Safety Analysis Report Acceptance Review Guide for a Design Certification (DC) and Standard Design Approval (SDA) Applications

Background Information

This Review Guide is intended to be used by the technical branches to perform their acceptance reviews of design certification (DC) and standard design approval (SDA) applications. The findings of the acceptance review will be provided to the Lead Project Manager (PM) so they can evaluate completeness, technical sufficiency and the impacts of the technical issues identified during this review on the schedule. This review guide contains Table 1, "Safety Analysis Report Acceptance Review Results Table," which is organized by review area(s)/topic(s) within assigned safety evaluation report (SER) sections.

Prior to the acceptance review, the project managers and technical reviewers should be familiar with:

- The anticipated scope of review of the DC or SDA application including the following:
    - Assigned DC or SDA application section(s) and relevant supplemental information (e.g., Technical Reports), Design Acceptance Criteria (DAC) and/or Inspections, Test, Analysis, and Acceptance Criteria (ITAAC).
    - Applicability of RG 1.206
        - Cognizant of complete COL application contents
        - Cognizant of COL applicant responsibility when referencing a DC or SDA (site characteristics, site-specific design information, operational programs)
    - Corresponding section(s) of the standard review plan (SRP).
    - The List of safety analysis report (SAR) Review Areas Potentially Involving More Detailed Review in ADAMS (ADAMS Accession No. ML072430683). This list does not replace the SRP, but rather represents a list of review areas contained within the SRP that may potentially involve more detailed technical review (e.g., involve computer code evaluation, detailed data analysis, new safety feature, or emerging operating experience).
    - The design-centered-review approach (DCRA),[1] such that staff decisions made on the "reference COL" would apply to all "subsequent COLs."
    - Concurrent reviews (e.g., other DC application reviews, COL application referencing the DC review or related topical report reviews).
    - Available risk insights applicable to DC application sections under review.

The following directions should be used by PMs and technical staff in performing the acceptance review. Table 1 should be used to document the acceptance review effort. The information in this table may be used to evaluate the acceptability of the COL application for docketing as discussed in Office Instruction LIC-117. Each branch may choose to make entries for each review area or SRP section in Table 1. Alternatively, a branch may choose to enter information only for those technical areas that are found to be incomplete or not technically sufficient or those areas that will require changes to resource planning assumptions (+ or -).

[1]Additional information on DCRA is provided in SECY-06-0019, dated January 31, 2006.

I. Completeness Review: Verify that the DC or SDA application contains all of the information required by the applicable regulations for your assigned review(s) as discussed in LIC-117.

A. Document the review area(s)/topic(s) in Column 1 (List all review topics or only those found to be not complete, not technically sufficient, or requiring changes to planning assumptions).

B. Determine whether the applicant has addressed the applicable regulations for the assigned review area. (s*ee Enclosure 7 list for 10 CFR 52.47, "Contents of applications, technical information,")* [yes/no in Column 2].

10 CFR 52.47 and 10 CFR 52.137 identifies prescriptive requirements for the contents of DC and SDA applications, respectively, as well as cross-cutting requirements. For the following cross-cutting requirements, determine if any apply to your review section(s). For those in your review area, determine if the applicant addressed the proper items. The applicant's compliance with these requirements should be provided in Chapter 1 of the FSAR:

1. Three Mile Island (TMI) requirements [10 CFR 52.47(a)(8)];
2. Proposed technical resolutions of unresolved safety issues and medium-and high-priority generic safety issues [10 CFR 52.47(a)(21)];
3. Introduction of new safety features [10 CFR 52.47(c)(2)];
4. Operating experience insights incorporated into the plant design [10 CFR 52.47(a)(22)];
5. Conformance with SRP [10 CFR 52.47(a)(9)]; and
6. A description and analysis of design features for the prevention and mitigation of severe accidents [10 CFR 52.47(a)(23)].

Additional cross-cutting issues specifically related to a DC or SDA application include:

The DC or SDA applicant will address interface requirements for those design features that are outside the scope of the certified design as identified by the applicant; a representative conceptual design for those portions of the plant for which the application does not seek certification; and justification that the interface requirements can be verified with the inspections, tests, or analyses and that the method for verification is included in the proposed ITAAC [52.47(a)(24), (25), and (26)].

