Technical Challenges Associated with Shutdown Risk when Licensing Advanced Light Water Reactors

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Abstract: The United States (U.S.) Title 10 Code of Federal Regulations (CFR), 10 CFR 52.47(a)(27), requires applicants seeking a design certification (DC) to submit a description of the design specific probabilistic risk assessment (PRA) and its results. A DC applicant's final safety analysis report (FSAR) is expected to contain a qualitative description of PRA insights and uses, as well as some quantitative PRA results, such that the U.S. Nuclear Regulatory Commission (NRC) staff can perform the review and ensure risk insights were appropriately factored into the design. As referenced in the NRC Standard Review Plan (SRP) (NUREG-0800) Chapter 19 [1], the staff ensures the risk associated with the design compares favorably against the Commission's goals [2] of less than 1×10^{-4} per year (/yr) for core damage frequency (CDF) and less than 1×10^{-6}/yr for large release frequency (LRF). The staff expects that this PRA covers all modes of operation, including shutdown modes. The NRC has reviewed or is in the process of reviewing shutdown risk for evolutionary reactors and advanced passive reactors. The NRC is also preparing to review shutdown risk for small modular reactors (i.e., as part of pre-application activities). At the time the PRA information is submitted to the NRC, detailed shutdown procedures and outage plans have not been developed. Additionally, a low power and shutdown (LPSD) PRA standard has not been formally issued for general use. Therefore, reviews of plant configurations during shutdown combined with the impact of temporary equipment, penetrations, and a potentially open containment; evaluations of new design features; LPSD PRA scope issues have presented several challenges that will be discussed in this paper.

Keywords: PRA, Low power, Shutdown Operation, Midloop Operation

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INTRODUCTION

As discussed in SRP Section 19.0, Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors, the applicant's PRA and severe accident evaluation are used to do the following:

(1) Identify and address potential design feature and plant operational vulnerabilities, where a small number of failures could lead to core damage, containment failure or large release;
(2) Reduce or eliminate significant risk contributors of existing operating plants;
(3) Identify the design's robustness and tolerance of severe accidents form internal or external events and levels of defense in depth;
(4) Identify the risk significance of potential human errors; and,
(5) Demonstrate how the risk with the design compares against the Commission's Goals.

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The PRA results and insights are used to support design and operational programs such as; Inspection, Tests, Analyses, and Acceptance Criteria (ITAAC); Reliability Assurance Program; Technical Specifications (TSs); and Combined License (COL) applicant action items and interface requirements.

As discussed in the SRP Section 19.0, the scope of a DC review is limited to design specific aspects within the scope of the certification. The design specific PRA developed during the DC stage may not identify site-specific information (e.g., seismic hazards, switchyard and offsite grid configuration) and may not explicitly model all aspects of the design (e.g., balance of plant). Therefore, the applicant's design specific PRA may include assumptions regarding site parameters and the interface with undeveloped aspects of the design. If a COL applicant references a standard DC, these issues are evaluated when a COL applicant submits their PRA information in accordance with 10CFR52.79(d)(1) which states, " …the plant-specific PRA information must use the PRA information for the design certification and must be updated to account for site-specific design information and any design changes or departures."

When the LPSD PRA information is initially submitted with the DC document (DCD), details regarding outage plans and equipment maintenance are not developed. Typically, the LPSD PRA is built considering a "typical" refueling outage to identify shutdown Plant Operational States (POSs) that must be evaluated. POSs, as defined in Regulatory Guide 1.200 [3], "are used to subdivide the plant operating cycle into unique states, such that the plant response can be assumed to be the same within the given POS for a given initiating event. Operational characteristics (such as reactor power level; in-vessel temperature, pressure, and coolant level; equipment operability ; and changes in decay heat load or plant conditions that allow new success criteria or reactor coolant system or containment configuration) are examined to identify those relevant to defining POSs."

Based on reviewing shutdown events through the NRC reactor oversight process, the staff recognizes that POSs during forced outages may take place with a decay heat level higher than a refueling outage, in anticipation of returning to full power. The staff has to consider POSs during shutdown and refueling operations given the highest possible decay heat levels that could occur.

