followed for this action is to supply water to the reactor vessel as soon as injection capability is available. It is appropriate, nonetheless, to include warnings concerning side effects of increased hydrogen production in OSSA.

3.7 *Management of Reactor Debris*

Reactor debris presents several containment integrity challenges including hydrogen production, generation of hot gases, fuel-coolant interaction (FCI), RPV failure, and basemat integrity. The reactor debris state is considered to be the condition in which core material has melted and relocated either within the core or into the lower head region.

Following the expected event progression described in Attachment C, an early concern regarding reactor debris (besides hydrogen production) is the generation of hot gases. Circulation of these gases through the RCS can result in damage to the piping. Of particular concern are the steam generator tubes. Failure of the steam generator tubes introduces a path leading beyond the containment, assuming downstream containment isolation fails or is otherwise not attempted. The sustained contact with elevated gas temperatures may result in piping creep, which may eventually rupture without sufficient mitigation.

An emergency response team can protect the steam generator tubes against creep failure by depressurization of the primary system, which, if successfully performed immediately on entry, obviates the need to place very high priority on refilling steam generators. Creep rupture of a hot leg prior to the rupture of the steam generator tubes would have the same effect (RCS depressurization).

Melt retention within the RPV was not the primary focus for the U.S. EPR™ design. Rather, the U.S. EPR™ is equipped with a dedicated ex-vessel system to accommodate molten core debris, including the entire core inventory and reactor internals, which penetrates the RPV. However, preservation of the RPV is an outcome of successful plant response from procedures defined in the OSSA Phase 1 domain (see Section 4.1.1). These events are described as “limited core damage sequences.” RCS depressurization and the recovery of core heat removal systems are operator actions that can contribute to the success of this objective.

FCI is a process by which molten fuel transfers its thermal energy to the surrounding coolant, leading to break-up of corium with possible formation of a coolable debris bed or potential evolution to an energetic steam explosion. Two modes of contact between the molten corium and coolant are considered while the corium remains in-vessel:

1. A pouring contact mode, where corium is poured into a pool of water. This mode could conceivably occur within the RPV when corium relocates into the water-filled lower head of the vessel.

2. An injection or stratified contact mode, where a pool of corium is flooded by water. This mode can occur within the RPV as a consequence of reflood of the RPV, or later, during either molten pool formation inside the lower head or the designed flooding of the melt in the spreading area.

The in-vessel FCI threat is assumed to be realized as a large steam explosion causing a breach of the RPV that generates containment-failing missiles while the ex-vessel threat is related to global pressure and temperature effects. For the in-vessel threat, extensive research, and elicitation of experts (see Section 5.4.2 of Reference [1]) conclude that the risk of containment failure from steam-explosion-induced missiles is negligible in LWR designs. The design of the U.S. EPR™ RPV is consistent with current LWR designs and includes a missile barrier to eliminate any direct pathways from the reactor vessel to the containment.

With regard to the preclusion of ex-vessel steam explosions, the U.S. EPR™ design supports an initially dry reactor cavity and spreading area, the addition of silica-rich sacrificial material to the melt before ex-vessel flooding, and the controlled addition of water to the top of the melt after spreading. Tests have been performed examining the addition of small amounts of water (simulating condensation effects) and large amounts of water introduced both prior and following the appearance of hot molten material. While violent boiling is commonly observed, FCI, involving the breakup of molten particles, was not observed in prototypic scenarios.

3.8 *Management of Combustible Gases*

The U.S. EPR™ design relies on the use of 47 PARs for the reduction of H_2 and CO concentrations while keeping containment pressure low. PARs have the distinct advantage that they require no operator action. PARs work both individually, as a remover of free hydrogen in the containment, and collectively, to drive atmospheric circulation into the containment, thus, encouraging the homogenization of hydrogen. PARs use a catalytic coating to transform molecular hydrogen and oxygen into water vapor.

PARs are self-starting and self-feeding, even under cold and wet conditions. They can also reliably perform under steam-inerted atmospheres, very low H_2 concentrations, and in the presence of aerosols (Section 5.1 of Reference [1]). The buoyancy of hot gases expelled at the top of a PAR vertical flow channel sets up natural convective flow currents that promote mixing of combustible gases in the containment. Recombination of these gases commences as soon as hydrogen is released into the containment as a result of a design-basis or severe accident.

The aim of the recombiners is not to prevent hydrogen combustion but to limit the consequences, in particular to avoid containment failure. The period of greatest concern for a combustion event is during the period of in-vessel hydrogen release, expected to be pronounced during primary system depressurization. This will result in a strongly non-uniform hydrogen concentration (rising plume), dependent on characteristics of the initiating event and timing of PDS actuation.

Reactor vessel failure resulting from lower head ablation will lead to molten debris in contact with concrete. MCCI introduces a second source for hydrogen production; however, high temperatures associated with the molten debris auto-ignite this source and result in a standing flame. As such, no additional hydrogen is added to the containment at-large as a result of this source.

3.9 *Management of Containment Pressure and Temperature*

The addition of mass and energy into a contained volume typically results in increased pressure and temperature within that contained volume. The U.S. EPR™ containment is designed so that containment pressure and temperature are rapidly reduced and maintained at acceptably low levels following any breach of the primary or secondary coolant circuits, thus confirming that the design leak rate is not exceeded. In response to design-basis events, the large free volume and heat capacity of the containment and internal structures means the U.S. EPR™ design does not require active containment heat removal systems to verify short-term pressure and temperature control. Steam condenses on these surfaces and drains to the IRWST. IRWST heat removal is provided by the SAHRS or LHSI heat exchangers located outside of the containment.

Under a design-basis LOCA, SI pumps draw water from the IRWST and reject containment heat to the CCWS and the essential service water system through the LHSI heat exchanger. The cooled LHSI water is then split between SI to the RCS and return to the IRWST for direct cooling of the IRWST.

During a severe accident, the primary sources of mass and energy that could cause containment over-pressurization can occur as the result of RCS depressurization (either by LOCA or actuation of the PDS valves) coupled with the generation of non-condensable gases from MCCI and steam addition resulting from quenching and stabilization of molten core debris in the spreading compartment. Following the initial pressure rise from RCS depressurization, the containment pressure is passively moderated by the heat capacity of the containment walls and internal structures. Further pressure reduction occurs first by the CGCS, which, through the use of PARs, in the presence of oxygen, recombines hydrogen and oxygen into water vapor. The recombination of hydrogen alone does not impact containment pressure; however, the conversion of hydrogen into a condensable form enhances the performance of the SAHRS containment spray by maximizing the water vapor concentration.

Like primary depressurization, actuation of the SAHRS containment sprays is a keystone severe accident management action used after approximately 12 hours into a severe accident. This delay allows the PARs to function optimally to bring combustible gas concentrations to a level that eliminates the possibility for combustion. This feature decreases containment pressure by condensing the steam generated in the containment and reduces the potential for further pressure increase by removing decay heat from within the containment airspace and from the molten core material in the spreading compartment.

Containment venting could be made available in the U.S. EPR™ design to avoid late failure due to over-pressurization. Containment venting is a backup response in the event that there is a complete loss of containment heat removal capability.

3.10 *Management of Radiological Releases*

The containment stands as the last barrier preventing releases of fission products. The final defense-in-depth goal is the mitigation of such radiological releases. To achieve the goal of terminating fission product releases and eliminating release pathways from the plant, several conditions must be met:

1. The isolation of the containment boundary, including penetrations and steam generator tubes, must be maintained.

2. The fission product inventory of the containment atmosphere must be minimized.

3. Significant leakage (i.e., driving force) though the containment boundary must be stopped.

The containment boundary interfaces with the environment both directly and indirectly through the Safeguards, Auxiliary, and Fuel Buildings. This boundary includes the containment structure, steam generators tubes, and piping of systems interfacing with the RCS or containment up to the first (operable) isolation valve. Isolation of the containment boundary includes:

- Maintaining existing containment boundaries.
- Closing appropriate valves that isolate systems directly connected to the containment atmosphere or the reactor coolant system.
- Creating a water seal whose static head is greater than the driving force where the first two methods are not available.

A key objective of the U.S. EPR™ design and associated SAMG is that large early releases are "practically eliminated." The U.S. EPR™ design has several features and available emergency response actions to address the mitigation of large radiological release.

A strategy for reducing the inventory available for release into the containment commonly considered in conventional PWRs, is the initiation of containment sprays. While for the U.S. EPR™ design the SAHRS has emphasized steam condensation and pressure suppression in the containment during a severe accident, the sprays can produce effective aerosol deposition due to interception of droplets. Sprays can remove some of the gaseous molecular iodine. The effectiveness of sprays depends on the availability of AC power and the extent of the areas covered by the spray system. Iodine volatility can be reduced by means of additives that are included in the design of IRWST or the SAHRS.

The U.S. EPR™ Reactor Building and Shield Building are physically independent except at the basemat. The annular space between these structures is maintained at sub-atmospheric pressure by the annulus ventilation system (AVS). The AVS is a safety-related system used in the event of a DBA or severe accident to filter any leakage from the Containment Building prior to exhausting it from the plant stack. The AVS provides 2 x 100 percent extraction capability and consists of high efficiency particulate air (HEPA) filters and charcoal absorbers in series with air handling equipment.

Reduction in the probability of a significant steam generator tube rupture (SGTR) may be addressed by secondary side flooding. Flooding the impacted steam generator to a level above the rupture location creates an effective seal for mitigating releases directly to the environment. This process also serves to prevent creep-related damage to the steam generator tubes.

In some circumstances an action may be necessary that requires an intentional, controlled, and short-term fission product release to prevent a larger, uncontrolled, and long-term release. Specifically, this is in reference to containment venting, if there is believed to be an immediate threat to the integrity of the containment structure. Any action that violates the primary goal of terminating fission product releases should be done in a manner that minimizes the release. Another example is the case of steam generator depressurization. There are pathways that blow down directly to the environment, and other pathways (such as through the condenser) that would allow fission products to be scrubbed.

Table 3-1—Emergency Response Team Responsibilities

Operations	

Shift Manager	• Transition from EOP to OSSA
	• Assess and prioritize challenges
	  ○ Safety function status monitoring
	  ○ Monitor plant response to actions
	  ○ Monitor for exit conditions
	• Authorize strategy implementation

Operators	• Perform immediate actions
	• Implement new actions
	  ○ To implement new strategies
	  ○ To verify ongoing strategies can be continued

Damage controllers	• Implement actions to recover failed equipment
Technical Support Center	
Emergency Director	• Recommend new strategy implementation

Technical Support Center	• Identify and prioritize candidate actions
	• Evaluate candidate actions
	  ○ Prioritize / review status of equipment recovery
	  ○ Assess positive and negative aspects of new strategies
	  ○ Recommend implementation of new strategies and identify any limits

Emergency Response Center	
Emergency Center Team	• Evaluate offsite consequences of recommended actions (if applicable)

Note:

This is an initial overview of the roles and responsibilities (R&R), and the detailed R&R will be drafted according to the requirements set for by 10CFR50.47, NUREG-0654 rev 001, and Reg. Guide 1.101.

4.0 OSSA DESIGN AND DEVELOPMENT METHODOLOGY

As a mitigative accident management guidance package for the development of SAMG, OSSA considers the best-practices and innovations within this field. OSSA streamline the guidance material while verifying comprehensive guidance for response to any severe accident. OSSA is similar to conventional plant SAMG in that it uses a ‘reference plant’ concept to set up the overall structure, principles, diagnostics, and strategies. .

The primary goal and purpose of OSSA is to address the regulatory expectation for the development of a comprehensive accident management plan for severe accidents. Provisions to cope with severe accidents including core melt situations are included in the U.S. EPR™ design, by inclusion of specific design features. Situations that would lead to large early releases such as containment bypass, strong reactivity accidents, high pressure core melt, or global hydrogen detonation, are practically eliminated by design. Early containment failure, around the time of vessel failure, would have the highest consequences (in terms of dose) to the public. It is, therefore, a major goal of the U.S. EPR™ design to suppress early containment failure by design measures linked to the most important phenomena. Ultimately, the OSSA methodology provides the technical basis supporting 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.

OSSA emphasizes severe accident management strategies to plant-specific containment challenges, rather than specific scenarios to verify the protection of the containment regardless of the accident sequence. This requires a symptom-based approach; thus, all sequences may be dealt within OSSA, even if it has not been analyzed or quantified through the Level 2 PRA or through the support safety studies. Identification of these challenges is evaluated through preliminary analysis (both deterministic and probabilistic) of the plant response to a broad spectrum of severe accident challenges. The diagnostic element of OSSA, a product of OSSA providing the selection of applicable mitigation strategies, is based on the list of plant-specific challenges

4.1 *Principal Accident Management Elements of OSSA*

OSSA considers the full spectrum of severe accident management from the EOP to plant recovery. In doing so, several technical and policy issues must be addressed. The principal elements of OSSA are as follows:

- Entry and exit conditions – The definition of the entry and exit conditions for the severe accident management guidelines identifies measures for the onset of core damage and the establishment of a controlled stable end state.
- Diagnostic - The OSSA diagnostic element reflects the rationale and process leading to different management strategies.
- Ergonomic - A severe accident is a stressful situation, where the decision depends on a good evaluation of available strategies and their consequences on the situation.
- Coordination with regional emergency response policy - The content of the OSSA will be consistent with the existing policy from the national, state, and local emergency jurisdictions and related crisis organizations.
- Training – Appropriate staff training improves performance within all of the above severe accident management elements.

Ultimately, the OSSA approach for the U.S. EPR™ design is based on the definition of an appropriate mitigation path dependent on the plant state described by symptoms interpreted in the control room or the TSC. The mitigation path corresponds with all severe accident sequences in which the severe accident systems perform as designed.

4.1.1 *OSSA Entry/Exit Criteria*

Control room response to any accident or abnormal event is dictated by EOP. OSSA should be entered when significant fission product release from the fuel has started or is imminent. Generically, a severe accident is declared when a specified set of plant conditions is met. At that time, control switches to the appropriate OSSA management guidance. Various measures have been considered in making this transition. For example, Combustion Engineering plants use indications of core uncovery, Westinghouse plants monitor a measure of the time available before recovery of injection may not prevent damage, and Électricité de France uses indications of fission product releases. For the U.S. EPR™ design, core outlet temperature is the primary measure of core state.

The transition to OSSA-based guidance places an emphasis on reliable information from plant instrumentation. Instrumentation and control qualification for both design basis and beyond DBA conditions is essential. In addition, signal redundancy is necessary. For the U.S. EPR™ design, core outlet temperature is considered one of the most direct measures of core cooling, and experience has shown that instrumentation has sufficient accuracy under a broad spectrum of conditions. Containment radiation is a useful redundant measure for the OSSA entry conditions. Its advantage is that it generally applies for all plant states. Given the uncertainties in predicting containment radiation response, it is better to give priority to core outlet temperature whenever it is available.

The initial response in the U.S. EPR™ power plant following indication of high core outlet temperature is primary system depressurization. Primary system depressurization serves both preventive and mitigative functions. As such, a mechanism is needed which confirms that if the preventive function of depressurization is successful, procedures are retained to continue the effort of cooling the core in-vessel. For this reason the OSSA approach identifies two severe accident management domains: OSSA Phase 1 and OSSA Phase 2.

The OSSA Phase 1 accident management domain is called “Extended Core Heat-Up Response” (ECHUR) and provides unique accident response guidance with an emphasis on core and vessel cooling in the event that safety injection or other means of cooling can be recovered. Implementation of the OSSA Phase 1 procedures is the responsibility of the control room staff.

The OSSA Phase 2 accident domain is called damaged core response (DCR). As the name implies, when plant indicators point towards serious damage of the core, a separate set of procedures, recommended by the TSC, are implemented to preserve containment integrity and limit offsite fission product releases. Figure 4-1 shows the interfaces between EOP and OSSA.

This entry condition approach has several advantages:

- It preserves the distinction between preventive and mitigative measures, and thereby supports separate use of EOP and SAMG.
- It provides an unambiguous criterion for the transition to OSSA, confirming that when the prescribed core conditions are met, prompt transition occurs.
- It uses a transition criterion in which core conditions at transition are independent of the accident scenario.

Extended Core Heat-Up Response

OSSA Phase 1 acknowledges the importance of early termination of core damage. As a consequence of success in this domain, the reactor vessel is preserved as a barrier for fission product releases. Accident management priority during this period shifts from preserving the core intact to terminating core damage before reactor vessel failure through the restoration of heat removal systems (e.g., safety injection, emergency feedwater). While late reflood can result in increased hydrogen generation, possible recriticality, and increased steam production, these hazards do not necessarily outweigh the benefits of retaining the degraded core inside the vessel. The criteria for this action is to supply coolant to the reactor vessel as soon as injection capability is available.

The main accident management activities during the OSSA Phase 1 domain are:

- Depressurization of the RCS by opening the PDS (if not already depressurized).
- Recovery of SI and/or secondary heat sink.
- Monitoring instrumentation for event progression and conditions exceeding qualification limits.

Damaged Core Response

OSSA Phase 2 acknowledges that significant core damage has potentially occurred and that the RPV integrity may fail or may have already failed. Reliance on the U.S. EPR™ design inherent ex-vessel features for core debris cooling and combustible gas control is emphasized. Actions are re-prioritized toward containment protection and the minimization of radiological releases. The choice of entry condition shall be consistent with changing from preventive to mitigative measures. Upon entry, the accident management responsibility is shared with the TSC and actions associated with the OSSA Phase 1 are abandoned.