The DC or SDA applicant should also address applicable licensing and policy issues developed by the U.S. Nuclear Regulatory Commission (NRC) and documented in SECY-93-087, dated April 2, 1993 (ADAMS Accession No. ML083370250) and the associated SRM for advanced and evolutionary light-water reactor designs [per guidance provided in SRP Chapter 1].

For each review area/topic not addressed, summarize deficiency in Column 5, and promptly notify management of the projects branch.

II. Technical Sufficiency Review: Identify significant technical deficiencies in the DC or SDA application associated with your assigned review using the attached table and the following guidelines. The information contained in the various parts of the DC or SDA application that are discussed above in the Background Information should also be considered. A technical deficiency is defined as missing, improper, inadequate, or incorrect technical information needed by the NRC staff to conduct the assigned review. A significant technical deficiency is missing information that results in the staff being unable to conduct its review of the application against the acceptance criteria in the SRP or conduct its review within a predictable timeframe. If a significant technical deficiency is identified, the application should not be docketed unless it is able to be addressed through RSIs. Minor technical deficiencies, by contrast should be able to be addressed with a reasonable round of RAIs and without notably impacting the length of the review (i.e., the applicant indicated that the information is available, but was not included as part of the application).

Additional consideration should be given to any review areas/topics contained in the Safety Analysis Report Acceptance Review List that could require more extensive review time than is reflected in the generic review schedule.

As noted in LIC-117, risk insights may be available during the acceptance review. If so, these insights should be used to help determine the scope of the technical sufficiency review. If a review area/topic is associated with to a risk-significant SSC, indicate a yes in Column 6, in the attached table.

For the determined scope of technical sufficiency review:

A. Document additional review areas/topics in Column 1, as needed
B. Determine whether the DC or SDA application section(s) is(are) sufficient to conduct the detailed technical review for the review areas/topics identified in Column 1 [yes/no in Column 3]
C. Determine whether the review areas/topics identified in Column 1 can be resolved through the RAI process. Discuss with management whether to categorize the deficiency as “significant.” [yes/no in Column 4]
D. Document the technical deficiency(ies) that could prevent you from conducting your detailed technical review in Column 5. Describe the basis(es) for the deficiencies. These review area/topics may involve a significant amount of time to address (e.g., development of computer codes or first-of-a-kind testing), so estimate how this could impact the overall review schedule for your DC application section.
E. Notify the Lead PM of significant deficiencies as soon as they are identified.
F. Determine whether the identified technical deficiency is related to a risk-significant SSC [yes/no in Column 6].

III. Changes to Planning Assumptions:

Augmenting Planned Resources (Staff-Hours):

Re-evaluate the total review time that will be needed to conduct a review based on the significant technical deficiencies or new, unplanned review items documented in the Table 1. The acceptance review allows the reviewer to identify potential changes from the generic review schedule and estimated staff-hours so that adjustment can be

made to the staff-hours. The following characteristics of a DC application may require additional review time:

A. Inclusion or reference to new safety features;
B. Alternative approaches to SRP acceptance criteria (including alternatives to regulatory guides);
C. Other miscellaneous review topics that have not been adequately represented within the baseline model.

Next, determine whether the review effort is reflected within the generic review schedule (e.g., the estimated staff-hours are sufficient to perform the review of the alternative to an SRP acceptance criteria?) [yes/no in Column 7]. For each "no" in table, identify any change to the staff-hour planning assumptions and provide a basis (e.g., "departure not addressed in generic review schedule") in Column 8. Identify the projected review time in staff-hours needed to address all of the applicable items above for your DC application section.

Reducing or Eliminating Planned Resources (Staff-Hours):

For some SRP sections, the applicant may have incorporated by reference a technical report that has previously been approved by staff. This could allow for a reduction in the level of effort for a particular review area as it would relate to ensuring applicability of the technical report and the balance of information within the scope of the review. The responsible branch then needs to determine how much to reduce the generic review schedule and answer "no" for Column 7 in Table 1. For each "no" in the table, identify the change to the staff-hour planning assumptions and provide a basis (e.g., "limited review not reflected in the generic review schedule") in Column 8. Identify the projected review time in staff-hours needed to address all of the applicable items above for your DC application section.

Review the generic review schedule and notify the branch chief and Lead PM if schedule changes are needed.