NRC STAFF REVIEW APPROACH

As the PRA staff evaluate the applicant's POS definitions, they work with other technical staff (e.g., reactor systems) to review all the relevant design sections of the final safety analysis report (FSAR) to understand (1) how the reactor coolant system (RCS) will respond to a loss of the decay heat removal (DHR) function during each POS and (2) how the RCS will respond to a loss of inventory during each POS. Additional focus is placed on what happens to the RCS during a postulated re-pressurization for each POS. For each design application, the NRC staff seeks the following information:

1. An understanding of the piping and circulation paths of the residual heat removal (RHR) system, particularly entry points into and exiting the reactor vessel. The staff considers the potential for flow bypass of the reactor core and any necessary vessel level limits to prevent thermal stratification within the core given an extended loss of DHR.
2. An understanding of the RHR system instrumentation and RCS level indication. The staff needs information regarding where these instruments are tapped into the RCS. This information is necessary to evaluate the validity of this indication to assess RCS conditions when the DHR system is not functioning and during a postulated RCS re-pressurization.
3. An understanding of how the RCS is dismantled during the refueling process. The staff needs to understand the size, the location, and the relative elevation of each opening in the RCS. This information is necessary to evaluate: (1) the differential pressure capability of temporary penetrations, (2) the feasibility of pumped injection, (3) the feasibility of gravity feed, and (4) the

feasibility of passive heat exchangers such as steam generators (SGs) and isolation condensers, should the RCS re-pressurize following an extended loss of DHR.

4. The feasibility of alternate decay heat removal paths during cold shutdown and refueling such as passive heat exchangers (Isolation Condensers, Steam Generators, etc.) when the RCS is intact and feed and bleed/steam strategies when the RCS is not intact.
5. A brief outline of the core alteration process.
6. An understanding of routine maintenance to be performed during outages (e.g., SG tube inspections, inspection and repair of reactor coolant pumps, inspection of screens, repair of the first isolation valves of the RCS, etc.)
7. The process by which containment is accessed for outage preparation (e.g. containment breach, de-inerted, etc.)
8. An understanding of the systems needed to keep the containment intact following a severe accident during cold shutdown and refueling (whether containment pressure suppression or hydrogen controls are necessary)

The staff will ask the applicant for the following information, if the information is not covered in the DCD:

1. Does the RCS need to be drained from full power operation levels during a refueling outage (SG tube inspections, head removal, etc.)?
2. How is vessel head removal performed and what equipment needs to be disconnected (i.e., instrumentation such as core exit thermocouples)?
3. How is fuel being moved to the spent fuel pool (e.g., is fuel being moved someplace other than the spent fuel pool such as a temporary fuel rack on the refueling canal/cavity wall)?
4. What RCS pipes exist below the hot legs and cold legs in pressurized water reactors (PWRs) and below the top of active fuel in boiling water reactors (BWRs), which if breached, could lead to an inadvertent drain down that could lead to core uncovery and would bypass the normal piping system performing the DHR function?
5. Where does the RCS level instrumentation piping tap into the RCS? These locations are important in understanding if these indications will accurately reflect actual RCS level given a postulated RCS re-pressurization following an extended loss of DHR. Under what conditions is the RCS level instrumentation calibrated during shutdown, i.e., if calibrated for cold conditions, will the instrumentation appropriately reflect the actual water level if the RCS returns to hot, pressurized conditions due to an event?
6. How are core exit temperatures monitored during the outage?
7. Are there components connected to the RCS or the reactor vessel that could lead to RCS drain down and core uncovery if single or multiple operator errors occurred during maintenance (such as work on control rod drive pumps, or the reactor water cleanup (RWCU) draining line) that would bypass the residual heat removal (RHR) system?
8. What is the time to RCS boiling and time to core damage for each POS given an extended loss of DHR and no operator action?

The staff also reviews the Technical Specifications (TSs) to identify the mitigative systems and automatic system actuations and indications that are required to be operable. The staff acknowledges that equipment and indication not required by TS may be out of service. It will depend on the COL licensee's operational needs. Additionally, the staff reviews the DC and COL applications using NRC shutdown guidance such as NRC Generic Letter 88-17[4] and NRC and industry shutdown event reports such as NUREG-1410[5] to check whether potential significant risk contributors are omitted in the applicant's PRA information.

After the above information is provided, then the staff reviews the completeness of the POSs, initiating events, and event trees. Based on past reviews, after the staff asks the above questions, POSs, initiating events, and event tree top events are added and/or redefined by the applicant. The staff does not focus on any numerical results until a majority of the above information is received from the applicant.