The main accident management activities during this OSSA Phase 2 include:

- Prepare and/or confirm readiness of SAHRS passive cooling for actuation (open MOVs protecting the passive flooding lines, if necessary).
- Continue efforts to recover secondary heat sink to protect steam generator tubes.
- Recover or confirm IRWST cooling.
- Monitor instrumentation for event progression and conditions exceeding qualification limits.

4.1.2 *Diagnostic Tool*

The OSSA diagnostic tool is a graphical computer system used by the main control room and TSC staff. The development of the diagnostic tool considers plant-specific characteristics related from design information, design analyses and safety studies, and PRA. In particular, the objective is to identify potential challenges to the integrity of barriers to the release of fission products, and their mechanisms and associated phenomena. It is also important to identify potential means to monitor each challenge. Two examples include the coolant injection rate needed for the removal of core heat (e.g., decay heat, metal oxidation) and hydrogen production due to metal oxidation.

The diagnostic tool addresses a system of three broad safety function categories:

- Releases to the environment.

- Containment integrity.
- Heat removal.

During mitigative accident management, priority is given to minimizing and terminating releases. In addition, each safety function category may take one of four states (controlled and stable, controlled but not yet stabilized, potentially challenged, and challenged). This system of prioritizing challenges allows flexibility to adapt the response to the event as the accident progresses without the need to diagnose the specific cause and subsequent detailed progression (i.e., a fully symptom based approach).

The OSSA diagnostic interfaces with the plant's ERDS which is a direct near real-time electronic data link between the licensee's onsite computer system and the NRC Operations Center.

4.1.3 *Ergonomic Considerations in Guidance and Procedure Development*

The third major element of the package is the development of accident management guidance and procedures. Guidance is usually used to describe a less strict and prescriptive set of instructions. A guideline can be structured and consist of a sequence of steps and branch points. Procedures are comprised of a step-by-step list of required actions and responses. These are symptom-based, matrix format instructions that allow the responsible emergency response teams to identify and evaluate potential actions and formulate recommended strategies or system recovery priorities. The OSSA package also contains a generic communication tool which identifies key information that must be exchanged between the decision-makers and the emergency response team. The communication tool can be adapted for plant specific application.

4.2 *Considered Operating Modes*

The OSSA methodology has been developed to address all plant operating states, which include the following categories: at-power, shutdown, and refueling conditions. OSSA anticipates that upon entry the plant state can be further described as one of these categories. These categories are differentiated in a manner analogous to defining core damage end states (CDES). The distinction depends on plant operating mode, the status of the RCS (e.g., open or closed, RCS pressure), and instrumentation available. Characteristic scenarios are evaluated from PRA mission success analysis and OSSA support studies. Examining these different conditions will consider additional assumptions as necessary to address various severe accident challenges.

At-Power

During at-power the RCS is closed and pressurized. On transition to OSSA, the PDS valves are opened to depressurize the RCS (if not yet performed). If the RCS is already depressurized, or if the depressurization and injection of the safety injection system (SIS) accumulator or the LHSI do not establish enough time for recovery of other injections into the RCS, core degradation may continue.

Shutdown

In the first category, the RCS is closed (water full and core exit thermocouples available) and pressurized. This condition is similar to at-power conditions and the similar accident management strategies are followed (including RCS depressurization). If depressurization and injection of the accumulators or the LHSI do not establish enough time for recovery of other injections into the RCS, core degradation may continue.

In the second category, the RCS is closed, repressurizable (water full and core exit thermocouples available), and depressurized (SI and accumulators valve closed). Opening PDS valves is performed at a core outlet temperature of 1200°F (650°C). Depending on the status of the LHSI and the operating state of the RCS pressure, core degradation may continue.

In the third category, the RCS is open (water level at mid-loop level and core exit thermocouples unavailable) and depressurized (SI and accumulators valved closed). No RCS depressurization is required and core degradation may continue.

Refueling

Core damage is considered in the spent fuel pool if the fuel assemblies are uncovered during an extended period. If no alternative to bring water to the spent fuel pool is feasible, fuel element degradation may continue.

4.3 *Computer Codes*

The AREVA NP OSSA methodology for SAMG development recognizes that severe accident management issues are resolved using both deterministic and probabilistic methods supplemented by appropriate research and development. The principal deterministic analysis tool supporting the application of OSSA is the Modular Accident Analysis Program (MAAP), Version 4 (Reference [19]). MAAP4 is an integrated system code that combines, in one package, models for heat transfer, fluid flow, fission product release and transport, plant system operation and performance, and operator actions. MAAP4 was developed to address all phases of severe accident studies, including severe accident engineering, PRA, and accident management. Models for accident phenomena that can occur within the primary system, the containment, or auxiliary-type buildings are also included. For a specified reactor and containment system, MAAP4 calculates the progression of postulated accident sequence (including the deposition of the fission products) from a set of initiating events to either a safe, stable, or impaired containment condition (by over pressure or over temperature), and the possible release of fission products to the environment.

MAAP4.0.7 contains specific models for U.S. EPR™ design features. The U.S. EPR™ design has unique containment regions devoted to debris stabilization and long-term cooling should a severe accident lead to melting of the reactor core and RPV failure. Modifications performed to the MAAP4 code address the ways in which these specific containment features are represented in the MAAP4 framework. Section 6 of Reference [1] provides further information on MAAP 4.0.7.

PRA models have been developed and quantified using the RiskSpectrum® Professional software package (Reference [20]). This software supports the use of linked fault-tree methodology. Analysis cases are created for fault tree analysis, event tree sequence analysis, and consequence analysis. To create these analysis cases, the basic fault-tree models are specialized to the sequence of interest using house events, exchange events, and boundary-condition sets. When multiple sets of minimal cutsets are obtained, they can be merged to provide an integrated set of results for the PRA. A cutset editor allows for further refinement of the results. Several event trees can be linked, including Level 1 event trees with Level 2 containment event trees. A comprehensive set of importance factors can be generated with uncertainty.

4.4 *OSSA Development Activities*

Like other Generation-III nuclear power plant designs, provisions to cope with severe accidents are included in the U.S. EPR™ design. OSSA credits such plant-specific design elements that may influence the overall structure, principles, diagnostics, and strategies identified in a documented SAMG. Situations that would lead to large early releases such as containment bypass, strong reactivity accidents, high pressure core melt, or global hydrogen detonation are practically eliminated in this design. The OSSA methodology involves several tasks that consider the nature of the more likely damaged-plant states during severe accident sequences and credits unique features and capabilities that improve the plant's overall response to a severe accident.

OSSA development tasks address identifying damaged- and recovered-plant states and OSSA entry, monitoring safety function performance, defining candidate immediate actions and recovery strategies, as well as quantifying instrumentation setpoints for performing actions, functional specifications for computation aids, and V&V of recovery strategies. The final product is expected to provide resolution to the accident management issues presenting the greatest challenge to the plant.

Severe accident analysis complements the SAMG development by providing insights into the accident progression and sensitivities to uncertainties associated with phenomena, initiating event and plant state, and operator response. As described in Section 4.3, AREVA NP applies the MAAP4 code for deterministic severe accident simulation and the RiskSpectrum code for quantifying PRA models. AREVA NP uses a 3-phase approach to establish support studies: 1) preliminary analysis for assessing accident progression and identifying challenges to the various defense-in-depth objectives; 2) OSSA development analysis for identifying damaged- and recovered-plant states and OSSA entry criteria, defining computation aids, assessing plant behavior to operator action, and test recovery strategies; and 3) V&V analysis.

In contrast to analyses performed for design certification, which focus on the minimum requirements for the prevention and mitigation of severe accidents, the OSSA development analyses provide greater perspective of the effectiveness of candidate response strategies. Best-estimate assumptions are incorporated to consider not only the effectiveness of normal performance of systems, but to also assess the capability of systems of performing functions beyond those which they were designed.

The scope of OSSA tasks does not include the development of the required training program in NUREG-0737 (References [8] and [9]); however, the computer models developed while defining OSSA may be useful in establishing the severe accident component of a simulator. Figure 4-2 shows the OSSA structure, including accident progression roadmaps, immediate actions, challenge/system matrix, controlled area strategies, recovery matrix system guidelines, and the communication tool.

4.4.1 *Preliminary Analyses Supporting OSSA*

Design-basis and beyond design-basis analysis supporting the U.S. EPR™ design certification provide a valuable resource to OSSA development by identifying and characterizing the challenges to and functional objectives for protecting a plant's fission product barriers. These include the following key accident management objectives:

- Reliable instrumentation and equipment necessary for monitoring and mitigating accidents.
- Reactivity control that verifies that the reactor is sub-critical, thus reducing core heat generation.
- Coolant inventory control providing adequate coolant to the RCS and RPV so that heat may be removed from the fuel rods by the reactor coolant.
- Heat removal system providing heat transfer from the reactor coolant to the ultimate heat sink.

- Reliable primary system depressurization to support the inventory control function and to reduce or eliminate the consequences of HPME.
- Heat removal system providing heat transfer from the containment to prevent exceeding the containment design limits for both temperature and pressure.
- Combustible gas control capable of accommodating hydrogen generation equivalent to a 100 percent metal-water reaction of the fuel cladding and limiting containment hydrogen concentration in any compartment to no greater than 10 percent.
- Core debris coolability to protect the containment liner and other structural members from damaging effects of high temperature molten corium.

This task involves a review of the deterministic and probabilistic analysis and conclusions appearing in the U.S. EPR™ FSAR Tier 2, Chapter 6 (containment), Chapter 15 (safety analysis), and Chapter 19 (severe accidents). These U.S. EPR™ FSAR chapters and their supporting documents provide useful information that helps to show the relative importance of the principal fuel, RCS, and containment protection features. Attachment A includes this review for the U.S. EPR™ design, identifying the safety roles of the plant systems, structures, and components.

Key Deliverable(s): Report summarizing review of existing analyses

4.4.2 *Identification of Severe Accident Challenges*

A necessary step in accident management planning is to identify those vulnerabilities that are likely to cause challenges to the plant's safety functions and, hence, the fission product barriers preventing the release of radioactive materials. Vulnerabilities are assessed on the basis of an analysis of the plant's response to beyond-design-basis accidents. This is to be done in a realistic manner using best-estimate assumptions, taking note of the uncertainties associated with such methods.

This task involves using the output of the preliminary design and beyond-design-basis analysis to make selections of which severe accident challenges need to be explicitly treated with OSSA. Identifying these challenges should be complemented by identifying existing design measures and any related manual actions that cope with each one.

Potential recovery strategies and actions should be included to complement this activity. Attachment A includes this assessment for the U.S. EPR™ design for at-power and shutdown operating modes.

Key Deliverable(s): Report summarizing review of severe accident challenges.

4.4.3 *Severe Accident Simulation Model Development*

This task involves the reconciliation of SAMG objectives listed in Section 4.4.1 and related plant models prepared for the primary analysis tool (i.e., MAAP4). In particular, the plant models need to accurately reflect the plant design and incorporate features that emulate the full set of ERDS plant parameters.

Best-estimate assumptions should be incorporated to the extent quantifiable. This means that phenomenological conservatisms incorporated into preliminary analyses should be eliminated; however, uncertainties associated with potential performance degradation of mitigating features should be retained. Examples modeling assumptions that should be incorporated into the base U.S. EPR™ model for OSSA are:

- Prompt actuation of PDS valves upon entry into OSSA.
- Normal safety injection delivery from 3 of 4 accumulators.
- Nominal performance degradation for MHSI and LHSI pump flow.
- Auto-ignition of combustible gas release from MCCI.
- Nominal performance degradation for PAR recombination rate.
- Best-estimate coolant flow delivery to spreading room.

The plant model should include measures representing all of the required ERDS plant parameters. Intrinsic MAAP4 code output variables are available for most of these parameters; however, in some instances special functions may need to be developed to emulate an instrument, such as dose rates in various locations. Section 3.3 includes monitoring instruments in addition to the ERDS requirement that must also be incorporated into the base plant model.

Convenience functions are incorporated to aid in diagnosing and tracking the plant state. Such functions can include unique plant state setpoints useful for triggering operator action or principal severe accident event milestones (see Section 2.0), measures quantifying deviations from a mitigation path, functions related to NUREG-0737 Item II.B.3 – Post Accident Sampling Capability, and functions projecting a future plant state (e.g., core uncovery or instrument failure).

Calculations simulating a set of more likely or relevant scenarios per the methodology given in Reference [1] are performed to verify the expected performance of the model to demonstrate the analytical capability of the model and to define an analytical baseline for future calculations.

Key Deliverable(s): Updated MAAP4 U.S. EPR™ model and calculational results from the set of more likely scenarios.

4.4.4 *Baseline Uncertainty and Sensitivity Analysis*

To begin the process of characterizing the best-estimate progression of a severe accident, the domain of possible severe accident scenarios is examined considering a broad range of uncertainties. Any conservatism inherent in the U.S. EPR™ MAAP4 model is replaced with best-estimate modeling, including assumptions regarding auto ignition of hydrogen and induced pipe or tube rupture. The uncertainty analysis is a monte carlo approach of uncertainty sampling from a large set of parameters important to event progression.

The mechanics of the uncertainty analysis are similar to that described in Reference [1] and [22]. The principal MAAP4 parameter contributors associated with the phenomenological uncertainty developed in Reference [1] are shown in Table 4-1. Calculation of between 60 to 100 samples is adequate to evaluate meaningful performance trends and boundaries (i.e., coverage of event domain exceeding 95 percent with 95 percent confidence).

A sensitivity analysis, applying the results from the uncertainty analysis, is then performed to identify the more important parameters associated with the figures-of-merit used to quantify margin for particular severe accident challenges. Reference [23] describes a useful sensitivity (importance) analysis methodology that can be applied for this purpose. The sensitivity analysis should be conducted multiple times to capture various event phases.

Results from these analyses serve as the baseline for the validation effort provided as the last analytical step in the OSSA process.

4.4.5 *Characterization of Plant State Symptomatic Signatures*

Enabling an emergency responder to perform a logical series of actions leading to the eventual recovery from a severe accident requires plant design knowledge, an understanding of the performance limits of systems available to protect the fission product barriers, and a means to process a large amount of plant instrumentation and control signal data and translate that information into appropriate actions. An effective SAMG simplifies this effort by capturing this information in an ergonomic package. Among these SAMG prerequisites, the processing and distillation of plant signals presents a unique challenge. In meeting this objective, the SAMG developer must identify quantitative measures that lend themselves to meaningful instruction.

This task involves defining quantitative and qualitative measures (i.e., plant state symptomatic signatures) in the form of criteria and setpoints that capture the degree of various severe accident challenges and an appropriate recovered-plant state and considers the at-power and shutdown operating modes. In deriving the damaged-plant state symptomatic signatures, the SAMG developer evaluates the previous analyses involving the more likely severe accident scenarios for quantified ERDS parameter values at the principal severe accident event milestones, including uncertainty analysis. Qualitative descriptions of the damaged-plant state identify relevant damaged-plant characteristics (e.g., status of fission product barriers and location of corium). Information about the first-order trends (i.e., increasing/decreasing, higher/lower, hotter/colder) during the principal event phases outlined in Attachment C is to be included. Recovered-plant state symptomatic signatures are expected to be dependent on the degree of event progression and the status of the fission product barriers.

This task of characterizing plant/system/component state symptomatic signatures is captured in the development of safety function monitoring diagrams qualitatively highlighting plant/system/component transitions through status changes in the safety functions (heat removal, containment integrity, and fission product release). These transitions are evaluated through examination of both previous analysis and the uncertainty and sensitivity analyses, emphasizing the severe accident challenges.

Industry precedence in addressing NUREG-0737 Item II.B.3 (Post Accident Sampling Capability) provides additional insight into how this information may be compiled and presented. For example, from Westinghouse's WCAP-14696-A Core Damage Assessment Guidelines (Reference [21]) a list of core damage symptoms for the U.S. EPR™ design can be derived as shown in Table 4-2 and Table 4-3. In compiling this information, supports studies are performed to identify instrumentation setpoints associated with a particular event phase.

4.4.6 *Entrance/Exit Criteria Assessment*

As previously described, the OSSA material are split into two parts: OSSA Phase 1 and OSSA Phase 2 dealing with different priorities:

- OSSA Phase 1 puts a high priority on trying to depressurize the primary system and recover safety injection to preserve the core in-vessel.
- OSSA Phase 2 enhances the containment protection actions.

While the ultimate primary depressurization criterion is set at a T_{COT} value of 1200°F (650°C), this determination involves performing MAAP4 analyses to characterize a second T_{COT} based curve correlating to constant cladding temperature (T_{CLAD}) representative of an advanced core degradation status. Measures equating to the maximum clad temperature 2200°F (1477°C) are considered a good indication of severe accident progression and are commonly used as SAMG entry criteria.

The primary system pressure and core outlet temperature are measures available through instrumentation, and a good correlation exists between them and the maximum clad temperature. Support studies in this category define this correlation using several calculations of varying reactor pressure. As is the common practice, the values of reactor system pressure and core outlet temperature are extracted when the predicted maximum clad temperature reaches 2200°F. These data points are then fit into a curve similar to Figure 4-3.