IV. Identification of Dependencies between Concurrent Reviews: Identify any known dependencies between concurrent reviews. These dependencies could be between a DC review and a concurrent COL review (e.g., EPR DCD review concurrent with the EPR RCOL review), or there could be dependencies between reference COLs and subsequent COLs. These dependencies could potentially result in changes to planning assumptions. For example, the staff-hours associated with the review of a topical or technical report may be captured separate from the baseline review schedule. Those hours should not be double counted in related reviews. However, this does not change the duration of the task.

A. Identify and document review dependencies in Table 1(Columns 10 and 11).

Table 1: Safety Analysis Report Acceptance Review Results for [Applicant Name] [Design Center Name] [Application Type][2]

SAR Section:_____ Technical Branch:_____(Primary/Secondary) Technical Reviewer: _____
Branch Chief:_____ SRP Section: _____ Date:_____

Are there any technical deficiencies, changes in planning assumptions, or dependencies on concurrent reviews? Yes/No, Identify specific review area/topic in table below.

1. Review Area/Topic*	Completeness and Technical Sufficiency Which Form Basis for Acceptability for Docketing					Changes to Planning Assumptions to be Considered in Development of Generic Review Schedule			Review Dependencies Among Concurrent Reviews	
	2. Does DCA section address the items required by regulation (refer to Enclosure 7? (Yes/No)	3. Is DC section technically sufficient for this review area/ topic? (yes/no)**	4. Can the technical deficiency be resolved through the RSI or RAI process within a predictable timeframe? (yes/no)*** RSI	RAI	5. If no, for either completeness or technical sufficiency, identify deficiency(ies). This information will be needed for technical review.	6. Is the identified technical deficiency related to a risk-significant SSC)? (yes/no)****	7. Are the generic review schedule and estimated staff-hours appropriate? (yes/no)	8. For each no, identify the change (+ or -) or basis for change.	9. Identify the total review time in staff-hours*****	10. Can the review of the area/topic be completed without the completion of a concurrent review? (yes/no)	11. For each no, identify which application (DCD or COLA) and section.

*Review Area/Topic: Item identified in RG 1.206, SRP or the regulations.

**Technical Sufficiency: The application is compared against the SRP acceptance criteria. Note: New safety features, alternate regulatory compliance approaches, should not be treated as deficiencies and factored into the basis for rejecting the application, unless staff determines that there is insufficient technical information associated with the respective item. These items are factored into confirmation of planning assumptions.

***Significant deficiencies are those review area/topic which impact the staff's ability to conduct the detailed technical review or complete its review within a predictable timeframe.

**** Division of Safety Systems & Risk Assessment will provide risk significance information at time of review, if available.

*****Identification of new review time is on a FSAR section basis and consistent with the review phases of the generic schedule. Changes from the generic review schedule and estimated hours should be on that basis.

[2] **NOTE: Branches may use this form to make entries for all review area/topics; or alternatively, to make entries only for those areas that have issues with completeness, sufficiency, or those that require changes (+ or -) to baseline estimated staff-hours.**

Table 1

Design Certification and Standard Design Application Acceptance Review Checklist

The DC application must include the following technical information required by 10 CFR 52.47:

Item	Information Required in DC Application 10 CFR 52.47(a)	DCD Section	Yes	No
1	The site parameters postulated for the design, and an analysis and evaluation of the design in terms of those site parameters	Ch. 2		
2	A description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefore, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished.	System-related chaps. and/or Ch. 15		
	It is expected that the standard plant will reflect through its design, construction, and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products. The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluations. Such items as the [ ] shall be discussed insofar as they are pertinent:			
	• Reactor Core	Ch. 4		
	• RCS	Ch. 5		
	• I&C Systems	Ch. 7		
	• Electrical Systems	Ch. 8		
	• Containment Systems	Sec. 6.2		
	• Other engineered safety features	Ch. 6		
	• Auxiliary Systems	Ch. 9		
	• Emergency Systems	Ch. 6		
	• Power Conversion Systems	Ch.10		
	• Radioactive Waste Handling Systems	Ch. 11		
	• Fuel Handling Systems	Sec 9.1		
	The following power reactor design characteristics will be taken into consideration by the Commission:			
	i) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials;	Ch. 1, 11, and 12		
	ii) The extent to which generally accepted engineering standards are applied to the design of the reactor;	Ch. 3		
	iii) The extent to which the reactor incorporates unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials;	Ch. 1		