Typically, an applicant's shutdown PRA is defined as when the RHR function is initiated (e.g., hot shutdown). At temperatures and pressures above the RHR entry conditions, the plant response to a loss of core cooling and/or a loss of inventory is similar to the full power PRA. TS coverage of mitigating systems and actuation signals is generally comparable to the full power analysis with a few notable exceptions. The reactor is assumed to be sub-critical and decay heat is lower, which helps reduce risk, although some automated functions are removed. In addition, the staff evaluates whether containment integrity and containment systems are required to be operable. For example, for BWRs, a de-inerted containment is often not required to be operable below certain power levels, which impacts severe accident hydrogen control and large release frequency (LRF). These exceptions, as applicable, are handled through sensitivity studies using the full power PRA model to see if there are additional risk insights.

Design certification applicant's evolutionary and advanced passive PWR PRAs typically start in the hot shutdown mode with the plant cooling provided by RHR in preparation for refueling. The PRA will model entry into cold shutdown with an intact RCS and RCS temperature less than 200 °F. PWR PRAs will then model draining the RCS in anticipation of isolating the steam generators from the RCS using SG nozzle dams to perform SG tube inspections. For both BWRs and PWRs, the shutdown PRA will model vessel head removal and refilling of the RCS and the refueling cavity to reach the necessary RCS level for core alterations. The staff typically asks DC applicants to evaluate plant conditions when the refueling cavity is flooded. Although given a flooded cavity, the time to core damage following a loss of DHR often extends well beyond 24 hours. In this case, the staff is concerned about drain down events which could shorten the time to core damage. The staff will also evaluate the use of temporary fuel racks in the refueling cavity. Use of these racks could shorten the time to fuel uncovery as compared to the time to uncover with fuel in the vessel given an inadvertent drain down path.

Based on staff review of current operating PWR LPSD PRAs and staff review of evolutionary and advanced passive PWR PRAs, roughly 80 percent of the LPSD risk occurs during drained RCS conditions. This can be caused by: (1) reduced times to RCS boiling and core damage given a loss of the DHR function, (2) the increased reliance on operator actions (although some applicants have automated mitigative actions which significantly reduces the risk), (3) the lack of diverse mitigation paths to prevent core damage, particularly when the RCS is breached, (4) and the lack of mitigating equipment required to be operable according to TSs. An extended loss of DHR, RCS boiling and subsequent RCS re-pressurization can lead to unanticipated RCS system responses and instrumentation issues that are often not initially modeled in the applicant's PRA.

PWR reduced inventory operation receives a significant amount of staff review and has involved many technical challenges due to lack of application understanding of NRC expectations. For example, often the staff has questioned how the RCS is to be drained. If the RCS is drained with the RCS closed, the staff reviews the use of air or nitrogen injection into the RCS to speed the draining process and the risk of any RCS vacuum conditions that may develop. Then, the staff reviews the RCS plant response given an extended loss of the decay heat removal function and RCS pressurization. If SG cooling is to be credited, the staff will need assurance through confirmatory calculations that SG cooling will remain sustainable given potential nitrogen and/or air inventory.

Pressurizer Surge Line Issues

If the PWR applicant plans to drain the RCS using a vented pressurizer with an open pressurizer manway or pressurizer safety valves, the staff evaluates the associated RCS system response given a postulated loss of DHR and subsequent RCS pressurization. The staff identifies the size and elevation of each RCS vent with respect to the pressurizer. For POSs early in the outage with high decay heat and a high elevation vent in the pressurizer, a loss of decay heat removal and subsequent RCS boiling in the reactor vessel can result in high steam generation rates and high steam velocities in the pressurizer surge line. This phenomena is often referred to as surgeline flooding. Surgeline flooding combined with head losses in the pressurizer can result in hot leg water inventory filling the pressurizer. The increased pressurizer level can negate the elevation head necessary for gravity injection. This phenomena is documented in EGG-EAST-9337, Revision 1, February 1991, "Thermal-Hydraulic Processes Involved in Loss of Residual Heat Removal During Reduced Inventory Operation." [6]. Surge line flooding has caused PWR DC applicants to revise their success criteria regarding gravity injection by either adding a low elevation vent in the hot leg or SG, or removing credit in the PRA for gravity injection when RCS is vented via the pressurizer.

As discussed in the enclosures to GL 88-17, surge line flooding is an example of shutdown phenomena which can cause instrumentation inaccuracies. With the RCS vented via the pressurizer, after surgeline flooding has initiated, the water level in the pressurizer can read higher than the water level in the reactor. Any instrumentation using upper level taps in the pressurizer may not appropriately reflect water available to cool the core. If this condition can exist, the staff will ask the applicant questions regarding other sources of indication available to the operators such as the core exit thermocouples and how the human reliability analysis (HRA) addresses this issue.