Complicating this objective is that instrumentation may only be rated to a given temperature for a set time. The early OSSA phase recognizes that the keystone operator action of primary system depressurization may provide alternative recovery strategies; therefore, plant control remains with the acting main control room staff. An alternate entry into the OSSA Phase 2, and the involvement of the TSC, must also consider at-power and shutdown operating modes and the instrumentation available and their survivability characteristics. Figure 4-4 shows the entry into the OSSA domains as a function of both RCS pressure and core outlet temperature. OSSA Phase 1 is entered from EOP when the core outlet temperature is greater than 1200°F (650°C).

4.4.7 *Specification of Diagnostic Tool Elements*

One of the key aspects defining the structure of the severe accident management approach is the means used to monitor and assess plant conditions and identify potential actions for evaluation. OSSA diagnostic describes a tool aimed at easing the event evaluation during an emergency. In particular, when the required parameters cannot be directly evaluated or may require a complicated evaluation (e.g., evaluation of decay heat as a function of time, minimum water injection flow rate needed to retain the core in vessel as a function of primary pressure, the containment leak rate).

In developing severe accident management guidance, the choice of diagnostic scheme is fundamental in determining the overall structure of the guidance package. Existing approaches worldwide are symptom based. Symptom-based guidance simply means that everything in the guidance is based on directly measurable plant parameters. For example, T_{COT} is used as an indicator of core cooling status, rather than the cladding temperature.

Essential plant parameters related to SAMG-derived procedures (e.g., those required in the ERDS) are included in the OSSA diagnostic. Such measures allow the responsible emergency response team to determine whether these parameters may lead to a possible challenge to the integrity of the containment.

In OSSA, severe accident mitigation is based on following three goals:

- To avoid/limit radioactive releases.
- To verify containment integrity (i.e., from pressure, temperature, H_2 concentration, and bypass challenges).
- To verify heat removal from the core debris and the containment.

To fulfill these goals, OSSA diagnostic provides continuous monitoring of plant conditions, considering the operating modes (i.e., at-power and shutdown), from entry to plant recovery. The status of three safety functions, called “releases”, “containment”, and “heat removal” are monitored in the main control room and the TSC.

During a severe accident, the status of each safety function may change depending on the situation. Four different conditions are defined in OSSA as shown in Figure 4-5.

The color code is used in the safety function monitoring diagrams. The strategies and actions, as well as the systems used for their implementation, depend on the status of the safety functions. The severe accident situation is considered as controlled (or ‘in controlled area’) when the three safety functions have green or yellow status.

This task involves the specification of diagnostic functional requirements. It should be contained within an easily-used framework (e.g., a looping flowchart, continuously monitored safety function status indicators), which allows different sets of actions to be considered, and prioritizes the evaluation process so that the different action sets are evaluated in an appropriate order (i.e., most important first). The method consists of a high-level monitoring scheme which allows “change of direction” if inappropriate actions were taken and negative impacts of actions become unacceptable or a misdiagnosis occurs. The scheme also includes a check for “success” (i.e., severe accident management guidance exit conditions). In compiling this information, previous analyses and new support studies are used to identify instrumentation setpoints associated with transitions between condition severity levels.

Stealthy scenarios are considered. Stealthy scenarios are non-severe accidents that briefly exhibit damaged-plant symptom signatures indicating a severe accident. Transition to OSSA is avoided in such cases to prevent escalating the event into a severe accident.

Key Deliverable(s): Internal report on the specification of the OSSA diagnostic and related computer aids.

4.4.8 *Instrumentation Survivability and Setpoints Analysis*

Instrumentation is a key element of severe accident management. It is required for assessing plant conditions, evaluating applicable mitigative strategies, implementing selected mitigative strategies, and assessing long-term evolution of the situation. A list of instrumentation, their primary and possible alternative roles in tracking severe accident challenges, and corresponding setpoints will be documented during the OSSA development process for the U.S. EPR™ design, beginning with the instruments identified in Section 3.3. The capabilities of each instrument are characterized through analyses with first-order thermal-hydraulic computer codes such as RELAP5 (Reference 24), FATHOM (Reference 25), and GASFLOW (Reference 26). Setpoints associated with each instrument are based on the specification of the OSSA diagnostic.

Not all instrumentation can be qualified for severe accidents, therefore a criteria is used to decide which instrumentation will be assessed for survivability to severe accident conditions during the OSSA development process. This is done by first deciding which safety functions require dedicated severe accident I&C. Only the safety functions that are absolutely essential to manage the severe accident are considered.

When the safety functions are identified, a complete analysis of each determines which instrumentation are essential to manage the severe accident. The OSSA documentation will clearly define survivability requirements that distinguish severe accident I&C and non-severe accident I&C and their role in different operating modes (i.e., at-power and shutdown). This process considers the plant conditions for which the instrumentation is expected to withstand, plus any other applicable standards that are determined during the development of the instrumentation list.

Support studies are used to evaluate the potential operating conditions to which the instrumentation could be subjected. The chosen scenarios consider the location, physical conditions, and the periods of time that the instrumentation have to endure. Different instrumentation is required during the progression of a severe accident. The instrumentation is also identified by its role in severe accident response. These categories include immediate actions, instrumentation required for diagnostic, monitoring safety functions, monitoring exit condition, instrumentation required for mitigative strategies assessment, and instrumentation required for long-term strategies.

A complete summary of the required instrumentation for each severe accident management function includes the survivability needs of the instrumentation, setpoints datasheet (for all operating states), dependency on the 12 hr UPS batteries, and availability of the information within the control room.

Key Deliverable(s): Internal report on instrumentation survivability and specification

4.4.9 *Define Candidate Immediate Actions and Recovery Strategies*

Accident management considers those actions taken during the course of an accident by the responsible emergency response team (i.e., the plant operators, technical support and plant management staff) in order to:

- Prevent the accident from progressing to core damage.
- Terminate core damage progression when it begins.
- Maintain the integrity of the containment as long as possible.
- Minimize onsite and offsite releases and their effects.

The latter three actions constitute the subset of severe accident management.

This task involves identifying detailed strategies addressing the spectrum of initial plant states (i.e., at-power and shutdown) and verifying the proper management of the principal safety functions through any severe accident. The list of detailed strategies cover all potential challenges defined in Section 4.4.2, to verify that severe accident management is capable of dealing with any situation regardless of its probability of occurrence. System guidelines are developed to address SAMG strategies (see Section 5.0). The guides build upon previous experience gained by industry, considering the insights from GL-88-20, NEI-91-04 and EPRI 101869 (see Section 1.2.4). In particular, they diagnose plant conditions, prioritize response, assess equipment availability, identify and assess negative impacts, and after a strategy is implemented, determine whether implemented actions take effect.

Results from the previous tasks are carried forward into this task as criteria applied to describe recovery strategies devised to mitigate the damaged-plant condition and prevent or delay each of the stages of progressing accident severity. Support studies, such as those presented in Table 4-4 (see Attachment B for additional detail), are assessed for impacts to the principal plant safety functions. The timing of recovered systems should also be considered in the suite of support studies. Release estimates derived by those support studies are considered with emergency plan information to determine the maximum releases for OSSA scenarios, and then relate to the emergency levels. The impact of both dedicated severe accident response features, such as the SAHRS, and configurations of non-dedicated systems are assessed within this context.

As part of the development of the OSSA diagnostic, two categories of challenges are considered: immediate and mitigation path.

Immediate challenges are challenges that might occur before the TSC, as part of the emergency crisis team, is available. In this case the shift team takes actions without the help from TSC, and therefore with no possible independent evaluation. Immediate actions are implemented for all fast severe accident scenarios, and deal with potential challenges to the containment to which such scenarios can lead.

Severe accident conditions may deviate from a desired mitigation path. These challenges can be addressed through an evaluation process performed by the emergency organization using the OSSA diagnostic safety function monitoring. For each challenge, indicated by a change in the safety function status, mitigative high-level strategies can be derived. For each of these high-level strategies, the potential systems that can be used for mitigation are listed.

Severe accident specific features are implemented in the U.S. EPR™ design in order to maintain the plant on an effective mitigation path with no challenges to the confinement of radioactive substances. When these specific features operate as foreseen in the plant design, the severe accident conditions are considered to be on the mitigation path. The mitigation path is a subset of a controlled area. The controlled area corresponds with plant conditions where no future or existing challenge to the containment exists. Only few actions are required in this area.

This task results in several end-user products based on insight from previous tasks to identify emergency response priority actions to return the plant to a mitigation path, confirming long-term stabilization of safety functions.

Key Deliverable(s): Report on Candidate Immediate Actions and Recovery Strategies, includes List of Immediate Actions; Load Lineup Charts; Safety Function Recovery Plans (Severe Accident Control Room Response Guides); Challenge System Matrix/System Guides.

4.4.10 *Fuel and Safeguard Buildings Analysis*

The Reactor Building is not the only location where a severe accident can occur. Application of OSSA also considers severe accidents in the Fuel Building and Safeguard Buildings. The probability may be small, but fuel damage in the spent fuel pool is a possibility and OSSA-based guidelines are needed to prevent the release of fission products. Because the Fuel Building is not designed for reliable radioactivity confinement in case of spent fuel damage, it is recommended that an earlier entry criteria compared to an in-containment type severe accident is considered. The main strategies for the Fuel Building are heat removal in the spent fuel pool and limiting radioactive releases.

This task requires analyses and end-products similar to the previous tasks described; however, the scope of the Fuel Building and Safeguard Buildings analysis focus on a much smaller set of heat removal and radioactive release challenges.

Key Deliverable(s): Fuel Building and Safeguard Buildings analysis report.

4.4.11 *OSSA Verification and Validation Analysis*

V&V analyses can address several areas of OSSA development. Most fundamental to the development of OSSA is the confirmation of various accident management assumptions incorporated into the developmental support studies. These V&V support studies are prepared to help the appropriate decision makers understand the effects of various potential actions considered during OSSA development.

4.4.11.1 *Verification Tasks*

Verifying OSSA-based guidance establishes the consistency, completeness, and correctness of the supporting documentation performed in a thorough peer review. Specifically, verification demonstrates that mission requirements have been correctly translated into guidance and procedural requirements. It addresses the OSSA mitigation strategy in its generic application environment. The task of verification should address the following types of errors:

- Logic Errors – failure to accurately reflect a mission requirement.
- Documentation Errors – failure to accurately define a mission requirement.
- Overload Errors – conflicts arising from consideration of the full suite of available data.
- Timing Errors – Logic Errors that are a function of timing conditions or coincidental combinations of events.
- Throughput and Capacity Errors – guidance or procedural requirements that fail to consider the logistics of implementation.
- Fallback and Recovery Errors – failure to clearly define guidance or procedural requirements.

This activity involves a through review of the underlying documentation as described in Section 6.0.

The preferred way to address V&V activity is through training using a full-scope simulator facility. Real-time simulation of plant system response to a severe accident is challenging for most simulators. A SAMG validation program consists of a combination of simulator (for testing the EOP–OSSA transitions and the early phase of the accident) and table top exercises (to test TSC usage and long-term recovery). Tabletop exercises require some severe accident analysis prior to validation to serve as a basis for simulated plant response. The amount and scope of such analysis is defined when the detailed approach to validation is finalized.

4.4.11.2 *Validation Tasks*

Validation is addressed by considering uncertainties in event progression and recovery strategies. This is accomplished by using analytical calculations that consider plant-specific candidate high-level actions. Recovery actions applicable to the U.S. EPR™ design to consider include:

1. Coolant injection into RPV or RCS (i.e., feed and bleed strategies).
2. Depressurize (i.e., vent) the RPV/RCS.
3. Restart reactor coolant pumps.
4. Depressurize (i.e., vent) or flood steam generators.
5. Spray or flood containment.
6. Vent containment.

Analyses should incorporate uncertainties associated with the implementation of recovery actions (e.g., operator response time) and with possible candidate lower level actions. These include sensitivity to possible adverse effects that may occur as a consequence of taking mitigating measures, such as pressure spikes, hydrogen generation, return to criticality, steam explosions, thermal shock or hydrogen deflagration or detonation. Such analysis is conducted with the original code used to develop the OSSA (i.e., MAAP4) or with an independent code that retains applicability and is approved by the NRC for severe accident analyses. These analyses are conducted on a best-estimate basis and should test the symptom-based instruction considering the uncertainty associated with initiating event and event progression.

A validation strategy similar to the uncertainty analysis described in Section 4.4.4 and Reference [22] is applied. Gains in margin resulting from an effective accident management plan appear as tighter uncertainty bands and gains in safety function margins.

Supplemental sensitivity analyses complement the initial set of studies to resolve cliff-edge or cusp behavior such as that resulting from the possible degradation of equipment performance. Cliff-edge behavior may also arise from consideration of alternative mitigation strategies for a common plant symptom and containment challenge. Analysis can resolve preferred strategies in these situations. Consistent with the U.S. EPR™ design philosophy on severe accident response, the preferred strategy in such circumstances is one that minimizes the overall uncertainty to successful mitigation of a severe accident, in particular, minimizing radiological releases. Supplemental analyses should also include a small set of cases examining damaged-plant states resulting from low frequency, high-consequence scenarios.

4.4.12 *Guideline Development*

This task involves developing the documentation described in Section 6.0. The information developed in the previous tasks is to be compiled into a complete end-user OSSA product.

4.5 *Task Implementation*

OSSA activities are categorized into six elements:

A1 – Severe Accident Challenges:

- Preliminary Analyses Supporting SAMG (Section 4.4.1).
- Identification of Severe Accident Challenges (Section 4.4.2).

A4 – Basis and Principles:

- Severe Accident Simulation Model Development (Section 4.4.3).
- Baseline Uncertainty and Sensitivity Analysis (Section 4.4.4).
- Characterization of Plant State Symptomatic Signatures (Section 4.4.5).

A2 – Support Studies:

- Entrance/Exit Criteria Assessment (Section 4.4.6).
- Specification of Diagnostic Tool Elements (Section 4.4.7).

A5 – Main Operating Strategies:

- Define Candidate Immediate Actions and Recovery Strategies (Section 4.4.8).
- Fuel and Safeguard Buildings Analysis (Section 4.4.9).

A3 – Instrumentation and Setpoints:

- Instrumentation Survivability and Setpoints Analysis (Section 4.4.10).

A6 – V&V and Technical Background Reports:

- SAMG Verification and Validation Analysis (Section 4.4.11).
- Guideline Development (Section 4.4.12).

Table 4-1—Select MAAP4 Uncertainty Parameters

Parameter	Description
FAOX	Zr-H2O Oxidation Multiplier
FZORUP	Fraction of Zr oxidized to keep cladding intact
TCLMAX	Cladding Melt Breakout Temperature
LMCOL	Fuel Rod Collapse Temperature
IEUTEC	Enable/disable the U-Zr-O eutectic model
TEU	Fuel Melt Temperature
TEUBS	Control Rod Melt Temperature
EPSCUT/EPSCU2	Melt relocation HTC
XDJETO	Particulate debris size in lower plenum
EPSPB	Porosity of fuel debris beds
TJBRN	Local auto-ignition temperature
TAUTO	Global Auto-ignition temperature
QCR0	Total Power (decay power)
XROF0	Initial radius of the local vessel failure
FDAMLH	Lower head damage fraction for failure
FRCOEF*	Corium friction coefficient
FCHF* (max)	Flat Plate CHF Kutateladze #
FCHF* (steaming rate)	Spreading room steaming rate (kg/s)
EWL, EEQ, ECM	Reactor pit emissivities
FEFPAR	PAR capacity scale factor
NFH2MN	PAR Threshold for operation

Note:

Key Deliverable(s): Uncertainty analysis and sensitivity (importance) analysis.

Table 4-2—Symptoms of Core Damage

Indication	Damaged-Core States		
	OSSA Phase 1 (Cladding Failure w/ Possible Melting)	**OSSA Phase 2 (Damaged Core)**	**OSSA Phase 2 (Ex-Vessel Core Melt)**
Core Exit Thermocouples	Core outlet temperature **>** 1200°F	Core outlet temperature **>** [...]°F	N/A
Reactor Vessel Water Level	Collapsed water height at or below core mid-plane	Collapsed water height at or below [...]% core height for more than [...] minutes	N/A
Core Nuclear Instrumentation	External core power Monitors increasing	External core power monitors increasing	N/A
Hot Leg Thermocouples	Considerable superheat (>[...]°F above Tsat)	Core outlet temperature if available	N/A
Containment Hydrogen Inventory	Some or increasing hydrogen measured in containment	Increasing hydrogen measured in containment	Substantial hydrogen measured in containment (equivalent to **>** [...]% Zr-water reaction)
Containment Radiation Monitor	Limited radiation in containment perhaps due to reactor coolant activity, spiking and release of fuel rod gap activity	High radiation in containment	Rapid increase in radiation levels in containment

Table 4-3—Summary of Core Damage Indicators

Core Outlet Temperature < [...]°F	No Fuel Rod Cladding Damage
Core Outlet Temperature < 1200°F	Fuel Rod Cladding Damage Not Likely
Core Outlet Temperature > 1200°F	Fuel Rod Cladding Damage Likely and RCS Pressure < [...] psig
Core Outlet Temperature > [...]°F	Fuel Rod Cladding Damage Likely and RCS Pressure > [...] psig
Core Outlet Temperature > [...]°F	Certain Fuel Rod Cladding Damage; Core Overtemperature Damage Likely
RPV Level > Upper Core support Plate	No Fuel Rod Cladding Damage
RPV Level < Core Mid-Plane	Fuel Rod Cladding Damage Likely
Hot Leg RTDs < T_{sat}	No Fuel Rod Cladding Damage
Hot Leg RTDs > [...]°F	Core Overtemperature Damage Likely
Containment H_2 Monitor < 1%	No Core Overtemperature Damage
Containment H_2 Monitor > 1%	Possible Fuel Rod Cladding Damage; Core Overtemperature Damage Likely
Containment H_2 Monitor > [...]%	Widespread Core Overtemperature; Likely Damage
Containment Rad Monitor > RCS Plus Pre-Existing Iodine Spike	Fuel Rod Cladding Damage Likely
Containment Rad Monitor > [...]%	Fuel Rod Cladding Damage (Gap Release); Possible Core Overtemperature Damage
Containment Rad Monitor > [...]%	Core Overtemperature Damage (Core Release) Likely

Note:

Key Deliverables: Internal report on safety functions monitoring diagrams (for at-power and shutdown operating modes) and an updated MAAP4 U.S. EPR™ model, as necessary.