Item	Information Required in DC Application 10 CFR 52.47(a)	DCD Section	Yes	No
	iv) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. In performing this assessment, an applicant shall assume a fission product release[3] from the core into the containment assuming that the facility is operated at the ultimate power level contemplated. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable postulated site parameters, including site meteorology, to evaluate the offsite radiological consequences. The evaluation must determine that; (A) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem[4] total effective dose equivalent (TEDE); (B) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem TEDE; [3] The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. These accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products. [4] A whole body dose of 25 rem has been stated to correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, its use is not intended to imply that this number constitutes an acceptable limit for an emergency dose to the public under accident conditions. This dose value has been set forth in this section as a reference value, which can be used in the evaluation of plant design features with respect to postulated reactor accidents, to assure that these designs provide assurance of low risk of public exposure to radiation, in the event of an accident.	Ch. 6		
3	The application contains the design of the facility, including:			
	i) the principle design criteria for the facility [see Enclosure 1 to this appendix for a tabulated list of Appendix A to 10 CFR Part 50] establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants in establishing principal design criteria for other types of nuclear power units	Sec. 3.1		
	ii) the design bases and their relation to the principal design criteria	Chps. 2–12 and 15		

Item	Information Required in DC Application 10 CFR 52.47(a)	DCD Section	Yes	No
	iii) information relative to materials of construction, arrangement, and dimensions, sufficient to provide reasonable assurance that the design will conform to the design bases with adequate margin for safety	Chps. 3–12		
4	An analysis and evaluation of the design and performance of structures, systems, and components with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.	Chps. 3–12 and 15		
5	Analysis and evaluation of emergency core cooling system (ECCS) cooling performance and the need for high-point vents following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §§ 50.46 and 50.46a of this chapter.	Secs. 5.4.12, 6.2, and 6.3		
6	The kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in part 20 of this chapter.	Chps. 11 and 12		
7	The information required by 10 CFR 20.1406	Chps. 11 and 12		
8	The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter;	Sec. 1.4		
9	The information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v); [see Enclosure 2 to this appendix for §50.34(f) requirements checklist]	Sec. 1.9, Ch. 19		
10	For applications for light-water cooled nuclear power plants, an evaluation of the standard plant design against the Standard Review Plan (SRP) revision in effect 6 months before the docket date of the application. The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for the design and those corresponding features, techniques, and measures given in the SRP acceptance criteria. Where a difference exists, the evaluation shall discuss how the proposed alternative provides an acceptable method of complying with the Commission's regulations, or portions thereof, that underlie the corresponding SRP acceptance criteria. The SRP is not a substitute for the regulations, and compliance is not a requirement.	Sec. 1.9**		
11	The information with respect to the design of equipment to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations described in 10 CFR 50.34a(e);	Ch. 11		

Item	Information Required in DC Application 10 CFR 52.47(a)	DCD Section	Yes	No
12	Proposed TS prepared in accordance with the requirements of 10 CFR 50.36 and 10 CFR 50.36a	Chps. 16 &11		
13	An analysis and description of the equipment and systems for combustible gas control as required by 10 CFR 50.44;	Sec. 6.2.5		
14	The list of electric equipment important to safety that is required by 10 CFR 50.49(d);	Sec. 3.11, Ch. 8		
15	A description of protection provided against pressurized thermal shock events, including projected values of the reference temperature for reactor vessel beltline materials as defined in 10 CFR 50.60 and 50.61;	Sec. 5.3.2		
16	Information demonstrating how the applicant will comply with requirements for reduction of risk from anticipated transients without scram events in § 50.62;	Secs. 4.2 and 15.8		
17	A coping analysis, and any design features necessary to address station blackout, as required by 10 CFR 50.63	Sec. 8.4		
18	Information demonstrating how the applicant will comply with requirements for criticality accidents in § 50.68(b)(2)–(b)(4);	Sec. 9.1		
19	A description and analysis of the fire protection design features for the standard plant necessary to comply with 10 CFR part 50, appendix A, GDC 3, and § 50.48 of this chapter;	Sec. 9.5		
20	A description of the quality assurance program applied to the design of the structures, systems, and components of the facility. Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," sets forth the requirements for quality assurance programs for nuclear power plants. The description of the quality assurance program for a nuclear power plant shall include a discussion of how the applicable requirements of appendix B to 10 CFR part 50 were satisfied;	Ch. 17		
21	The information necessary to demonstrate that the standard plant complies with the earthquake engineering criteria in 10 CFR part 50, appendix S;	Sec. 3.7		
22	Proposed technical resolutions of those Unresolved Safety Issues and medium- and high-priority generic safety issues which are identified in the version of NUREG–0933 current on the date up to 6 months before the docket date of the application and which are technically relevant to the design;	Sec. 1.9		
23	The information necessary to demonstrate how operating experience insights have been incorporated into the plant design;	Sec. 1.9**		
24	For light-water reactor designs, a description and analysis of design features for the prevention and mitigation of severe accidents, (e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass);	Ch. 19		