Over Draining Of RCS

The potential for vortexing and air binding of the RHR pumps during PWR reduced inventory operation has presented another technical challenge for new reactor reviews. Based on staff review of operating PWR PRAs and reports, overdraining (e.g., failure to stop the drain down process, causing the operating RHR pump(s) to loose suction) of the RCS to achieve midloop conditions is a significant contributor to operating PWR risk. Often, the DC applicant has used over drain frequencies that are significantly reduced compared to operating data. These reductions occur from the use of new features such as automated closure of RCS letdown valves and automated tripping of the RHR pumps on low hot leg level. Typically, the frequency of overdraining the RCS to achieve midloop with resulting loss of the DHR function includes the failure of automated features and the failure of the operator to terminate the overdraining manually. The PRA staff works with other technical reviewers to evaluate the adequacy of the set points for automated features, given that vortexing is a function of hot leg level and RHR pump flow rate. The staff reviews any applicable test results that support the setpoints for the automated features. Once the set points for automated features have been established, the PRA staff uses this information to check the time available for the operator to terminate an overdrain event and how this operator error was modeled in the shutdown PRA, especially if RHR system recovery is credited in the event trees.

Gravity Feed and Passive Cooling

For both passive PWR and BWR designs, credit for gravity injection during LPSD has represented another technical challenge that may not be sufficiently addressed in the applicant's PRA. The staff has carefully evaluated gravity feed by considering a possible RCS re-pressurization, given a loss of DHR. Water entrainment and/or RCS re-pressurization could negate the gravity head necessary for gravity injection. Re-analysis of these issues using thermal-hydraulic (T/H) modeling and tests have resulted in increasing RCS vent path sizes and requiring these vents to be operable by TS. The staff's intent is to eliminate the operator failing to vent the RCS sufficiently for gravity feed to work during a postulated loss of RHR. This operator action is important as gravity feed represents a significant source of risk reduction for passive designs.

For advanced BWR and PWR PRAs, during cold shutdown with an intact RCS, use of passive heat removal features such as isolation condensers and steam generators are often credited in the PRA as a potential heat removal path. These alternate heat removal paths represent a significant source of risk reduction. The PRA staff works with technical staff to identify issues unique to cold shutdown that could impact passive cooling such as non-condensable gases or steam condensation, as the RCS heats up and re-pressurizes.

PWR Nozzle Dam Issues

The installation of low differential pressure capability temporary penetrations such as SG nozzle dams[2] represents another technical challenge. As discussed in detail in NRC Generic Letter 88-17, Enclosure 2, Section 2.1.1, "Pressurization" [3] and NRC Information Notice (IN) 88-36[7], inappropriate use of SG nozzle dams can result in complete core voiding in 15 or 20 minutes. As stated in GL 88-17, Section 2.1.1, cold leg openings can allow water to be ejected from the vessel following a loss of DHR until sufficient water is lost that steam is relieved by clearing of the crossover pipes. As further discussed in GL 88-17, the impact of a postulated re-pressurization of the RCS and/or safety injection on SG nozzle dams following an extended loss of DHR is also evaluated by the staff. Before the staff can accept the numerical results of the PRA, the staff has to ensure that the guidance in GL 88-17 is being implemented. To avoid COL holder issues with SG manway and nozzle dam installation and removal, the staff requests that the order of nozzle dam installation and removal is specified in the risk insights table in Section 19 and Section 5.4.7 of the FSAR, based on the wording similar to IN 88-36.

The staff then evaluates the adequacy of the RCS vent path when the nozzle dams are installed, and the reactor vessel head is in place. The staff has to ensure that the RCS vent path is large enough to prevent RCS re-pressurization that could fail the nozzle dams. Typically, the differential pressure capability of nozzle dams range from 30 to 50 psi. Due to staff concerns about the adequacy of thermo hydraulic codes to model RCS pressure when the RCS is vented via the pressurizer and the reactor vessel head is installed (surgeline flooding), the staff will have the applicant assess the capability of the nozzle dams to withstand: (1) feed and bleed from safety injection and (2) a fully entrained pressurizer.

The applicant must resolve these thermo hydraulic issues before the event tree and human actions review can be completed by the staff. Omission could potentially invalidate: the success criteria, event tree construction and human error probability (HEP) estimates. Although most applicant's use of HRA methods has been acceptable, often the consideration of the thermo hydraulic issues discussed above and their impact on instrumentation and operator timing has not been factored adequately into the HEP analysis. The staff uses sensitivity studies and risk importance analyses to identify key sources of uncertainty and improvements to TS, ITAACs, Reliability Assurance Program, and Regulatory Treatment of Non Safety Systems (RTNSS).