Table 4-4—Support Studies to Consider

In-Vessel
Chemical and volume control system charging system recovery
Steam generator heat removal
Filling faulted steam generator
Restart reactor coolant pumps
Time-at-Temperature (TCOT > 1200°F) and late reflood
Delayed RCS depressurization
Ex-Vessel
Early activation of containment spray
Late reflood
Early/delayed active cooling
Recovery of LHSI cooling chain
Sump clogging and backflush
Containment depressurization/venting

Figure 4-1—EOP and OSSA Domain Map

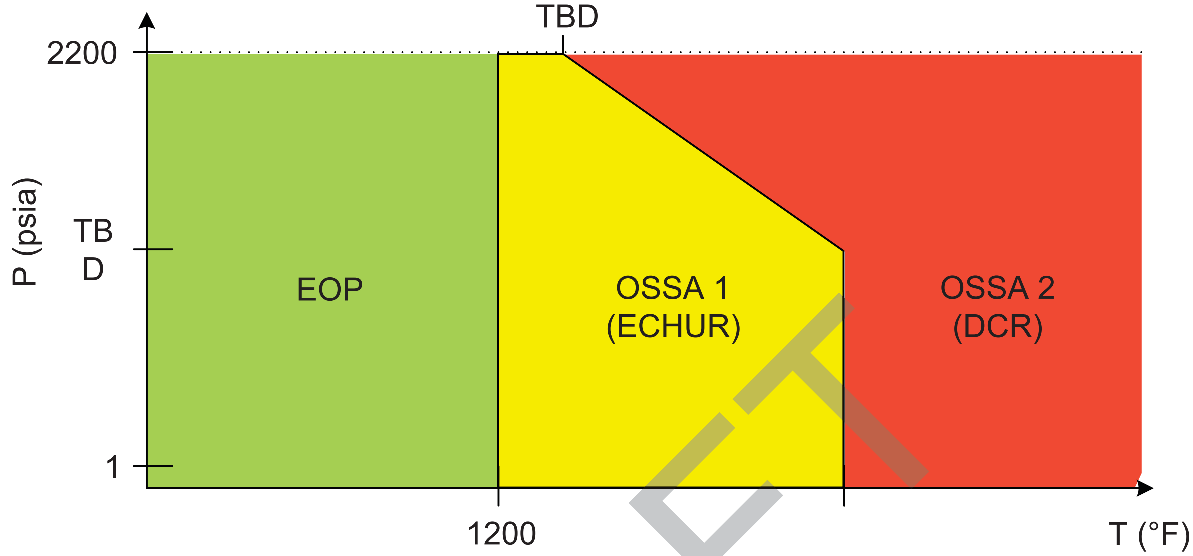


Figure 4-2—OSSA Structure

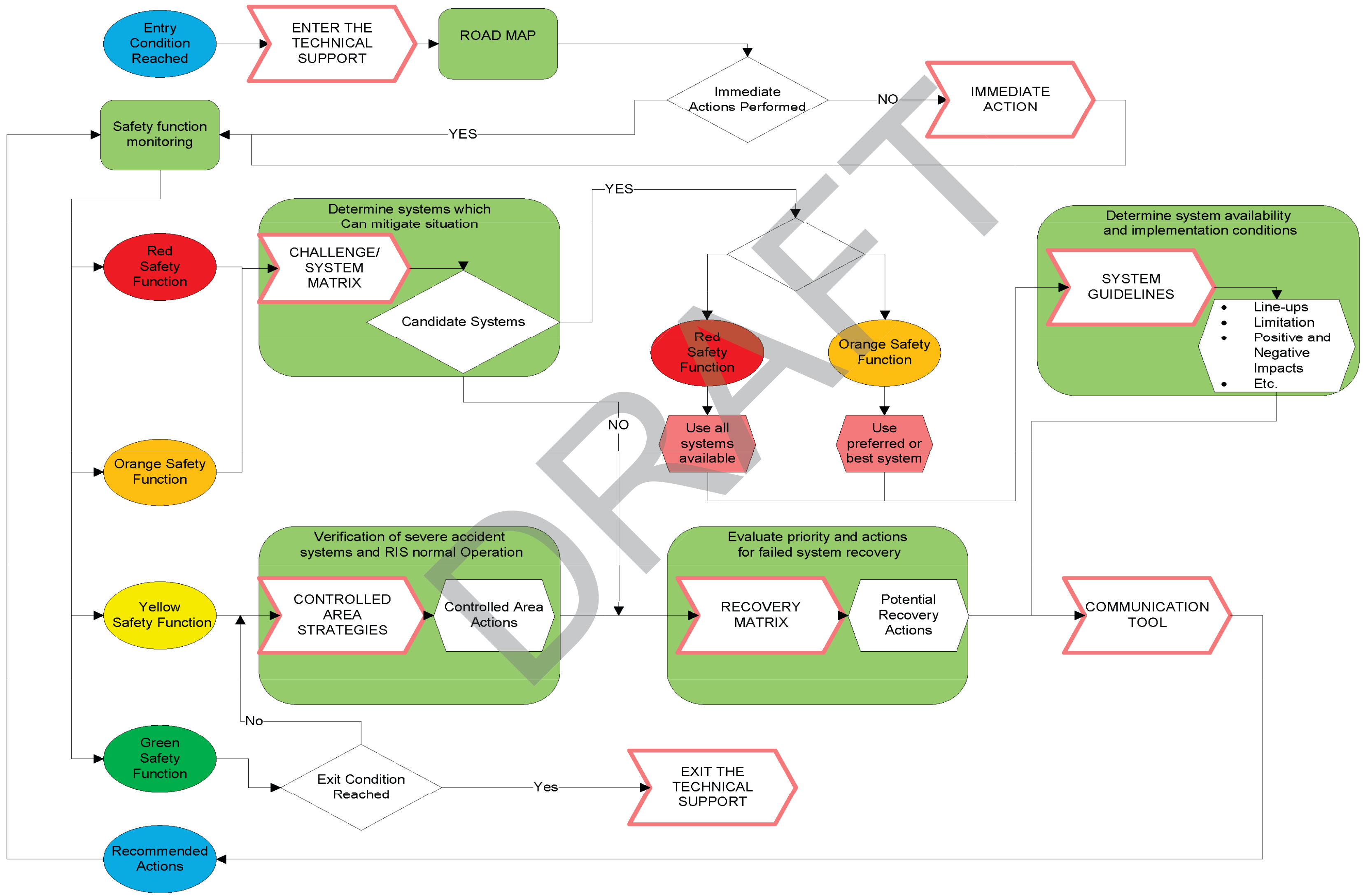


Figure 4-3—One Common Entry to OSSA from EOP with COT Available

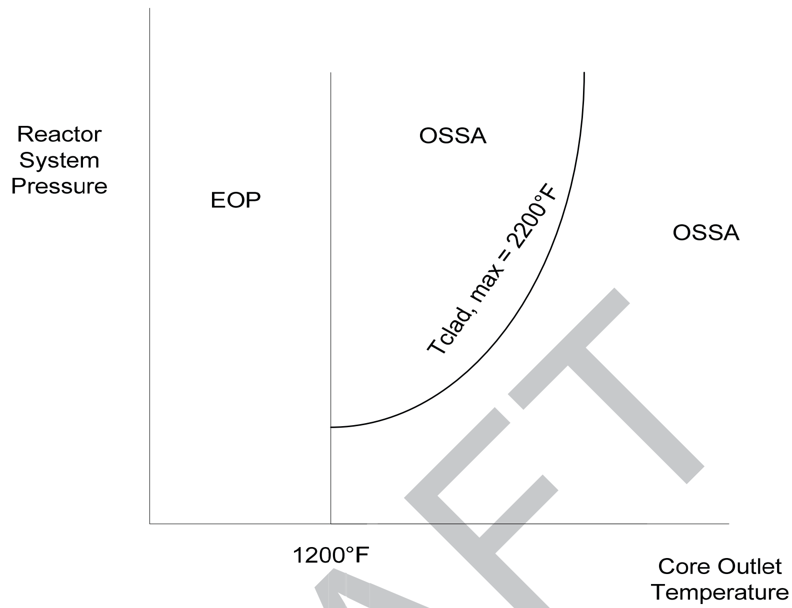
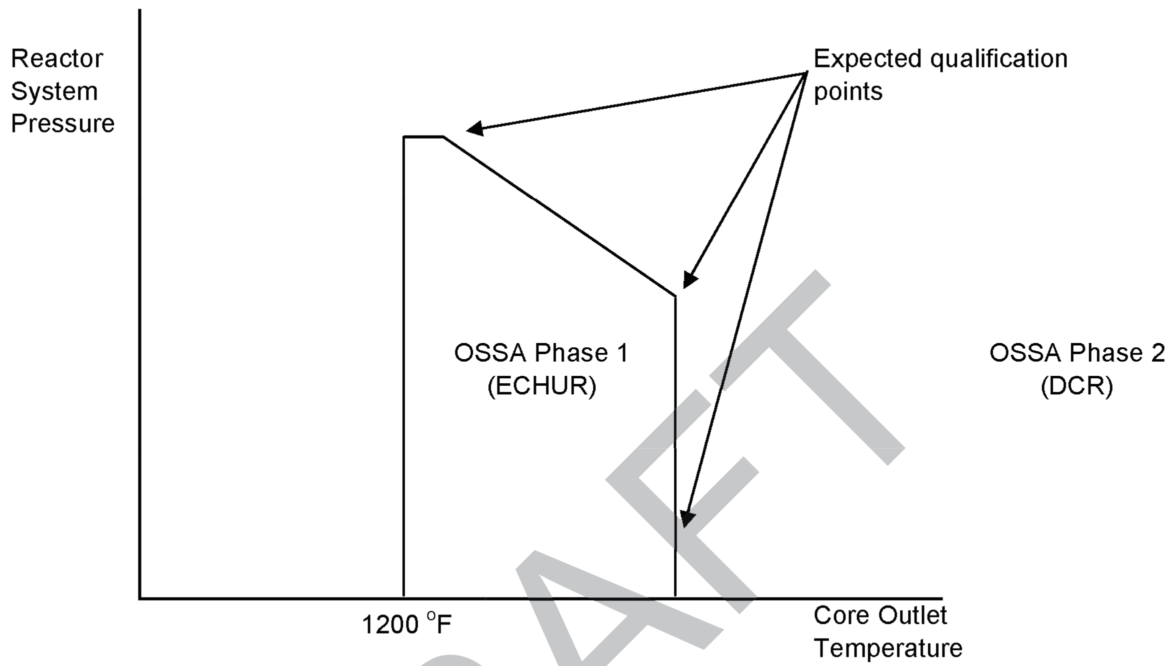


Figure 4-4—Entry to OSSA Phase 2 from OSSA Phase 1



Note:

Key Deliverable(s): Internal report on specifying entry/exit criteria.

Figure 4-5—Safety Function Color Code

Green	Controlled stable condition
Yellow	Controlled but not yet stable condition
Orange	Uncontrolled situation with a potential future challenge
Red	Severe challenge to the safety function

5.0 CANDIDATE SYSTEM-LEVEL ACTIONS

Accident management evaluation with the OSSA methodology involves assessing numerous candidate system-level actions. System guides are developed incorporating the candidate action prioritized by their relationship to the principal safety functions. They are linked to the safety function challenge status through the challenge–system matrix.

These system guides serve the TSC and operators in the U.S. EPR™ design as the basis for their evaluation of candidate system-level actions when the status of the plant and any challenges are known. They provide the elements needed for decision-making on whether or not the actions should be performed, including guidance to help evaluate potential negative impacts or to determine any applicable limitations for the strategy under consideration. In order to link the system to be evaluated for each plant status, the development of a challenge/system matrix relates each safety function and safety function status to the system that can be used.

The system guides are expected to capture the TSC decision process that relies on balancing different system configurations. This process depends on the availability of equipment and the potential risks and benefits. For each system listed in the challenge/system matrix, a dedicated system guideline is developed. The system configurations are part of the system guidelines. The purpose of the system guidelines is to detail the general objective of the configuration, conditions of use (ambient, system, flow rate…), and possible risks. Furthermore, system flow diagrams are provided on which the configuration piping lines are highlighted.

The system guide information is separated into two parts:

- A general system description including:
  - Presentation of the system.
  - Normal and emergency operation of the system.
  - Related support systems.

- Pump curves (if applicable).
- List of potential severe accident system configurations.
- Simplified system description.

- System configuration:
  - Configuration objective.
  - Operating configuration conditions.
  - Limiting ambient conditions.
  - Summary system matrix (water source, pumps and valves requirement).
  - Associated line-up.

These worksheets include essential information to analyze the availability of a system configuration and assess their positive and negative impact in a given severe accident situation.

Table 5-1 identifies a selection of candidate actions specifically relevant to the U.S. EPR™ design based on a review of the EPRI Severe Accident Management Guidance Technical Basis Report (see Section 1.2.4).

5.1 *Inject into RPV/RCS*

Injecting water into the RCS is the most direct approach to mitigate the progression of a severe accident. By applying this action, stored energy, decay heat, energy generated from metal-water reaction can often be effectively removed, regardless of the event phase. The underlying cause of severe accidents is the inability to remove energy generated by the core, and this may not be achieved until SIS can be recovered.

5.2 *Depressurize the RCS*

RCS depressurization using the PDS occurs prior to other actions are performed for recovering from a severe accident. Many benefits can be realized by depressurizing the RCS during a severe accident. If accumulators and safety injection are available, the depressurization of the RCS should lower pressure below the setpoints for which these engineered safety features are designed to deliver emergency coolant. If normal sources of emergency coolant are available, then it may be possible to recover control of the core state without significant fuel failure, melting, and RPV failure.

If the ECCS is unavailable and severe accident progression continues unabated without PDS actuation, HPME becomes a concern. This is a phenomenon that may occur if the RCS pressure is elevated at the time of vessel failure. During HPME, the momentum of the core debris along with the driving force of high velocity gases released from the vessel transports molten core debris away from the reactor cavity leading to DCH. This threat from HPME is considered eliminated if the RCS is depressurized below 147 psia (10 bar).

There are also other positive effects of depressizing the RCS that are unique to core damage scenarios. Creep rupture is a plastic deformation process that occurs under high temperatures and sustained loads. The possibility of creep rupture of the steam generator tubes and the RCS pipes can be reduced or eliminated if the RCS pressure is lowered. Decreasing the RCS pressure can also help isolate the containment and reduce fission product releases for containment bypass sequences. If there are ruptures or leaks in the steam generator tubes, the reduction of the RCS pressure reduces the driving force on the fission products, and helps maintain them within the primary system. If injection of water occurs due to the reduction in RCS pressure, the water inventory helps to scrub fission products.

5.3 *Restart the Reactor Coolant Pumps*

To reduce the rate of coolant inventory loss, the TMI-2 experience led to a trip requirement for RCPs during a LOCA. Under other conditions, use of RCPs is permissible. Restarting RCPs provides forced flow of any cooling residing in the RCS through the RPV and core region. With wet steam generators, heat rejection is possible. Even under dry conditions, single-phase steam cooling may be beneficial to maintain the core in a coolable geometry.

5.4 *Depressurize Steam Generators*

Depressurizing the steam generators may be the first step to enable injection of water into the steam generators (SG), to establish a heat transfer path from the RCS to the

SGs, or to depressurize the RCS. The purpose of this action may be the depressurization of the RCS, or the establishment of a long-term decay heat removal pathway.

The principal negative impacts from depressurizing the SGs are related to the potential for creating a release pathway. Not only might the steam generator inventory be lessened, but any fission products within the steam generators may be released to the environment. Furthermore, if there is a steam generator tube rupture, the lower steam generator pressure will increase the driving force of fission products from the primary to the secondary side (assuming no actuation of the PDS). Even if no steam generator tube ruptures currently exist, the lowering of the SG pressure could increase the differential pressure across the steam generator tubes, inducing a rupture or increasing leakage of fission products from the RCS through leaking steam generator tubes.

