Proposed - For Interim Use and Comment

U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER^{TM} iPWR DESIGN

15.0 INTRODUCTION - TRANSIENT AND ACCIDENT ANALYSES

REVIEW RESPONSIBILITIES

Primary - Organizations responsible for review of transient and accident analyses for iPWRs

Secondary - None

The evaluation of the safety of a nuclear power plant requires analyses of the plant's responses to postulated equipment failures or malfunctions. Such analyses help to determine the limiting conditions for operation, limiting safety system settings, and design specifications for safety-related components and systems to protect public health and safety. These analyses are a focal point of the license amendment request (LAR), design certification (DC), and combined license (COL) reviews.

I. <u>AREAS OF REVIEW</u>

The specific areas of review are as follows:

1. <u>Categorization of Transients and Accidents</u>. The reviewer ensures that the applicant's selection and assembly of the plant transient and accident analyses represent a sufficiently broad spectrum of transients and accidents, or initiating events.

   Initiating events are categorized according to expected frequency of occurrence and by type. Categorization by frequency of occurrence provides a basis for selection of the applicable analysis acceptance criteria for each initiating event. Categorization of initiating events by type provides a basis for comparison between events, which makes it possible to identify and evaluate the limiting cases (i.e., the cases that can challenge the analysis acceptance criteria).

   A. <u>Categorization According to Frequency of Occurrence</u>. Each initiating event is categorized as either an anticipated operational occurrence (AOO) or as a postulated accident.

      AOOs, as defined in Appendix A to Title 10, *Code of Federal Regulations* (10CFR) 10 CFR Part 50, are those conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit.

      The DSRS uses the term AOOs to refer to the events that are categorized in Regulatory Guide (RG) 1.70 and in RG 1.206 as incidents of moderate frequency (i.e., events that are expected to occur several times during the plant's lifetime) and infrequent events (i.e., events that may occur during the lifetime of the plant).

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Incidents of moderate frequency and infrequent events are also known as Condition II and Condition III events, respectively, in the commonly used, oft-cited but unofficial American Nuclear Society (ANS) standards. The reviewer will continue to evaluate applications, according to the categorizations and acceptance criteria of References 4 and 5, for licensees that have these categorizations in their licensing bases, or if they wish, according to the categorizations and acceptance criteria of this DSRS section. The reviewer will evaluate new applications (i.e., those pertaining to plants that are not yet constructed) according to the categorizations and acceptance criteria of this DSRS section.

The following are some examples of AOOs in pressurized-water reactor (PWR) designs:

- Inadvertent control rod or rod group withdrawal
- Loss or interruption of core coolant flow, excluding reactor coolant pump locked rotor
- Inadvertent moderator cooldown
- Depressurization by spurious operation of an active element, such as a relief valve
- Blowdown of reactor coolant through a safety valve
- Loss of normal feedwater
- Loss of condenser cooling
- Steam generator tube leaks
- Reactor-turbine load mismatch, including loss of load and turbine trip
- Control rod drop (inadvertent addition of absorber)
- Minor reactor coolant system (RCS) leak or loss of reactor coolant such as from a small ruptured pipe or from a crack in a large pipe
- Minor secondary system break
- Loss of offsite power
- Operation with a fuel assembly in an improper position
- Loss of feedwater heating

The individual event sections of the DSRS address specific AOOs and their appropriate variations.

Anticipated transients without scram (ATWSs) are AOOs in which a reactor scram is demanded but fails to occur because of a common-mode failure in the reactor scram system. ATWS events, therefore, are AOOs that postulate complete failure of the required (single-failure proof) protection system. As such, they are beyond the design basis, and consequently, ATWS events are addressed separately (see DSRS Section 15.8).

Postulated accidents are unanticipated occurrences (i.e., they are postulated but not expected to occur during the life of the nuclear power plant).

Postulated accidents are also known as Condition IV events in the unofficial ANS standards.

The following are some examples of postulated accidents in PWRs of current designs:

- Major rupture of a pipe containing reactor coolant up to and including double-ended rupture of the largest pipe in the reactor coolant pressure boundary

- Ejection of a control rod assembly

- Major secondary system pipe rupture up to and including double-ended rupture

- Single reactor coolant pump locked rotor

The sections of the DSRS dealing with the individual events address specific postulated accidents and appropriate variations (e.g., design-specific variations).

B. Categorization According to Type. AOOs and postulated accidents are also categorized according to type. The type of AOO or postulated accident is defined by its effect on the plant. For example, one type of AOO or postulated accident will cause the RCS to pressurize and possibly jeopardize RCS integrity. Another type will cause the RCS to depressurize and possibly jeopardize fuel cladding integrity. It is useful to categorize and organize analyses of AOOs and postulated accidents according to type, so that analysts can compare them on common bases, effects, and safety limits. Such comparisons can help to identify limiting events and cases for detailed examination and eliminate nonlimiting cases from further consideration.

AOOs and postulated accidents can be grouped into the following seven types:

1. Increase in heat removal by the secondary system
2. Decrease in heat removal by the secondary system
3. Decrease in RCS flow rate
4. Reactivity and power distribution anomalies
5. Increase in reactor coolant inventory
6. Decrease in reactor coolant inventory
7. Radioactive release from a subsystem or component

The review of AOOs and postulated accident analyses, within a type, can (and should) encompass a variety of cases, each designed to produce effects or results that challenge designated safety limits. For example, one case study of the turbine trip event, an AOO that causes a decrease in heat removal by the secondary system, can be designed to yield a high peak RCS pressure, and another case study of the same AOO can be designed to yield a low, minimum thermal margin. The former case tests the safety limit for RCS pressure boundary integrity, while the latter case tests the safety limit that protects fuel cladding integrity.