Containment Closure and Hydrogen Control

Issues impacting LPSD LRF and Level 2 LPSD PRA represent additional technical challenges with DC applicants. As noted in Regulatory Guide (RG) 1.200 and GL 88-17, the capacity of the containment to remain intact following a severe accident is dependent on: (1) severe accident containment performance,

[2] A steam generator nozzle dam is a mechanical "dam" that is installed in the steam generator nozzle areas to allow testing and repairs to the steam generator tubes while RCS level is raised to a level to permit refueling or other work on the reactor internals.

(2) the differential pressure capability of any temporary penetrations and (3) the feasibility of operators to close containment if initially opened before adverse environmental conditions (e.g., temperature, radiation, humidity, noise, etc.) prevent its closure. For both BWRs and PWRs, the staff evaluates the status of containment during hot standby, hot shutdown, cold shutdown and refueling (e.g., open, closed, de-inerted) and the status of containment systems. The staff also checks TSs to see what is required to be operable. The PRA staff uses GL 88-17 for its interpretation of containment closure. Specifically, it states, "containment closure is defined as a containment condition where at least one integral barrier to the release of radioactive material is provided." As stated in GL 88-17, "reliable assurance of containment closure should include consideration of activities which must be conducted in a harsh environment. For example, once boiling initiates in the RCS, a large volume of steam may be entering containment potentially leading to high containment temperature and increased pressure."

For BWR designs that rely on inerting the containment to control hydrogen combustion, preparations for containment entry involving de-inerting result in a loss of the ability to control hydrogen following a severe accident. For these designs, CDF is assumed to be equal to LRF for comparison with the Commission Goals, since containment is not expected to remain intact following a severe accident where significant hydrogen generation is assumed. This assumption is based on NUREG/CR-6595, Revision 1, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," October 2004.[8]

For PWRs, the NRC staff determines how the applicant will meet the recommended actions described in GL 88-17 regarding containment closure before steaming occurs inside containment. The staff also questions whether any temporary penetrations are used to define containment closure and identifies the structures, systems, and components (SSCs) needed to accomplish containment closure (e.g., a crane, offsite power, onsite power, etc.). The staff reviews the time to RCS core boiling given (1) an extended loss of DHR and (2) a loss of RCS inventory in each POS with an open RCS. This information is needed to determine the time available to close containment. Containment closure post RCS boiling is not credited as directed by GL 88-17. All of the PWR DC applicants have implemented TS to close containment before boiling when the RCS is below the RCS level needed for core alterations. TSs provides the assurance that procedures, equipment, and training will be implemented to meet the requirements.

In addition to containment closure, the NRC staff asks PWR applicants to evaluate whether containment closure requires hydrogen control for the containment to remain intact following a severe accident. The use of large capacity borated water tanks inside containment has introduced concerns about localized elevated hydrogen concentrations following a severe accident. With this design, applicant's often have to re-evaluate hydrogen control needs at shutdown.

Shutdown Fires

The DC applicant is also expected to perform an assessment of shutdown internal fires and internal floods. The Level 1 internal events shutdown PRA is used to develop a quantitative fire risk assessment. Often, the applicant's full-power fire assessment is completed according to the guidance in NUREG/CR–6850[9]. The guidance in NUREG/CR–6850 is not applicable to qualitative screening for shutdown conditions. Therefore, the applicants often perform the screening for the shutdown fire model assuming that the postulated fire results in one of the initiating events defined in the shutdown model. The critical safety functions essential to the shutdown model are DHR and inventory control. Power availability is modeled for its impact on DHR. Loss of power is evaluated as an initiating event, and the model includes power dependencies for systems.

The applicant must also evaluate initiators unique to shutdown such as fires that could lead to inadvertent draindown paths (e.g., fire induced failure of the RCS drain down path during reduced inventory operations). As in the full-power fire assessment, the applicant conservatively assumes that fires will propagate unmitigated in each fire area and damage all functions in the fire area. Fire suppression is not credited. During shutdown conditions, a fire barrier may not be intact because of maintenance activities. The shutdown fire analysis assumes that all barriers are intact, or a fire watch is added to increase the probability of fire detection and suppression. The staff has requested that DC applicants perform sensitivity studies identifying which fire barriers are particularly risk significant. The staff also requests that applicants evaluate fires in the control room and fires in containment.