The two principal methods of depressurizing the steam generators are through the main steam relief train (MSRT) which discharges directly to environment or through then turbine bypass valve which discharges to the main condenser. Programmed steam generator cooldown strategies using the MSRT (i.e., partial and fast cooldown) may be available; however, if fission product release is possible, steam dump to the main condenser may be preferred.

5.5 *Inject into the Steam Generators*

SGs are designed to provide a heat sink for the RCS during both normal and accident conditions. As such, preserving or recovering fluid delivery to the SGs is a key accident management objective.

Because much of the secondary side is located outside of containment, the SG tubes act as a containment boundary. As such, the prevention of induced steam generator tube ruptures is important for severe accident management. One of the methods of doing this is to inject water into the steam generator to keep the tubes cooled. This helps to protect them from rupturing due to heatup from hot gases on the primary side of the tubes. Nevertheless, if a tube rupture does occur, covering the break with water will scrub fission products from the primary system following core damage.

There are also several drawbacks associated with injecting water into the steam generators. These drawbacks have the potential to negatively impact the accident progression by allowing the direct release of fission products to the environment. The first concern is the thermal shock of the steam generators. If the steam generators have dried out during a severe accident, the tube temperatures may exceed 1000°F. The injection of cold water into hot, dry steam generators can place significant thermal stresses on the tubes, tube sheet, and other components. These thermal stresses can result in the failure of either the shell side of the steam generator or the steam generator tubes. Failure of the shell side of a steam generator during a severe accident reduces the amount of water that can enter the steam generator and increases flooding of the containment. Also, failure of the shell side of the steam generator results in a direct fission product release path to the environment if the steam generator relief/safety valves are not closed.

5.6 *Spray into Containment*

The U.S. EPR™ plant design includes a non-safety-related containment spray system that has been incorporated as one of the SAHRS operating modes. Operation of the sprays is not planned until the severe accident has progressed ex-vessel and corium has relocated into the spreading room. After the spreading, gravity-driven water flow begins from the IRWST to the spreading area until level equilibrium is reached between the two areas. The core melt is wetted with IRWST water without any operator action. During this passive cooling, the core heat is transferred to the containment atmosphere by steaming and evaporation. It is then transferred to the IRWST using SAHRS sprays and finally removed via SAHRS heat exchangers. The management of SAHRS spray lines in the OSSA is performed based on the containment conditions (pressure and hydrogen concentration), via the monitoring of the containment integrity safety function status.

After early event phase accident management recovery efforts have been considered and implemented to the satisfaction of the emergency response team, the SAHRS is switched from spray to active flooding mode for long-term cooling. A return to spray may be necessary in some situations to address fission product scrubbing or other departure from the recovered state condition. This switch of SAHRS back to spraying mode is managed by monitoring the containment integrity safety function status.

5.7 *Injection into Containment*

After passive cooling of the spread core melt is established, the SAHRS may be switched to the active cooling mode for long-term melt stabilization. When the active cooling mode is activated, heat is transferred directly to the IRWST and then removed by the SAHRS heat exchanger via suction lines. If LHSI cooling chain can be successfully recovered, it may also be used for containment heat removal.

The objective of switching to the SAHRS active cooling mode is to eliminate steam and evaporation from the spreading room. This is a consequence of developing a subcooled pool above the spread melt and maintaining heat removal from the IRWST. Active cooling leads to simultaneous increase of water levels inside the spreading area, reactor pit, and RPV. This enables the cooling of the remaining core debris inside the reactor pit and the RPV, while eliminating the spreading room steaming and associated consequences.

5.8 *Effect Recombiners*

Hydrogen generation can occur in a U.S. EPR™ plant during a severe accident due to oxidation on fuel rod surfaces, MCCI, and oxidation of the core support material. The largest contributor to hydrogen generation is oxidation of the fuel rod surface (i.e., metal-water reaction), which can vary depending on the timing of the melt progression. The CGCS is designed to promote atmospheric mixing in the containment and provide reduction in the hydrogen concentration during a severe accident.

Hydrogen reduction in the U.S. EPR™ design results from operation of 47 PARs which are used to reduce the hydrogen concentration in the containment atmosphere during a severe accident to minimize the risk of hydrogen detonation. PARs are arranged mainly inside the equipment rooms to support global convection within containment and, thus, homogenize the atmosphere as well as reduce local peak hydrogen concentrations. PARs are also included in the dome to cope with stratification and to improve depletion after atmospheric homogenization. PARs are installed above the floor to provide unobstructed inflow and easy access to facilitate maintenance. PARs automatically start when the threshold hydrogen concentration is reached at the catalytic surfaces. The recombination rate depends on the hydrogen density seen by the PAR. An increasing hydrogen concentration enhances the removal rate up to a design-specific upper limit.

5.9 *Vent Containment*

In case of severe accident in a U.S. EPR™ plant, the reactor building ventilation system is stopped and isolated. The isolation of corresponding containment penetrations is confirmed within the containment isolation performed as an immediate action.

An exception is considered when the equipment hatch is still open at the entry to OSSA. The closure of the hatch is initiated by the EOP or performed in the OSSA as an immediate action. If the complete closure of the hatch is not achieved when the core damage begins, the released fission products could escape the containment through this path. Continued operation of containment ventilation will favor the exhaust of the fission products via iodine filters and, thus, minimize the releases via the large opening of the equipment hatch.

A venting process may be activated in a severe accident situation to release (at a later stage of the accident) the non-condensable gases in order to depressurize the containment and prevent future releases to the environment. The containment venting system is activated when the containment vent pressure has been reached by opening containment isolation valves. The isolation valves can be operated from the main control room or by manual remote control. The venting process can be interrupted or terminated when the predetermined containment pressure has been reached by closing the containment isolation valves.

Table 5-1—U.S. EPR™ Candidate System-Level Actions

Inject into RPV/RCS
Depressurize the RCS
Restart the Reactor Coolant Pumps (RCPs)
Depressurize Steam Generators
Inject into the Steam Generators
Spray into Containment
Inject into Containment
Effect Recombiners
Vent Containment

6.0 END-USER DOCUMENTATION

OSSA directs the actions necessary for plant operators and emergency responders to mitigate the consequences of transients and accidents that cause plant parameters to exceed reactor protection system or engineered safety features performance objectives. It also directs operators to verify event progression and automatic transient mitigation actions.

A goal of OSSA for the U.S. EPR™ design is to identify all detailed main operating strategies required to verify the management of any severe accident. A list of detailed strategies shall cover all potential challenges to ensure that the responsible emergency response team is able to deal with any situation regardless of its probability of occurrence. The severe accident package includes operating guidance for all the emergency crisis organizations. The basic layout is shown in Figure 6-1.

Severe accident specific features are implemented in the U.S. EPR™ design in order to maintain the plant in controlled conditions with no challenges to the confinement of radioactive substances. The use of these features, and when they are used, are defined by the OSSA. Two main categories of challenges have been defined: the immediate and the mitigation path. Accordingly, the documentation will address both types of challenges.

As described in Section 4.4.9, immediate challenges are challenges that might occur when the plant state is in the OSSA Phase 1 domain. In this case the control room operators shall take actions without evaluation by or guidance from the TSC. Immediate actions verify that operators can deal with rapidly occurring scenarios. They are specific systematic actions taken by the operators when the transition is made into the OSSA. Support studies determine and validate the actions the operator may need to take given a particular plant state. These studies support the purpose of the action, relevant setpoints, instrumentation required, and substitute actions. Examples of potential key actions include the depressurization of the primary system; switch over to the 12 hr UPS batteries; opening of the passive flooding line motor operated valve (MOV); and containment isolation. The immediate actions may be performed either at the end of the

EOP or at the beginning of the OSSA, but will be listed in the “Immediate Actions” portion of the OSSA documentation.

Mitigation path challenges are addressed through the evaluation process during OSSA Phase 2 using safety function monitoring. For each challenge, indicated by a change in the safety function status, mitigative high-level strategies can be evaluated. For each of the high level strategies a list of the potential systems that can be used for mitigation are compiled. Support studies determine and validate the recommended actions, related setpoints, potential concerns, available means, and success criteria for each high-level strategy that the TSC and main control room operators may need to use during the progression of a severe accident.

To effectively mitigate a severe accident, the system guidelines must be accurately defined. System guidance is vital during the TSC evaluation and decision making process. A challenge / system matrix is developed to relate each safety function and safety function status to the system that can be used for mitigation. The matrix also lists configurations available to mitigate future and ongoing challenges. For each system to be mentioned in the challenge system matrix a dedicated system guideline will need to be developed. The purpose of the system guideline is to detail the general objective of the configuration, conditions of use, and possible risks. A system availability matrix needs to be developed to assist system guidance by listing the system configurations obtained from various system guidelines and provide information regarding the line-up availability.

The systems guidelines are developed from support studies. The following systems may require a system guideline: containment heat removal system, SI system, CCWS, essential service water system, containment filtered venting system, IRWST system, EBS, CVCS, AVS, leakage detection system on nuclear systems outside the reactor building, gas distribution and storage system / nitrogen subsystem, and fuel pool cooling and purification system.

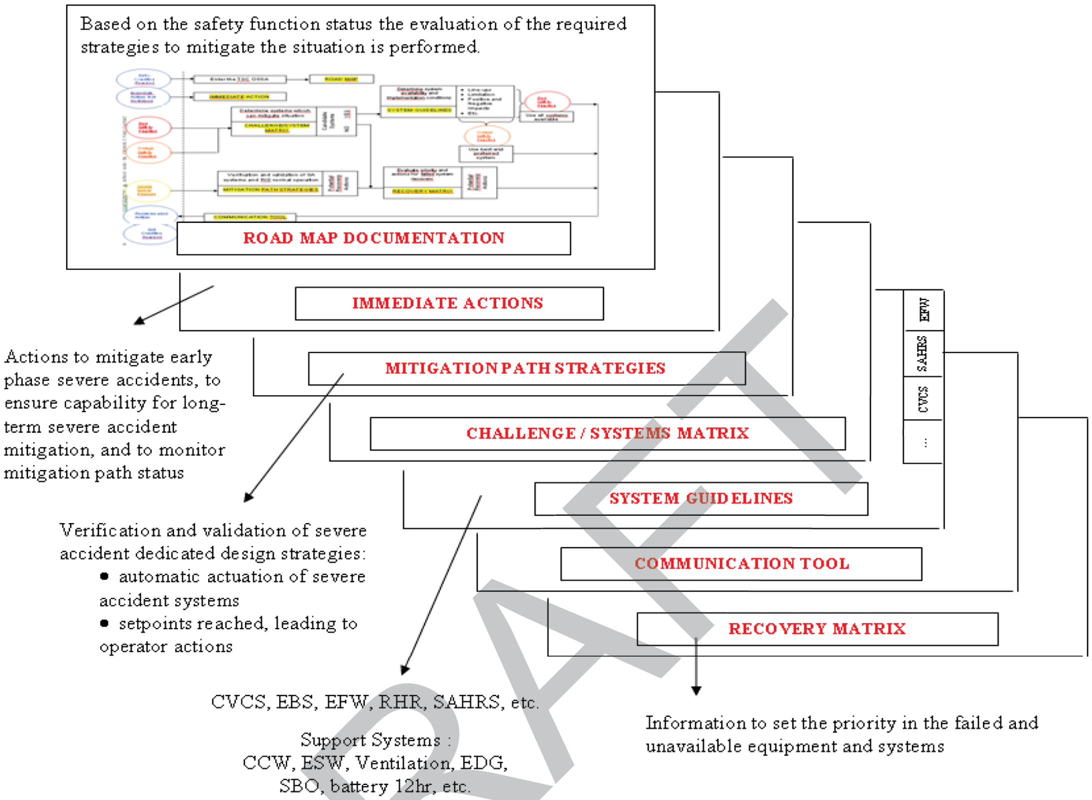
To assist the TSC and operators in recovering from a severe accident, a tool called a recovery matrix will be developed. The recovery matrix will indicate the current state of the accident on a recovery path, or if it has veered from recovery and requires actions to bring the accident back to a more stable condition.

While not a part of the OSSA package itself, support studies need to be performed to provide the foundation on which recommendations are made. These support studies consist of various simulations of severe accident phenomena using best estimate conditions to predict what will likely happen when particular scenarios are encountered and what the consequence will be of particular actions available to the operator. These will likely be performed using the MAAP model of the US EPR™ design and possibly other computer simulations as appropriate.

The following types of documentation will be produced for OSSA:

- Immediate challenges and corresponding actions.
- Remaining challenges documentations:
    - Mitigation Path Strategies.
    - Challenge / Systems Matrix.
    - System Guidelines.
    - Recovery Matrix.
- Support Studies.

Figure 6-1—OSSA Guidance Content



7.0 SUMMARY AND CONCLUSIONS

Severe accident management provides further protection through documented symptom-based guidance or explicit procedures to the responsible emergency response teams. Documented guidance and/or procedures include instructions for the responsible emergency response teams to set the plant upon a mitigation path with the initial objective of establishing a mitigation path leading to an eventual safe plant condition. The combined effect of each of these systems, a robust and leak-tight containment, and the OSSA-derived SAMG confirms that the offsite dose following a severe accident for the U.S. EPR™ design is minimal.

The results of the AREVA NP OSSA methodology appear as part of a utility customer documentation supporting its operating license. Application of the described approach:

- Provides insight into plant-specific severe accident processes and phenomena.
- Estimates the performance of a plant safety functions preserving critical fission product barriers.
- Demonstrates the effectiveness of immediate operator actions and recovery strategies.

Support studies incorporate both deterministic (MAAP4) and probabilistic (RiskSpectrum) methods. These tools are used to refine the understanding of the accident progression domain crediting the best-estimate accident management action during a severe accident at a U.S. EPR™ plant.

8.0 REFERENCES

1. Letter, Ronnie L. Gardner (AREVA NP Inc.) to Document Control Desk (NRC), Request for Review and Approval of ANP-10268P, Revision 0, “U.S. EPR™ Severe Accident Evaluation,” ADAMS Accession No. ML063100154, October 31, 2006.

2. “Defence in Depth in Nuclear Safety,” INSAG-10, IAEA, Vienna, 1996.

3. U.S. NRC, Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, Volume 50, page 32138, of the Federal Register dated August 8, 1985.

4. M. H. Fontana, et al., “IDCOR Technical Summary Report – Nuclear Power Plant Response to Severe Accidents, Technology for Energy Corporation, November 1984.

5. U.S. NRC, SECY-88-147, "Integration Plan for Closure of Severe Accident Issues," May 25, 1988.

6. U.S. NRC, SECY-89-012, "Staff Plans for Accident Management Regulatory and Research Programs," January 18, 1989

7. U.S. NRC Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities - 10CFR 50.54(f)," Supplement 2, April 4, 1990

8. NUREG-0737, “Clarification of TMI Action Plan Requirements,” U.S. Nuclear Regulatory Commission, November 1980.

9. NUREG-0737, “Clarification of TMI Action Plan Requirements: Supplement 1,” U.S. Nuclear Regulatory Commission, January 1983.

10. NUREG-0711, “Human Factors Engineering Program Review Model,” Revision 2, U.S. Nuclear Regulatory Commission, February 2004.

11. NUREG-0899, “Guidelines for the Preparation of Emergency Operating Procedures,” U.S. Nuclear Regulatory Commission, August 1982.NEI 91-04, "Severe Accident Issue Closure Guidelines," Revision 1, December 1994

12. EPRI TR-101869, Vol. 1, "Severe Accident Management Guidance Technical Basis Report, Volume 1: Candidate High-Level Actions and Their Effects," December 1992.

13. EPRI TR-101869, Vol. 2, "Severe Accident Management Guidance Technical Basis Report, Volume 2: The Physics of Accident Progression," December 1992.

14. International Atomic Energy Agency, IAEA Safety Standards No: NS-R-1, "Safety of Nuclear Power Plants: Design," October 31 2000

15. Electricité de France, Technical Guidelines for the Design and Construction of the Next Generation of Nuclear Power Plants with Pressurized Water Reactors, October 2000

16. Nuclear Power International, European Utility Requirements for Light Water Reactors Nuclear Power Plants, Volume 3, Revision A, December 1999.

17. OECD Report, “Implementation of Severe Accident Management Measures,” OECD Workshop Proceedings, NEA/CSNI/R(2001)20, PSI-Villigen, Switzerland, September 2001.

18. OECD Report, “Recent Developments in Level 2 PSA and Severe Accident Management,” NEA/CSNI/R(2007)16, November 2007.

19. Fauske and Associates, Inc., MAAP4—Modular Accident Analysis Program for LWR Power Plants, vol. 2, Part 1: Code Structure and Theory, prepared for Electric Power Research Institute, May 1994.

20. RELCON Scandpower, RiskSpectrum® PSA Professional Theory Manual, Scandpower Risk Management Group, Sweden, 1998.

21. Westinghouse Non-Proprietary Class 3 Document WCAP-14696-A Revision 1, “Core Damage Assessment Guidelines,” (available from the U.S. NRC), November 1999.

22. R. P. Martin, et al., “AREVA NP's Severe Accident Safety Issue Resolution Methodology for the U.S. EPR™,” Proceedings of the ICAPP 2008 Conference, Anaheim, CA, June 2008.