Item	Information Required in DC Application 10 CFR 52.47(a)	DCD Section	Yes	No
25	A representative conceptual design for those portions of the plant for which the application does not seek certification, to aid the NRC in its review of the FSAR and to permit assessment of the adequacy of the interface requirements in paragraph (a)(25) of this section;	As applicable		
26	The interface requirements to be met by those portions of the plant for which the application does not seek certification. These requirements must be sufficiently detailed to allow completion of the FSAR;	As applicable		
27	Justification that compliance with the interface requirements of paragraph (a)(25) of this section is verifiable through inspections, tests, or analyses. The method to be used for verification of interface requirements must be included as part of the proposed ITAAC required by paragraph (b)(1) of this section;	As applicable		
28	A description of the design-specific probabilistic risk assessment (PRA) and its results.	Ch. 19		

Item	Information Required in DC Application 10 CFR 52.47(b)	DCD Section	Yes	No
1	The proposed inspections, tests, analyses, and acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the Act, and the Commission's rules and regulations;	Sec. 14.3		
2	An environmental report as required by 10 CFR 51.55.	Ch. 19		

Item	Information Required in DC Application 10 CFR 52.47(c), as applicable to particular applications:	DCD Section	Yes	No
1	An application for certification of a nuclear power reactor design that is an evolutionary change from light-water reactor designs of plants that have been licensed and in commercial operation before April 18, 1989, must provide an essentially complete nuclear power plant design except for site-specific elements such as the service water intake structure and the ultimate heat sink;			
2	An application for certification of a nuclear power reactor design that differs significantly from the light-water reactor designs described in paragraph (c)(1) of this section or uses simplified, inherent, passive, or other innovative means to accomplish its safety functions must provide an essentially complete nuclear power reactor design except for site-specific elements such as the service water intake structure and the ultimate heat sink, and must meet the requirements of 10 CFR 50.43(e);	Ch. 1		
3	An application for certification of a modular nuclear power reactor design must describe and analyze the possible operating configurations of the reactor modules with common systems, interface requirements, and system interactions. The final safety analysis must also account for differences among the configurations, including any restrictions that will be necessary during the construction and startup of a given module to ensure the safe operation of any module already operating.			

Administrative Requirements

The DC application meets the following administrative requirements:

Item	Requirements	Yes	No
52.45	The application must comply with the applicable filing requirements of §§ 52.3 and §§ 2.811 through 2.819 of this chapter.		
52.46	The application must contain all of the information required by 10 CFR 50.33(a) through (c) and (j).		
50.33	(a) Name of applicant		
50.33	(b) Address of applicant		
50.33	(c) Description of business or occupation of applicant		
50.33	(j) If the application contains Restricted Data or other defense information, it shall be prepared in such manner that all Restricted Data and other defense information are separated from the unclassified information.		

10 CFR 50.34(f), "Additional TMI-Related Requirements" Checklist

The application contains the information with respect to compliance with technically relevant positions of the TMI requirements in 10 CFR 50.34(f), with the exception of the combustible gas control requirements of §50.34(f)(1)(xii), (f)(2)(ix), and (f)(3)(v), which have been superceded by 10 CFR 50.44.