Shutdown Floods

Similar to the full-power assessment, the applicant performs the shutdown internal flooding analysis using equipment locations based on existing plant layout drawings. The applicant then divides buildings into flood zones based on separation for flooding. The applicant screens those flood zones that do not contain flood sources or equipment identified in the PRA.

Depending on the building and the origin of the flood, the applicant considers the following SSCs for flood propagation: automatic flood detection systems, automatic systems to terminate flooding, watertight doors to prevent the progression of flooding, sump pumps, and other design or construction characteristics that minimize the consequences of flooding. The flood features that are found to be risk significant for the LPSD PRA are captured in the risk insights table with the appropriate reference to the deterministic portions of the FSAR.

Updates to COL PRA

As the above technical challenges are being evaluated, the PRA staff and the applicant keep track of risk important key LPSD features such as structures, systems, components, procedures, and instrumentation. The applicant documents these features in the FSAR Section 19 risk insights table with dispositions to all relevant sections of the FSAR (e.g., Technical Specifications, Reactor Coolant System Design, Instrumentation and Controls). As stated in 10CFR52.79(d)(1), for a COL applicant that references a standard design certification, "the plant-specific PRA information must use the PRA information for the design certification and must be updated to account for site-specific design information and any design charges or departures." A review of the differences between the as-built plant, the basis for the DC PRA and the risk insights table will be completed prior to fuel load to update the PRA when used for regulatory applications.

CONCLUSION

At the time the PRA information is submitted to the NRC, the design certification applicants have not yet developed detailed shutdown operating procedures and outage plans. Therefore, the staff is challenged to evaluate and consider inadvertent adverse plant configurations that may occur during shutdown, combined with the impact of temporary equipment, penetrations, and a potentially open containment. In this context, the staff must also evaluate new design features. Before the staff evaluates the applicant's POS definitions and LPSD event trees, the PRA staff works with other technical staff to review all the design sections of the FSAR to gain an understanding of how the RCS responds to a loss of the DHR function and a loss of inventory in each POS. Additional focus is placed on what happens to the RCS during a postulated re-pressurization for each POS. The staff reviews information from operating event reports and generic communications to ensure that previous shutdown issues have been addressed in the new design. This process is iterative and often results in the modification of POS definitions and events trees. During the review, the staff ensures that risk important structures, systems, components, instruments, and procedures are documented in the applicant's risk insights table. Each item in the risk insights table has a disposition/reference to all relevant sections of the FSAR (i.e., the engineering chapters relating to Technical Specifications, Reactor Coolant System Design, Instrumentation and Controls, etc.) The COL licensee that references the design certification is required to review the differences between the as-built plant and the DCD design used as the basis for the PRA and the risk insights table.

Finally, many risk insights discussed in this paper relating to (1) how the RCS responds to a loss of the DHR function and a loss of inventory and (2) the feasibility for containment closure and hydrogen control to achieve an intact containment are equally relevant to operating reactors. These same risk insights are incorporated in the NRC risk informed regulatory applications such as the Reactor Oversight Process.

References:

[1] NUREG-0800, Standard Review Plan, Section 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors", Revision 2, June 2007.
[2] Staff Requirement Memorandum to SECY 90-016, "SECY-90-16 –Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationships to Current Regulatory Requirements," U.S. Nuclear Regulatory Commission, Washington, DC, June 26, 1990.
[3] Regulatory Guide 1.200, "An Approach For Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009.
[4] Generic Letter 88-17, "Loss of Decay Heat Removal," U.S. Nuclear Regulatory Commission, Washington, DC, October 17, 1988.
[5] NUREG-1410, "Loss of Vital AC Power and the Residual Heat Removal System during Mid-loop Operations at Vogtle Unit1on March 20, 1990," June 1990.
[6]Idaho National Engineering Library Report EGG-EAST-9337, Revision 1, February 1991. "Thermal-Hydraulic Processes Involved in Loss of Residual Heat Removal During Reduced Inventory Operation," Prepared for the U.S. Nuclear Regulatory Commission. Available from DOE Information Bridge: http://www.osti.gov/bridge/ (search for title).
[7]Information Notice, 88-36," Possible Sudden Loss Of RCS Inventory During Low Coolant Level Operation," U.S. Nuclear Regulatory Commission, Washington, DC, June 8, 1988.
[8]NUREG/CR-6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," October, 2004.
[9] NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities" Volume 1: Summary & Overview," September 2005.