23. R. P. Martin, “Quantifying Phenomenological Importance in Best-Estimate plus Uncertainty Analyses,” Proceedings of the 13th International Topical Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-13), Kanazawa City, Japan, 2009.

24. Framatome ANP Richland Report, “S-RELAP5 models and correlations code manual,” EMF-2100, revision 4 (proprietary), 2001.

25. AFT FathomTM Pipe Flow Analysis System, User’s Guide, Version 6.0, 2004.

26. Travis J. R. et al., “GASFLOW: A Computational Fluid Dynamic Code; Gases, Aerosols, and Combustion, Vol. 1: Theory and Computational Models, Vol. 2: User’s Manual, Vol. 3: Assessment Manual”, LA-13357-M, FZK-5994, October 1998.

ATTACHMENT A
IDENTIFICATION OF SEVERE ACCIDENT CHALLENGES

The application of OSSA to the U.S. EPR™ design develops severe accident management guidelines (SAMG) derived from both deterministic and probabilistic analyses. The objective of these analyses is to identify mission success thresholds and related issues that are important to severe accident mitigation and to demonstrate that the plant-operator tandem is capable of meeting the objective of returning the reactor to a stable safe condition within prescribed limits for the release of radioactive material for each plant state.

Preliminary analyses provide an understanding of the response of the plant to various types of design-basis and beyond-design-basis accidents. Both deterministic analyses and PRA are performed. These analyses identify the nature of challenges to the defense-in-depth objectives (i.e., core and containment integrity), the timing of the various challenges, and the plant parameters useful for monitoring these challenges. To a great extent the preliminary analysis supporting SAMG development are those analyses supporting Chapter 6 (DBA containment analysis), Chapter 15 (DBA safety analysis), and Chapter 19 (severe accidents) content of the U.S. EPR™ FSAR. These analyses were developed to meet the regulatory expectation in NUREG-0800, SRP (complemented by RG 1.206) and SECY-93-087 (References [A.1] and [A.2]).

A.1 Summary of Design Basis Analysis

A plant design basis is characterized by deterministic analyses simulating the performance of relevant systems, structures, and components challenging fuel, reactor coolant system (RCS), and containment integrity. These analyses incorporate plant process and phenomenological uncertainties, including available plant safety-related systems, structures, and components, operator actions, and single-failure assumptions. As such, they define the performance threshold beyond which fission product barriers are assumed to have failed.

Table A-1 summarizes the analysis categories addressed in the U.S. EPR™ FSAR, identifying these categories by the fission product barrier of interest and the principal safety measures. Setpoints on thermal-hydraulic measurements (e.g., steam generator level, core power, pressurizer pressure, etc.) defined within the U.S. EPR™ reactor protection system are implied safety measures in all analyses.

A.1.1 Summary of Plant Systems and Components Available for Mitigation of Accident Effects

Plant systems and components that mitigate postulated design-basis events in the U.S. EPR™ FSAR Tier 2, Chapter 15 accident analyses appear in Table A-2. These safety-related systems are subject to single-failure criteria as described in Section A.1.3. Non-safety-related systems were assumed to function as described in Table A-2.

A.1.2 Summary of Credited Operator Actions

Operator action is credited in certain analyses to mitigate postulated events. In such cases, the action is not credited in the analysis before 30 minutes after event initiation if the action can be performed from the main control room, and 60 minutes if it cannot be performed from the main control room. In addition, operator errors are considered in developing event initiators and in considering limiting single failures (see U.S. EPR™ FSAR Tier 2, Section 15.0.0.3.8 for a more detailed description). The specific operator actions credited in U.S. EPR™ FSAR Tier 2, Chapter 15 accident analyses are as follows:

- Following a feedwater line break (FWLB), the operator is credited to trip two RCPs and redirect the emergency feedwater (EFW) train feeding the affected steam generator (SG) to an intact SG.
- For small main steam line breaks (MSLBs) and FWLBs, the operator is credited with closing the main steam isolation valves (MSIVs) when operating below permissive P12, where the low SG pressure MSIV closure signal is disabled. The small MSLBs do not actuate the low SG ΔP MSIV closure signal.
- Following MSLBs, the operator terminates EFW in the affected SG.
- For the Extra Borating System (EBS) malfunction event, the operator is credited

in terminating the event by either opening letdown or terminating EBS.

- For SG tube rupture (SGTR) event, the operator is credited to perform the following actions:
  - Trip the reactor when the chemical and volume control system (CVCS) is operating.
  - Reset the main steam relief train (MSRT) setpoints high on affected SG and, if necessary, initiate the partial cooldown in the unaffected SGs.
  - Close the MSIV on the affected SG.
  - Close the main feedwater (MFW) isolation valve on the affected SG.
  - Isolate the EFW to the affected SG.
  - Initiate and later manage the medium head safety injection (MHSI) pump.
  - Extend the partial cooldown of the unaffected SGs and depressurize the RCS.
  - Actuate the EBS to add boron to the RCS to maintain subcriticality.
  - For the radiological analysis of the failure of small lines carrying primary coolant outside the reactor building. In U.S. EPR™ FSAR Tier 2, Section 15.0.3.5, operator action is credited to isolate the failed line.

When the plant is in a stable, controlled state, the following additional operator actions are required to bring the plant to residual heat removal (RHR) entry conditions or establish long-term cooling for loss-of-coolant-accidents (LOCAs):

- Use the MSRTs to depressurize the SGs to cool down the RCS.
- Use the EBS to add boron to the RCS to maintain subcriticality.
- Use the pressurizer safety relief valves (PSRVs) to depressurize the RCS.
- Initiate RHR when the RCS reaches the conditions for RHR entry.
- Redirect half of the LHSI flow to the respective hot legs to prevent boron precipitation for LOCAs that are too large for the SI systems to refill the RCS.

A.1.3 Summary of Limiting Single Failures

The U.S. EPR™ FSAR Tier 2, Chapter 15 accident analyses incorporate the most limiting active single failure of a safety-related system. Table A-3 lists the most limiting single-failure for each event. Passive failures are not considered, except as event initiators, during the first 24 hours of the event.

The following pieces of equipment are considered either as passive devices or are designed to be single-failure proof and, therefore, are not subject to single-failure:

- Main steam safety valves (MSSVs).
- PSRVs, when actuated by a spring-driven pilot. A single-failure is considered when the PSRVs are switched to the electrically driven solenoids that reduce their opening setpoints for low-temperature overpressure protection.
- Main steam relief isolation valve (MSRIV), normally closed. This valve is designed to be single-failure proof. Maintenance on the actuating solenoids is limited by Technical Specifications.

A loss-of-offsite power (LOOP) and stuck rod control cluster assemblies (RCCA) are not considered single-failures. A stuck RCCA is incorporated into the reactor trip reactivity insertion. LOOP is incorporated whenever it makes the event more severe.

Operator errors are considered as potential single-failures. An operator error is considered as a potential single-failure for actions expected or directed by emergency procedure (e. g., failure to redirect EFW following FWLB). Operator error is not considered a potential single-failure for actions that are not expected or directed by procedure (e.g., safety injection system (SIS) termination following a legitimate safety injection (SI) signal).

A.2 Summary of PRA Methods

The U.S. EPR™ PRA included an evaluation of the types of accidents that could lead to core damage (including the impact of operator action or inaction), an assessment of event frequencies, an analysis of the containment response to these accidents, and characterization of the magnitude and frequencies of releases of radionuclides that could result. The PRA addressed all applicable internal and external initiating events and all plant operating modes. Event trees were constructed for both Level 1 and Level 2 analyses to represent plant response and graphically illustrate the combinations of successes and failures of systems and operator actions that define particular accident sequences. The definition of success at the end of the Level 2 PRA is the state with the containment intact.

A.2.1 PRA Level 1 Analysis

With regard to severe accidents, deterministic analysis were performed to demonstrate the regulatory objective that containment integrity be maintained for approximately 24 hours following the onset of core damage for the more likely severe accident challenges. In addition, following this period, the containment was shown that it continues to provide a barrier against the uncontrolled release of fission products. The more likely scenarios considered in the development of U.S. EPR™ FSAR Tier 2, Chapter 19 analyses included:

1. Loss of offsite power with Seal LOCA.
2. Loss of offsite power with a low pressure end state.
3. Loss of offsite power with a high pressure end state.
4. Loss of balance of plant.
5. Small LOCA (2 to 8.5-inch).

These scenarios were derived using results from Level 1 PRA by identifying those initiating events whose core damage frequency exceeds 1.0E-8/yr and identifying a corresponding Core Damage End State (CDES). CDES are used by PRA to link the Level 1 core damage event trees to the Level 2 containment event trees. This is done by bringing together core damage sequences with similar characteristics, and using those sequences as the initiating event for examining severe accident mitigation and containment failure probability.

A.2.2 PRA Level 2 Analysis

Success in the accident management guidelines is to remain on the mitigation path (i.e., a severe accident scenario that retains containment integrity and minimizes containment radiological releases). The Level 2 PRA identifies the broad range of severe accident scenarios that combine the failures of human actions (including acts of omission), failure of system operation, and severe accident phenomenology that can challenge containment integrity after core damage. The Level 2 PRA also examines the arrest of core damage prior to vessel failure, by successful primary depressurization and primary system injection. This scenario in the Level 2 PRA leads to end states defined as limited core damage, and this scenario is congruent with success in the OSSA Phase 1 domain.

The examination of severe accident progression in the Level 2 model includes examination of the heat removal, containment integrity, and radiological release safety functions. The scope of the U.S. EPR™ Level 2 PRA includes evaluation of all plant operating states, both at-power and shut down.

The Level 2 PRA is performed using a combination of deterministic and probabilistic analyses consisting of the following:

- Accident progression analysis to support development of the containment event trees (CETs).
- Integration of the Level 1 and Level 2 analyses through the definition of CDES as consequences in the Level 1 analysis, and linking these consequences to the input portion of the CET.
- Identification of physical phenomena important to containment integrity that could occur in the course of severe accidents.
- Examination of human errors and failures of system operation which can lead to containment challenge.
- Development of release category bins to characterize fission product magnitude and migration paths.

The following sections provide a discussion of the Level 2 PRA and its relationship to the OSSA.

A.2.3 General Structure of the Level 2 PRA

The general structure of the Level 2 PRA is shown in Figure A-1. Within the U.S. EPR™ RiskSpectrum model, core damage sequences that result in similar characteristics are collected in CDES. Each of the core damage sequences of a particular CDES are collected into a CDES link tree, where it is examined for possible early recovery of core heat removal and directed to one of the CETs. Within the CETs, the mechanisms for containment failure from severe accident phenomenon are examined, as well as the reliability of the systems and operators previously described.

Level 1 to Level 2 Interface through the CDES

CDES provide the interface between the Level 1-to-Level 2 analyses -- between core-damage accident sequences and fission product release categories. They are also useful in identifying characteristic plant state symptoms to be incorporated into SAMG.

The CDES have been designed to link the Level 1 core damage event trees to the Level 2 CETs by bringing together core damage sequences with similar characteristics, and using those sequences as the initiating event for the appropriate CET. CDES have been assigned to each instance of core damage in the Level 1 PRA.

Because the CETs for the U.S. EPR™ contains system-related top events not usually found in previous Level 2 studies supporting the current generation of PWRs, the CDES distinguish between significant groups of core damage sequence types by including the following information from the Level 1 event trees:

- Types of Sequences (e.g., Transients, LOCAs).
- Condition of the Containment (no bypass, SGTR, interfacing system LOCA).
- System-Related Plant Status:
    - Offsite Power.
    - Feedwater.

- Steam Generator Pressure and Isolation.
- Feed and Bleed.

Another significant distinction among the CDESs is driven by the difference in success criteria between the Level 1 and Level 2 PRAs. There are a number of CDESs where core damage is defined to occur because the systems available for feed and bleed do not meet the Level 1 success criteria. This same combination of systems can be successful in limiting the extent of core damage in Level 2.

Accident Progression from Level 1 through Level 2

After the assignment of a CDES to each instance of core damage in the Level 1 PRA, each individual endstate is transferred to an intermediate event tree, referred to as CDES link tree, prior to transfer to a Level 2 PRA CET. The use of these CDES link event trees serves the following purposes:

1. Provides a consistent structure for linking the Level 1 and Level 2 PRA models.
2. Allows marking the cutsets coming from each CDES with a flag event which can later be used to establish the contribution of different CDES to the release category frequencies.
3. Allows treatment of various core damage sequences where the Level 1 PRA had applied restrictive success criteria, so that these sequences could be reassessed and classified as limited core damage cases. These limited core damage sequences are the sequences that would be congruent with success in the OSSA Phase 1 domain.

When the incoming sequences from the Level 1 have passed through the CDES link trees they are then transferred to the appropriate CET model (see Figure A-1). The possible transfers from the CDES link trees to CETs are:

- CET-ISL – this is the CET for interfacing LOCA. All accident sequences initiated by interfacing system LOCA are evaluated in this tree.
- CET SGTR – this is the CET for steam generator tube rupture sequences. This

CET evaluates sequences that are initiated by SGTR, as well as those resulting from induced rupture of the RCS.

- CET SGTR FW – this is the CET for the sequences that are initiated by SGTR with feedwater running.
- CET1 HI PRESSURE – this is the CET for RPV failure at high pressure CDES
- CET LO PRESSURE – this is the CET for RPV failure at low pressure CDES or depressurized CDES
- CET LIMITED CD – this is the CET for sequences identified as limited core damage cases in CDES link trees.

When sequences are transferred into a CET, they generally pass through only that CET and are assigned to a release category which is marked on the end of each CET sequence. As shown in Figure A-1, the one exception to this pattern is CET1 HI PRESSURE. There are three possible outcomes for the accident sequences entering CET1 HI PRESSURE:

1. Accident sequences that result in induced SGTR are assigned to the SGTR release categories, the RC700s.

2. If manual primary depressurization is successful or a hot leg rupture occurs, the sequence is transferred to the low pressure CET (CET LO PRESSURE).

3. If the accident sequence remains at high pressure, the sequence is transferred to the second high pressure CET (CET2 HI PRESSURE).

The top events included in the CETs address the phenomenological events, systems, and human actions examined during severe accident progression in the Level 2 PRA. The top events included in the CET are those which are expected to have a significant impact on the severe accident progression, meaning that they can affect, directly or indirectly, either the likelihood of containment failure or bypass or the magnitude of the source term.

For convenience, the events considered within the Level 2 CETs are grouped into the following time frames:

1. Time frame 1 (TF1), which considers the period from the onset of core damage up to the time of vessel failure (if this occurs).

2. Time frame 2 (TF2), which considers the period close to the time of vessel failure.

3. Time frame 3 (TF3), which considers long-term events after the time of vessel failure.

Relevant events considered in time frame 1, which correspond to the OSSA Phase 1 domain, are:

- Failure of containment isolation.
- Induced RCS failures.
- Depressurization of RCS by the operators.
- Hydrogen combustion.

Relevant events in time frame 2, which correspond to the end of OSSA Phase 1 domain and the beginning of OSSA Phase 2 domain, are:

- Melt retention in-vessel.
- In-vessel steam explosion (failing containment or damaging the reactor pit).
- Loads at vessel failure leading to containment failure (direct containment heating (DCH), hydrogen or vessel rocketing).
- Ex-vessel steam explosion after vessel failure (damaging the reactor pit).

Relevant events considered in time frame 3, corresponding to OSSA Phase 2 domain, are:

- Melt transfer to the spreading area.
- Initial stabilization of melt ex-vessel.

- Steam overpressure during quenching leading to containment failure.
- Hydrogen combustion.
- Steam overpressurization long-term.
- Long-term containment overpressure failure.
- Basemat failure due to core concrete interaction.
- Use of containment sprays for source term mitigation after containment overpressure failure.

A.2.4 System Analysis in the Level 2 PRA

The operation and failure of a number of systems are examined in the Level 2 PRA. The systems modeled in the Level 2 PRA support the three OSSA safety function objectives: heat removal, containment integrity, and control of radiological releases. The reliability of instrumentation signals and indications is included in the determination of the reliability of any system included in the Level 2 PRA.

Heat Removal Safety Function

The Level 2 PRA examines the operator actions system failures that would inhibit the actuation of the LHSI system or the depressurization of the primary system either via the PDS valves or the pressurizer safety valves, to restore core heat removal after core damage.

For the function of containment heat removal, the SAHRS is included in the Level 2 PRA. The Level 2 analysis examines the failures of operator action and system failures that can lead to a loss of the SAHRS and a failure of this system to provide core spreading area cooling, containment spray cooling, and core spreading area basemat cooling.

Containment Integrity

The Level 2 PRA examines the containment isolation system, the primary system that supports the maintenance of containment integrity. The failure of automatic isolation of the containment, and the failure of the operator to back up the automatic isolation with manual actions are examined in the Level 2 PRA as failures to isolate, and lead to end states with the containment bypassed.

Control of Radiological Releases

Within the domain of control of radiological releases from containment, the SAHRS containment spray mode is examined for its contribution to the reduction of radiological releases from the containment by its use in the containment spray mode to provide scrubbing of the airborne fission products from the containment atmosphere prior to containment failure and fission product release.

A.2.5 Source Term Evaluation

The final step in the Level 2 PRA analysis is the evaluation of the frequency, the magnitude, the time dependence and composition of the fission product releases which characterize containment failure.