50.34(f) Item	Requirement	Action Plan Item*	N/A	Yes	No
(1) To satisfy the following requirements, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of these studies are factored into the final design of the facility. For licensees identified in the introduction to paragraph (f) of this section, all studies shall be completed no later than 2 years following issuance of the construction permit or manufacturing license. For all other applicants, the studies must be submitted as part of the FSAR.					
(1)(i)	Perform a plant/site-specific PRA, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant.	II.B.8			
(1)(ii)	Perform an evaluation of the proposed auxiliary feedwater system (AFWS), to include (PWRs only):	II.E.1.1			
	(A) A simplified AFWS reliability analysis using event tree and fault-tree logic techniques				
	(B) A design review of AFWS				
	(C) An evaluation of AFWS flow design bases and criteria				
(1)(iii)	Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA with LOOP. If damage cannot be precluded, provide an analysis of the limiting small-break LOCA with subsequent reactor coolant pump seal damage.	II.K.2.16 and II.K.3.25			
(1)(iv)	Perform an analysis of the probability of a small-break LOCA caused by a stuck-open power-operated relief valve (PORV). If this probability is a significant contributor to the probability of small-break LOCAs from all causes, provide a description and evaluation of the effect on small-break LOCA probability of an automatic PORV isolation system that would operate when the RCS pressure falls after the PORV has opened. (PWRs only)	II.K.3.2			

* Alphanumeric designations corresponding to related action plan items in NUREG-0718 and NUREG-0660, are provided herein for information only.

50.34(f) Item	Requirement	Action Plan Item*	N/A	Yes	No
(1)(v)	Perform an evaluation of the safety effectiveness of providing for separation of high-pressure coolant injection (HPCI) and RCIC system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and of providing that both systems restart on low water level. (For plants with high-pressure core spray [HPCS] systems in lieu of HPCI systems, substitute the words, “high-pressure core spray” for “high-pressure coolant injection” and “HPCS” for “HPCI.”) (BWRs only)	II.K.3.13			
(1)(vi)	Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. (BWRs only)	II.K.3.16			
(1)(vii)	Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system (ADS) design modifications that would eliminate the need for manual activation to ensure adequate core cooling. (BWRs only)	II.K.3.18			
(1)(viii)	Perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low-pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present. (BWRs only)	II.K.3.21			
(1)(ix)	Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the RCIC and HPCI systems, following a complete LOOP to the plant for at least 2 hours. (For plants with high-pressure core spray [HPCS] systems in lieu of high-pressure coolant injection systems, substitute the words, “high-pressure core spray” for “high-pressure coolant injection” and “HPCS” for “HPCI.”) (BWRs only)	II.K.3.24			
(1)(x)	Perform a study to ensure that the automatic depressurization system, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves. (BWRs only)	II.K.3.28			
(1)(xi)	Provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown. (BWRs only)	II.K.3.45			

* Alphanumeric designations corresponding to related action plan items in NUREG-0718 and NUREG-0660, are provided herein for information only.

50.34(f) Item	Requirement	Action Plan Item*	N/A	Yes	No
(2) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved GSI.					
(2)(i)	Provide a simulator capability that correctly models the control room and includes the capability to simulate small-break LOCAs. (Applicable to construction permit applicants only)	I.A.4.2			
(2)(ii)	Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with (the Institute of Nuclear Power Operations) and other industry efforts. (Applicable to construction permit applicants only)	I.C.9			
(2)(iii)	Provide, for Commission review, a control room design that reflects state-of-the-art human factors principles prior to committing to fabrication or revision of fabricated control room panels and layouts.	I.D.1			
(2)(iv)	Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded.	I.D.2			
(2)(v)	Provide for automatic indication of the bypassed and operable status of safety systems.	I.D.3			
(2)(vi)	Provide the capability of high-point venting of noncondensible gases from the RCS, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room, and their operation shall not lead to an unacceptable increase in the probability of LOCA or an unacceptable challenge to containment integrity.	II.B.1			

* Alphanumeric designations corresponding to related action plan items in NUREG-0718 and NUREG-0660, are provided herein for information only.

50.34(f) Item	Requirement	Action Plan Item*	N/A	Yes	No
(2)(vii)	Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term[11] radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. [11] Footnote 11 in 10 CFR 50.34(f) reads as follows: "The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products."	II.B.2			
(2)(viii)	Provide a capability to promptly obtain and analyze samples from the RCS and containment that may contain accident source term[11] radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, radioiodines and cesiums, and nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations.	II.B.3			
(2)(x)	Provide a test program and associated model development, and conduct tests to qualify RCS relief and safety valves and, for PWRs, PORV block valves, for all fluid conditions expected under operating conditions, transients, and accidents. Consideration of ATWS conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed.	II.D.1			
(2)(xi)	Provide direct indication of relief and safety valve position (open or closed) in the control room.	II.D.3			
(2)(xii)	Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide AFW system flow indication in the control room. (PWRs only)	II.E.1.2			
(2)(xiii)	Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available. (PWRs only)	II.E.3.1			

* Alphanumeric designations corresponding to related action plan items in NUREG-0718 and NUREG-0660, are provided herein for information only.