The end states of the CETs are grouped into release categories. The release categories are chosen to group containment failures with similar characteristics. Some of these characteristics include whether the containment is bypassed, the time frame in which the containment failure occurs, whether the core melt is retained in-vessel, whether or not the corium is flooded ex-vessel, whether or not MCCI occurs, and whether the operators spray the containment atmosphere for fission product scrubbing.

The release categories are grouped in the following “families”, based on the characteristics describe above:

- RC100s – Containment intact.
- RC200s – Containment isolation failures.
- RC300s – Containment failures prior to vessel breach.
- RC400s – Containment failures after vessel breach and up to melt transfer to spreading area.
- RC500s – Long-term containment failures during and after debris quench.

- RC600s – Basemat failure.
- RC700s – Steam Generator Tube Rupture.
- RC800s – Interfacing System LOCAs.

The source term evaluation for the Level 2 PRA showed the importance of in-vessel core melt retention as a major factor in reducing fission product releases from the containment. This insight underscores the importance of the actions taken in ECHUR OSSA domain to terminate core damage before containment failure.

The source term analysis also shows that operator initiation of containment sprays can help avoid containment overpressure failure and greatly reduce the magnitude of fission product release if a containment failure occurs. The source term analysis also shows the importance of re-filling the steam generators to avoid induced SGTR, and to scrub the releases from all SGTRs after they occur.

A.3 U.S. EPR™ Severe Accident Challenges

The emphasis on severe accident challenges is on containment integrity for at-power and shutdown operating modes. This is the domain of Level 2 PRA. The purpose of the Level 2 PRA is to examine the response of the containment and its related systems to potential loads and to assess characteristics of radiological releases accompanying severe core damage accidents. Success in the Level 2 PRA is defined as maintaining containment integrity throughout the progression of the severe accident. Failure in the Level 2 PRA results in events that can cause the loss of containment integrity, and the Level 2 PRA examines both the likelihood and magnitude of fission product release when containment integrity is lost.

Based on the results of the Level 2 PRA, Table A-4 shows the various challenges at-power, shutdown and refueling conditions for the containment. Additionally, if there are any manual actions required for the functioning of the design measure, these are noted as well in order to identify additional possible causes of the failure. Lastly, the final three columns present an initial attempt to try to identify potential mitigation strategies and subsequent actions than could be taken in the TSC, along with additional comments that may provide supplemental information regarding the accident or the analysis.

Table A-1—U.S. EPR™ Design-Basis Analysis Categories

FSAR Section	Subject	Fission Product Barrier	Principal Safety Parameters
6.2.1	Containment Functional Design	Containment	Pressure; Temperature
6.2.5	Combustible Gas Control in Containment	Containment	H_2 Concentration
6.3	Emergency Core Cooling System	Fuel	In-containment Refueling Water Storage Tank Heat Removal; Net Positive Suction Head
15.1	Increase in Heat Removal by Secondary System	Fuel; RCS	Departure from Nucleate Boiling Ratio (DNBR); Fuel Centerline Melt (FCM); RCS Pressure
15.2	Decrease in Heat Removal by Secondary System	Fuel; RCS	DNBR; FCM; RCS Pressure; Secondary Pressure
15.3	Decrease in Reactor Coolant System Flow Rate	Fuel	DNBR
15.4	Reactivity and Power Distribution Anomaly	Fuel; RCS	DNBR; FCM; Cladding Strain; RCS Pressure
15.5	Increase in RCS Inventory	Fuel; RCS	DNBR; RCS Pressure
15.6	Decrease in RCS Inventory	Fuel; RCS	DNBR; RCS Pressure; Peak Clad Temperature; Fuel Oxidation; H_2 Production

Table A-2—Plant Systems Used in the Accident

Incident	Reactor Trip Functions	Engineered Safety Features Functions	Other Equipment
15.1 Increase in Heat Removal by Secondary System			
Decrease in feedwater temperature	• Low DNBR • High Linear Power Density (LPD) • High core power		
Increase in feedwater flow	• High steam generator (SG) level • Low DNBR • High LPD		
Increase in steam flow	• Low DNBR • High LPD • High core power • Low SG pressure • High SG ΔP	• Main Feedwater (MFW)/Startup-Shutdown System (SSS) isolation on low SG pressure or high SG ΔP • Safety Injection System (SIS) and partial cooldown on low RCS pressure • Main Steam Isolation Valve (MSIV) closure on low SG pressure or high SG ΔP	
Inadvertent opening of a SG relief or safety valve	• Low DNBR • High LPD • High core power • Low SG pressure • High SG ΔP	• MFW/SSS isolation on low SG pressure or high SG ΔP • SIS and partial cooldown on low RCS pressure • Main Steam Relief Train (MSRT) isolation on low SG pressure • MSIV closure on low SG pressure or high SG ΔP	

Incident	Reactor Trip Functions	Engineered Safety Features Functions	Other Equipment
Steam system piping failure	• High core power • Low DNBR • High LPD • Low SG pressure • High SG ΔP	• MSIVs closure on high SG ΔP or low SG pressure • Affected SG MFW/SSS isolation on high-high SG ΔP or low-low SG pressure • Unaffected SG MSRTs opening on high SG pressures • Stuck-open-MSRCV MSRT isolation on low-low SG pressure • SIS and partial cooldown on low-low pressurizer (PZR) pressure, or SIS on low margin to RCS saturation	
15.2 Decrease in Heat Removal by Secondary System			
Turbine Trip	• High SG pressure • High PZR pressure	• MSRTs on high SG pressure	Pressurizer Safety Relief Valves (PSRVs), Main Steam Safety Valves (MSSVs)
Closure of a MSIV	• Low DNBR • High SG pressure • High PZR pressure	• MSRTs on high SG pressure	PSRVs, MSSVs
Loss of non-emergency AC power	• Low RCP speed • Low RCS flow (2 loops) • High PZR pressure	• Emergency Feedwater System (EFWS) on low SG level • MSRTs on high SG pressure	PSRVs
Loss of normal feedwater flow	• Low DNBR • Low SG level	• EFWS on low SG level	PSRVs

Incident	Reactor Trip Functions	Engineered Safety Features Functions	Other Equipment
Feedwater system pipe break	• Low SG pressure • High SG ΔP • High containment pressure • Low SG Level • High PZR pressure	• EFWS on low SG level • MSIV closure on low SG pressure or high SG ΔP • MFW/SSS isolation on low SG pressure or high SG ΔP	PSRVs
15.3 Decrease in Reactor Coolant System Flow Rate			
Partial loss of forced reactor coolant flow	• Low-low RCS flow (1 loop) • Low RCS flow (2 loops)		PSRVs
Complete loss of forced reactor coolant flow	• Low RCP speed • Low RCS flow (2 loops)		PSRVs
Reactor Coolant Pump (RCP) rotor seizure	• Low-low RCS flow (1 loop).		PSRVs
RCP shaft break	• Low-low RCS flow (1 loop)		PSRVs
15.4 Reactivity and Power Distribution Anomaly			
Uncontrolled Rod Control Cluster Assemblies (RCCA) bank withdrawal from a subcritical or low power startup condition	• High flux rate		
Uncontrolled RCCA bank withdrawal at-power	• Low DNBR • High LPD • High core power • High flux rate		
Single RCCA withdrawal	• Low DNBR		
RCCA misalignment	• Low DNBR		
RCCA drop	• Low DNBR		
Startup of a RCP in an inactive loop	N/A		

Incident	Reactor Trip Functions	Engineered Safety Features Functions	Other Equipment
Inadvertent decrease in the boron concentration in the RCS	• Low DNBR • High core power		Anti-dilution
Inadvertent loading and operation of a fuel assembly in an improper position	N/A		
RCCA ejection	• High flux rate • High flux		
15.5 Increase in RCS Inventory			
Inadvertent operation of the Emergency Core Cooling System (ECCS) or Extra Borating System (EBS)	High PZR level	• MSRTs on high SG pressure	PSRVs on PZR pressure
CVCS malfunction that increases reactor coolant inventory	High PZR level	• Control Volume Control System (CVCS) isolation on PZR level • MSRT on high SG pressure	PSRVs on PZR pressure
15.6 Decrease in RCS Inventory			
Inadvertent opening of a pressurizer relief valve	Low PZR pressure	• SIS/partial cooldown on low RCS pressure • Containment isolation	RCP trip
Steam Generator Tube Rupture (SGTR)	• Low DNBR • Low PZR pressure	• SIS/partial cooldown CS pressure • MSRTs on high SG pressure	EBS EFW level control

Incident	Reactor Trip Functions	Engineered Safety Features Functions	Other Equipment
Loss-of-coolant accident (LOCA)	• Low PZR pressure • High containment pressure • Low Hot Leg Pressure	• SIS/partial cooldown on low RCS pressure • Containment isolation • MSRTs on high SG pressure • EFWS on SG Level	RCP trip

Table A-3—Single Failures Assumed in the Accident Analysis

Event	Failure
Increase in Heat Removal by Secondary System	
Decrease in feedwater temperature	One protection division
Increase in feedwater flow	Failure of a feed line isolation valve
Increase in steam flow	One protection division
Inadvertent opening of a SG relief or safety valve	One MSRCV fails to close
Steam system piping failure	One MSRCV fails open
Decrease in Heat Removal by Secondary System	
Turbine Trip	One MSRCV fails to open
Closure of a MSIV	One MSRCV fails to open
Loss of nonemergency AC power	One EFW train
Loss of normal feedwater flow	One EFW train
Feedwater system pipe break	One EFW train One MSRT fails to open
Decrease in Reactor Coolant System Flow Rate	
Partial loss of forced reactor coolant flow	One protection division
Complete loss of forced reactor coolant flow	One protection division
RCP rotor seizure	One protection division
RCP shaft break	One protection division
Reactivity and Power Distribution Anomaly	
Uncontrolled RCCA withdrawal from a subcritical or low power startup condition	One protection division
Uncontrolled RCCA bank withdrawal at-power	One protection division
Single RCCA withdrawal	One protection division
RCCA misalignment	One protection division
RCCA drop	Failure of highest excore signal input to CRDCS
Startup of a RCP in an idle loop	No protection features are challenged
Decrease in the boron concentration in the RCS	One protection division
Inadvertent loading and operation of a fuel assembly in an improper position	No protection features are challenged
RCCA ejection	One protection division

Event	Failure
Increase in RCS Inventory	
Inadvertent operation of the ECCS or EBS	EFW train
CVCS malfunction that increases reactor coolant inventory	EFW train
Decrease in RCS Inventory	
Inadvertent opening of a pressurizer relief valve	One EDG (1 train of SI)
SGTR	MSRT stuck open
Loss-of-coolant accident	One EDG (1 train of SI)

Table A-4—Challenges and Potential Mitigation Strategies

Challenge	Design Measures in Place	Manual Actions Required for Design Measures	Potential Mitigation Strategies	Potential Mitigation Actions
Interfacing Systems LOCA	Isolation Valves	Close external isolation valves, if not automatic		
Hydrogen deflagration or flame acceleration during the in-vessel phase of high pressure accident sequences during discharge from the RCS via the pressurizer valves	PARs, H_2 Concentration Monitors, large containment volume	None	- Catalytic recombination - Inerting - Venting - Shut down heat sinks	- Inert atmosphere by addition of steam or non condensable. - Shut down the SAHRS to retain steam inerting.
Hydrogen deflagration or flame acceleration during the in-vessel phase of high pressure accident sequences that result in hot leg rupture and release of hydrogen to the containment.				

Challenge	Design Measures in Place	Manual Actions Required for Design Measures	Potential Mitigation Strategies	Potential Mitigation Actions
Hydrogen flame acceleration during low pressure scenarios with short-term fast MCCI following vessel failure				
Induced SGTRs	Dedicated and redundant lines for primary system depressurization via - 2 x 100% PDS - 3 x 33% PSRV Main steam isolation valves (MSIV), steam generator safety and relief valves & downstream valves	Open primary system depressurization line	- Isolate steam generators - Scrub releases	- Fill steam generators, use emergency feedwater system EFW or MFW to maintain steam generator level high, continue RCS depressurization. - Close MSIV, steam generator safety and relief valves & all downstream valves to avoid bypass.
Severe accident following SGTR initiating event (Containment Bypass)	No severe accident specific design measures in place.	None	- Isolate steam generators - Scrub releases	- Depressurize RCS, Fill SGs, use EFWS or MFW to maintain steam generator level high, continue RCS depressurization. - Close MSIV, steam generator safety and relief valves & all downstream valves to avoid bypass.

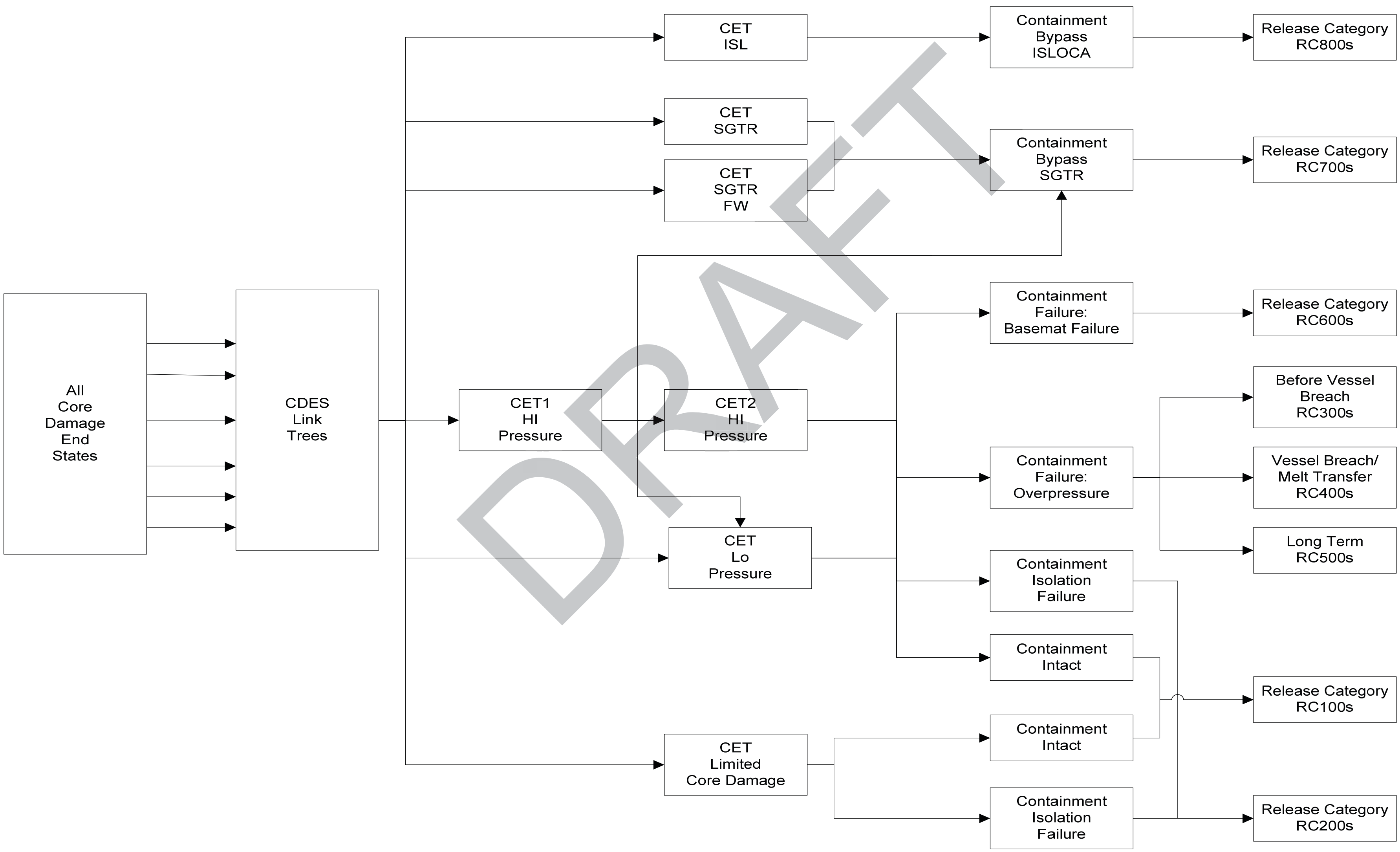
Challenge	Design Measures in Place	Manual Actions Required for Design Measures	Potential Mitigation Strategies	Potential Mitigation Actions
In-Vessel Steam Explosion	Robust RPV design	None	Nothing for in-vessel.	Analyses show that the no design measures are necessary because the mixing mass is not sufficient to supply enough energy to a slug in order for it to deform the upper head.
Containment Isolation Failure	Redundant containment isolation valves. - Exterior valves on severe accident I&C, inside available for first 2 hours of event, shut afterwards if not already	Close interior isolation valves	Isolate containment	- Close all containment isolation valves as part of the immediate actions. - In long-term diagnostics, continuously monitor containment isolation valve position.
RPV Rocket	Dedicated PDS for severe accidents via - 2 x 100% PDS valves - 3 x 33% PSRVs	Open PDS valves	Verify opening of PDS valves	
DCH	Dedicated PDS for severe accidents via - 2 x 100% PDS valves - 3 x 33% PSRVs	Open depressurization line	Verify opening of PDS valves	

Challenge	Design Measures in Place	Manual Actions Required for Design Measures	Potential Mitigation Strategies	Potential Mitigation Actions
Large containment failure due to hydrogen flame acceleration during outage	PARS, H_2 concentration monitors, large containment volume	None	- Inerting - Venting - Shut down heat sinks	- Inert atmosphere by addition of steam or non condensable. - Shut down the SAHRS.
Overpressure due to Generation of Non Condensable gases from MCCI	- Spreading corium over large surface in core catcher, - Refractory material able to handle the heat flux from the corium, - Passively flooding with water drained from IRWST above, along the side, and below - IRWST Equipped with leaktight liner - SAHRS	Long-term surveillance	Reestablish long-term cooling	Flood the spreading compartment, filling it by all means necessary if a train is not available
Quenching	Flooding of the melt at a moderate rate.	Start SAHRS within 12 hours upon entering into SA	Find supplemental means to cool the containment	Activate SAHRS. Avoid injecting water on molten corium still in the RPV or in Reactor Pit.