50.34(f) Item	Requirement	Action Plan Item*	N/A	Yes	No
(2)(xiv)	Provide containment isolation systems that:	II.E.4.2			
	(A) Ensure all non-essential systems are isolated automatically by the containment isolation system				
	(B) For each non-essential penetration (except instrument lines) have two isolation barriers in series				
	(C) Do not result in reopening of the containment isolation valves on resetting of the isolation signal				
	(D) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation				
	(E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs				
(2)(xv)	Provide a capability for containment purging/venting designed to minimize the purging time consistent with as low as reasonably achievable (ALARA) principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions.	II.E.4.4			
(2)(xvi)	Establish a design criterion for the allowable number of actuation cycles of the ECCS and reactor protection system consistent with the expected occurrence rates of severe overcooling events (considering both anticipated transients and accidents). (B&W designs only)	II.E.5.1			
(2)(xvii)	Provide instrumentation to measure, record, and readout in the control room (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples.	II.F.1			
(2)(xviii)	Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWRs, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWRs and BWRs.	II.F.2			
(2)(xix)	Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage.	II.F.3			

* Alphanumeric designations corresponding to related action plan items in NUREG-0718 and NUREG-0660, are provided herein for information only.

50.34(f) Item	Requirement	Action Plan Item*	N/A	Yes	No
(2)(xx)	Provide power supplies for pressurizer relief valves, block valves, and level indicators such that (A) level indicators are powered from vital buses; (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety, and (C) electric power is provided from emergency power sources. (PWRs only)	II.G.1			
(2)(xxi)	Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable. (BWRs only)	II.K.1.22			
(2)(xxii)	Perform a failure modes and effects analysis of the integrated control system (ICS) to include consideration of failures and effects of input and output signals to the ICS. (B&W designs only)	II.K.2.9			
(2)(xxiii)	Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip. (B&W designs only)	II.K.2.10			
(2)(xxiv)	Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. (BWRs only)	II.K.3.23			
(2)(xxv)	Provide an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a near-site Emergency Operations Facility.	III.A.1.2			
(2)(xxvi)	Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term[11] radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and the public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency.	III.D.1.1			
(2)(xxvii)	Provide for monitoring of in-plant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions.	III.D.3.3			

(2)(xxviii)	Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in an accident source term[11] release, and make necessary design provisions to preclude such problems.	III.D.3.4			

* Alphanumeric designations corresponding to related action plan items in NUREG-0718 and NUREG-0660, are provided herein for information only.

50.34(f) Item	Requirement	Action Plan Item*	N/A	Yes	No
(3) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy paragraph (a)(1) of this section or to address the applicant's technical qualifications and management structure and competence.					
(3)(i)	Provide administrative procedures for evaluating operating, design, and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant.	I.C.5			
(3)(ii)	Ensure that the QA list required by Criterion II in Appendix B to 10 CFR Part 50 includes all SSC important to safety.	I.F.1			
(3)(iii)	Establish a QA program based on consideration of (A) ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) performing QA/quality control (QC) functions at construction sites to the maximum feasible extent; (C) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction, and installation; (D) establishing criteria for determining QA programmatic requirements; (E) establishing qualification requirements for QA and QC personnel; (F) sizing the QA staff commensurate with its duties and responsibilities; (G) establishing procedures for maintenance of "as-built" documentation; and (H) providing a QA role in design and analysis activities.	I.F.2			
(3)(iv)	Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot-diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system.	II.B.8			
(3)(vi)	For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere.	II.E.4.1			

* Alphanumeric designations corresponding to related action plan items in NUREG-0718 and NUREG-0660, are provided herein for information only.

50.34(f) Item	Requirement	Action Plan Item*	N/A	Yes	No
(3)(vii)	Provide a description of the management plan for design and construction activities, to include: (A) the organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources director by the applicant; (C) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) proposed procedures for handling the transition to operation; (E) the degree of top-level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort.	II.J.3.1			

* Alphanumeric designations corresponding to related action plan items in NUREG-0718 and NUREG-0660, are provided herein for information only.