Challenge	Design Measures in Place	Manual Actions Required for Design Measures	Potential Mitigation Strategies	Potential Mitigation Actions
Basemat Penetration	- Spreading Corium over large surface in core catcher, - Refractory material able to handle the heat flux from the corium, - Passively flooding with water drained from IRWST above, along the side, and below - IRWST equipped with leaktight liner	Long-term surveillance	Reestablish long-term cooling	Flood the spreading compartment, filling it by all means necessary if a train is not available
Large containment failure due ex-vessel steam explosion	- Temporary retention in reactor pit - Reactor pit and spreading area are dry prior to gate failure - Incorporation of sacrificial concrete (light oxides) - Reactor pit structure designed to 20 MPa	None	- Inerting - Venting - Shut down heat sinks	- Inert atmosphere by addition of steam or non condensable. - Shut down the SAHRS.

Challenge	Design Measures in Place	Manual Actions Required for Design Measures	Potential Mitigation Strategies	Potential Mitigation Actions
Overpressure due to Steam	Large containment volume, SAHRS & possibly venting	Long-term surveillance	Find supplemental means to cool the containment	Repair SAHRS as may be necessary

Figure A-1—Structure of the Level 2 PRA



A.4 References

A.1. U.S. NRC, Regulatory Guide 1.206, “Combined License Applications for Nuclear Power Plants,” March 2007.

A.2. U.S. NRC, SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water (ALWR) Designs,” issued April 2, 1993, and the corresponding SRM, issued July 21, 1993.

ATTACHMENT B
POSSIBLE OSSA SUPPORT STUDIES

This technical guide does not provide specific detail on the analyses that should be performed and/or used in the frame of the OSSA. SAMG developers will have to evaluate the needs as the tasks progress. Nonetheless, several analysis categories and related studies are presented below.

Section	Description
1	**Basis for Entry Conditions**
1.1	(At Power) Relation between Tclad max and T_{COT} max for wider range of pressure and accident scenario:
1.2	(Shutdown) Releases from fuel and relation with T_{COT}: (Development of final setpoint definition based on international examples)
1.3	(Shutdown) Containment radiation versus time: Unique Considerations: RCS opening considered - RHR valves opening and/or primary vessel head removed
1.4	Source range response to core uncovery
1.5	S-RELAP5 sensitivity studies to identify latest point for RCS depressurization (part of EOP?)
1.6	Loss of feedwater accident, no SI, depressurization on different T_{COT} criteria
1.7	Hydrogen concentration near the PRT tank and in the dome at Tclad = 2200°F (1477°C): (Warning if used the hydrogen monitoring shall be started as immediate action.)
1.8	(Spent fuel conditions) Radiation in fuel building corresponding to both the low water level and the manipulation accident; clarify the range of instrumentation to allow monitoring of radiation levels.
1.9	EOP scenarios leading to T_{COT} = 1200°F (650°C) as an initial peak temperature
2	**Safety Functions / Diagnostic**
2.1	Releases safety function Computational Aid (CA): calculated releases for sequences on the mitigation path, and limits for DBA (Development of site dose, stack radiation, annulus radiation, safeguard building dose rate, safeguard building ventilation dose rate.)
2.2	Containment (Pressure/Hydrogen) safety function CA:

Section	Description
	Gas compositions and AICC pressure calculations (address impact of vented flowrate to be assessed)
2.3	Containment (Pressure/Hydrogen) safety function CA: Assess setpoints for peak hydrogen and pressure in containment for sequences on the mitigation path
2.4	Containment (Pressure/Hydrogen) safety function CA (MCCI): Assess relationship between hydrogen production and impact on containment pressure
2.5	Containment (Pressure/Hydrogen) safety function CA: Containment fragility/structural analysis needed for level 2 PSA at high and low pressure
2.6	Setpoint for SAHRS start: Evaluation of the sensitivity on maximum IRWST temperature
2.7	Setpoint for SAHRS start: Evaluation on the sensitivity on releases
3	**Strategies**
3.1	Potential impact of Steam Generator on core heat up (emphasis on intermediate pressures)
3.2	Consequences of injection at low flowrate:
3.3	Late phase injection - consequences on H_2, pressurization and in-vessel melt retention, timing of scenarios (dependant on flowrate):
3.4	Boron concentration requirements to avoid re-criticality:
3.5	Containment P/H2 response to reflood and depressurization of RCS (including off mitigation path):
3.6	Develop criteria for initiation of SAHRS (effect on H_2, releases), and switch to recirculation mode (parametric studies)
3.7	Response of vessel failure instrumentation following the relocation to RPV lower head in the configurations with reflooded core (i.e., like TMI) or not reflooded core
3.8	Response of the SAHRS spray usage on the hydrogen concentration
3.9	In-vessel injection: Assessment of the capability to ensure simultaneous long-term operation of the SAHRS and the SIS pumps in suction on the IRWST
4	**Additional Computational Aids (not used for Safety Function Monitoring)**
4.1	Containment depressurization rates for various initial containment conditions using

Section	Description
	different venting lines
4.2	Calculation of minimum injection flowrate needed to quench / reflood core
4.3	The efficiency of the PARs depending on different parameters (as hydrogen and steam concentration)
5	**Setpoints**
5.1	Assessment of the temperature evolution in containment for different scenarios: • without SAHRS and without active cooling • without SAHRS and with active cooling • with SAHRS and without forced recirculation • with SAHRS and with forced recirculation
5.2	Ex-vessel configurations: Assessment of the temperature evolution inside the core catcher chimney: • opening of the passive flooding valve(s) (on-mitigation path) • common cause failure on both flooding valves (off-mitigation path) Assessment of the temperature evolution beneath the core catcher (thermocouples in the central cooling channel) for: • opening of the passive flooding valve(s) (on-mitigation path) • common cause failure on all flooding valves (off-mitigation path)

ATTACHMENT C
SEVERE ACCIDENT PROGRESSION IN THE U.S. EPR™ DESIGN

The in-vessel phase of core melt progression for at-power scenarios without reflooding consists of the following phases (additional description provided in Section 4 of the Severe Accident Evaluation Topical Report):

t0: Core uncovery resulting from gradual loss of coolant water.

t1: Core heat up, cladding oxidation and fission product release.

t2: Core melt onset with eutectic interactions between core materials, relocation of cladding, structural materials and fuel with formation of blockages near the bottom of the core forming a molten pool.

t3: Onset of massive relocation into the lower head of the vessel. Generic behavior is characterized by natural convection in a volumetrically heated molten pool resulting in sideward relocation through the core barrel (i.e., heavy reflector in the U.S. EPR™ design) to the lower plenum.

This first relocation occurs into a water filled lower plenum. Some or all of the relocating material may break up and become part of particulate debris consisting of both the oxidic and metallic constituents. Between times t3 and t4, there is a phase of corium heat up leading to dry out of debris in the lower plenum which remelts and forms a molten pool involving development of crusts on the top and along the vessel wall.

t4: Vessel failure, according to several possible mechanisms: 1) molten metal located on top of the oxidic melt can thermally attack and weaken the vessel wall, or 2) internal residual pressure, weight of the corium and thermal loads result in creep rupture.

t5: Corium in reactor pit with molten-core-concrete interaction (MCCI) which causes ablation of a predefined thickness of sacrificial concrete and leads to a temporary phase of melt retention in the reactor pit in which all remaining melt from the vessel is collected.

t6: Corium in spreading room with MCCI leading to ablation of sacrificial concrete floor and side walls. Corium arrival in spreading room passively initiates gravity-driven overflow of water from the in-containment refueling water storage tank (IRWST) which cools and quenches the spread melt.

The phases listed above are described in more detail in the following sections.

C.1 Core Uncovery

When the fuel is located in the reactor vessel the loss of cooling water is likely to be due to either a loss of coolant accident (LOCA) or due to the opening of the pressurizer relief valves without any further water injection. When this occurs, core uncovery takes place as the water flows out of the RPV. In the Fuel Building the malfunction of the spent fuel pool cooling system can lead to the same situation. When the water level descends below the top of the fuel rods, the uncovered rods produce more heat than the generated steam below the mixture level is able to remove.

In order to try to mitigate the further accident progression and avoid a high pressure core melt scenario, when the core outlet temperature reaches 1200°F (650°C), there is a final call to open the primary depressurization system (PDS) valve(s) to depressurize the reactor coolant system (RCS) (if not yet performed). If the RCS is already depressurized, or if the depressurization and the injection of the safety injection system (SIS) accumulator or the low head safety injection (LHSI) do not ensure enough time for the recovery of other injection means to the RCS, core degradation may continue.

C.2 Core Heat-Up Phase

As the temperature may still continue to rise, an exothermic reaction occurs in which the Zirconium (Zr) of the cladding interacts with the oxygen of the steam. This chemical reaction is accompanied by the generation of hydrogen.

If the RCS is not depressurized and the secondary side heat removal is in progress, the circulation of hot steam throughout the core and into the upper plenum of the RPV dictates the temperature of the gas in the piping, in the pressurizer, and in the steam generators. When the cladding temperature reaches its melting point, the integrity of the cladding may become compromised and lead to the release of fission products. As the steam interacts with the Zr in the cladding, hydrogen production may begin. As the cladding temperature increases, oxidation becomes significant and begins to dominate the heat-up of the fuel rods. However, during postulated normal boil off scenarios, insufficient steam can limit the oxidation speed (steam starved conditions) and, because of this, it also limits the global heatup rate of the core.

C.3 Core Melting Phase

When the temperature increases even further, competition begins between the oxidation of the Zr and the dissolution of the UO_2 by the Zr towards a lower-melting Zr-U-O eutectic. After the formation of lower temperature eutectics, the first of which is B_4C with Fe, these eutectics become the driving force in the development of the melt pools between the fuel elements and their maximum temperature. When the melting fuel elements are located in the reactor vessel, the generated melt spreads axially and radially within the core. Several solidification and remelting processes lead to the formation of a molten pool, which is enclosed by a crust. The crust is supported by a non-molten core structure. In case of a global failure of the crust, or molten pool penetration of the heavy reflector and the core barrel, the melt relocates to the lower plenum.

When the RCS is pressurized, natural circulation of hot steam and hydrogen between the core and the upper plenum, and between the RPV and the steam generators leads to a heatup of the hot legs, surge line, pressurizer, and steam generator tubes.

C.4 In-Vessel Fuel-Coolant Interaction

While considered unlikely, in-vessel fuel-coolant interactions (FCI) from contact of molten corium and water in the lower plenum may result in a steam explosion with associated high mechanical loads on the RPV. For a severe accident taking place in the RPV, if the RPV fails in spite of the presence of residual water in the lower plenum, the mechanical loads resulting from a violent melt-water interaction in the reactor pit may jeopardize the integrity of the containment. At the same time, the corium water interaction can be expected to disperse part of the molten material in the primary system. The distribution of this material would lead to the dispersal of heat sources throughout the RCS, possibly resulting in the failure of the RCS piping due to thermal stress and giving way to evaporation of the deposited fission products.

C.5 RPV Failure Modes/Direct Containment Heating

After relocation of the melt into the lower plenum, a bed of core melt and quenched debris will form at the bottom of the RPV. The maximum heat fluxes which can potentially lead to the failure of the RPV result from either the relocation process of the corium into the bottom of the RPV or from the thermodynamics within the relocated corium pool. In fact, due to the density differences there may be a separation of the metallic and oxidic constituents of the melt, so that the oxidic melt is covered by a metallic melt. Within the oxidic melt, convection will transfer the heat towards the lower head and the top of the pool. Based on the physics of the convective heat transfer within a hemispherical pool with internal heat sources, the highest heat flux occurs near the surface of the oxidic melt, which is consequently the most likely location of the failure of the RPV. This RPV failure mode is the most likely as the heat fluxes on the lower head wall are greater than those involved in the relocation process.

In the unlikely event that the plant sustains a high primary system pressure at the moment of RPV failure, the melt that escapes via the failed area may partly fragment resulting in small melt droplets that are transported with the stream flow from the reactor pit into adjacent compartments of the containment. The interaction of hot melt particles with the containment atmosphere results in an energy input into the containment primarily by the cooldown of the particles to the temperature of the containment atmosphere; however, the oxidation of metals and the combustion of hydrogen are also possible.

While some of the molten material will convert into droplets and heat the atmosphere of the containment, the majority will tend to accumulate on the surface located beneath the RPV. In the U.S. EPR™ design, numerous mechanisms act to limit the dispersal including the presence of obstructions in the gas flow path, the changes of direction/particle deflection, the expansion of the flow paths, and the stagnation of gas flow at corners or obstacles.

C.6 Corium Behavior in the Reactor Pit

The U.S. EPR™ design involves a provision for temporary melt retention and conditioning in the reactor cavity. This engineered design feature presupposes a depressurization of the RCS prior to the formation of a molten pool within the lower plenum of the RPV. After RPV failure the molten corium is intended to first accumulate in the reactor cavity and later relocate, in one event, into a lateral compartment. Spreading of the melt will be followed by flooding, quenching and sustained cooling of the corium.

As an ex-vessel severe accident mitigation strategy, the consequences of MCCI contribute to the transformation of the melt into a stable configuration. In this two-stage stabilization process, retention and spreading, MCCI is not only unavoidable; but, it is actually incorporated into the U.S. EPR™ solution for severe accident mitigation. The molten corium, composed of both metallic and oxidic material, falls into the reactor pit where it encounters a layer of sacrificial concrete situated above a layer of heat resistant ZrO_2 bricks. The sacrificial concrete is engineered to introduce material into the molten corium mixture conditioning the melt such that the spreadability of the melt improves (i.e., lower viscosity). Long exposure within the reactor pit adds more of this engineered concrete into the molten corium, thus optimizing the conditioning objective.

The rate at which the sacrificial concrete ablates and mixes with the molten corium is dependent on the amount of energy absorbed by the concrete. The thickness of sacrificial concrete is such that even with a large release, complete ablation of the concrete can take about two hours. For smaller pours, the ablation time is longer; thus, providing more time to accumulate material from the core. In the center of the reactor pit resides a melt plug made of the same sacrificial concrete lining the reactor cavity. Rather than being backed by the heat resistant ZrO_2 brick, there is an aluminum gate leading to a transfer channel. Once the sacrificial material has interacted with the corium, the corium reaches the metallic part of the melt plug, also called the melt gate. Failure of the melt plug and gate allows the molten corium to flow freely into a transfer channel leading to the core spreading room.

C.7 Corium Behavior in the Core Spreading Room

Molten corium/containment structure interaction can lead to penetration by the core debris of the containment basemat. The U.S. EPR™ design provides both passive and active cooling functions to the core spreading room to remove both the short-term sensible and long-term decay heat from the melt.

When in the core spreading room, the molten corium will activate the passive flooding valves that then allow water residing in the in-containment refueling water storage tank (IRWST) to flood the spreading room. The water enters the spreading room from the bottom and first cools the spreading room at the bottom and its sides and finally pours over onto the surface of the spread melt.

The addition of the water subsequently leads to quenching of the melt. Water is injected onto the molten corium in the spreading room at a slow enough rate to avoid an energetic steam explosion. Steam explosions that could potentially jeopardize the containment integrity are prevented by the addition of sacrificial concrete that results in a composition layer inversion that raise the oxidic melt above the metallic melt. Cooling results in a safe enclosure of the melt in its own crust.

The steam and heat that are produced during the cooling process are released into the containment atmosphere. The severe accident heat removal system (SAHRS) is designed with the capability to remove residual heat from the spread melt. The SAHRS is also designed to control the containment atmosphere during a severe accident. In the near term following a severe accident, the SAHRS system will be able to keep the containment pressure well below the design pressure.

The SAHRS performs its function in the short-term by spraying via the SAHRS spray line. Following the short-term phase the SAHRS can be operated in two modes. The first mode consists of the removal of decay heat from the melt by direct cooling of the melt via an overflow into the IRWST, while the second mode involves spraying in the containment for atmospheric heat removal.

After the spreading room is completely flooded with water, the molten corium is expected to form a solid mass within days. However, provided the SAHRS is operable, the long-term containment temperature and pressure should remain low.

C.8 References

C.1. U.S. NRC, Regulatory Guide 1.206, “Combined License Applications for Nuclear Power Plants,” March 2007.