The reviewer considers the possible case variations of AOOs and postulated accidents presented to verify that the licensee has identified the limiting cases. The reviewer evaluates licensees' claims that individual AOOs and postulated accidents are limiting or nonlimiting, or bounded by other AOOs and postulated accidents, with particular attention to the bases used for comparison. Comparison of AOOs to other AOOs within a type, for example, is easily justified. Comparison of AOOs of one type to postulated accidents of another type requires closer scrutiny and more justification from the licensee.

2. Analysis Acceptance Criteria. If the risk of an event is defined as the product of the event's frequency of occurrence and its consequences, then the design of the plant should be such that all the AOOs and postulated accidents produce about the same level of risk (i.e., the risk is approximately constant across the spectrum of AOOs and postulated accidents). This is reflected in the general design criteria (GDC), which generally prohibit relatively frequent events (AOOs) from resulting in serious consequences, but allow the relatively rare events (postulated accidents) to produce more severe consequences.

   The reviewer will consider the results of licensees' analyses and evaluations of individual initiating events to ascertain whether the licensee has satisfied the applicable analysis acceptance criteria for each of the events. The licensee may propose the use of alternate acceptance criteria appropriate to the particular plant design and operation (e.g., or new reactor design applications). In such cases, the reviewer will consider the alternate criteria and determine whether they are equivalent, in function and consequences, to the current criteria (see below).

   A. Analysis Acceptance Criteria for AOOs. The following are the specific criteria necessary to meet the requirements of GDC for AOOs:

      i. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

      ii. Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit.

      The reviewer applies a third criterion, based on the ANS standards to ensure that there is no possibility of initiating a postulated accident with the frequency of occurrence of an AOO.

iii. An AOO should not generate a postulated accident without other faults occurring independently or result in a consequential loss of function of the RCS or reactor containment barriers.

B. Analysis Acceptance Criteria for Postulated Accidents. Unlike an AOO, a postulated accident could result in sufficient damage to preclude resumption of plant operation. A list of the basic criteria necessary to meet the requirements of GDC for postulated accidents appears below. Individual sections of the DSRS may specify additional criteria pertaining to certain postulated accidents.

i. Pressure in the RCS and main steam system should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.

ii. Fuel cladding integrity will be maintained if the minimum DNBR remains above the 95/95 DNBR limit ~~for PWRs~~. If the minimum DNBR does not meet these limits, then the fuel is assumed to have failed.

iii. The release of radioactive material shall not result in offsite doses in excess of the guidelines of 10 CFR Part 100.

iv. A postulated accident shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.

For loss-of-coolant accidents (LOCAs), the following analysis acceptance criteria of 10 CFR 50.46 also apply:

i. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

ii. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

iii. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

iv. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

v. After any calculated successful initial operation of the emergency core cooling system (ECCS), the calculated core temperature shall should be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

3. Plant Characteristics Considered in the Safety Evaluation. The reviewer ensures that the application contains the key plant parameters considered in the safety evaluation (e.g., core power, core inlet temperature, reactor system pressure, core flow, axial and radial power distribution, fuel and moderator temperature coefficient, void coefficient, reactor kinetics parameters, available shutdown rod worth, and control rod insertion characteristics). The reviewer checks that the range of values for plant parameters is representative of fuel exposure or core reload, and that the range is sufficiently broad to cover the predicted fuel cycle ranges, to the extent practicable, based on the fuel design and acceptable analytical methodology at the time of the LAR, DC, or COL application. The reviewer also ensures that the application specifies the permitted fluctuations and uncertainties associated with reactor system parameters and assumes the appropriate conditions, within the operating band, as initial conditions for transient analysis.

4. Assumed Protection and Safety Systems Actions. The reviewer ensures that the application lists the settings of all the protection and safety systems functions that are used (i.e., credited) in the safety evaluation. Typical protection and safety systems functions include reactor trips, isolation valve closures, ECCS initiation and ECCS. In evaluations of AOOs and postulated accidents, the performance of each credited protection or safety system is required to include the effects of the most limiting single active failure. This verifies satisfaction of the GDC criteria that require protection and safety systems to adequately perform their intended safety functions in the presence of single active failures. The reviewer also ascertains that the application lists the expected limiting delay time for each protection or safety system function and describes the acceptable methodology for determining uncertainties (from the combined effects of calibration error, drift, instrumentation error, and other factors) to be included in the establishment of the trip setpoints and allowable values specified in the plant technical specifications.

5. Evaluation of Individual Initiating Events. The reviewer ensures that the application includes an evaluation of each initiating event, using the format in Subsection I.6 of this DSRS section. For initiating events that are determined to be not limiting, the reviewer may evaluate qualitative justifications and conduct comparisons with the corresponding, more limiting initiating events.

6. Event Evaluation

   A. Identification of Causes and Frequency Classification. For each initiating event evaluated, the reviewer ensures that the application includes a description of the occurrences that can lead to the event and a categorization of the event as either an AOO or postulated accident. The reviewer also checks for clear definitions of the analysis acceptance criteria appropriate to the specific nature of the initiating event, as well as the event's categorization.

   B. Sequence of Events and Systems Operation. The reviewer verifies that the application addresses the following considerations for each initiating event:

      i. Step-by-step sequence of events, from event initiation to the final stabilized condition (i.e., identification on a time scale of each significant occurrence, including flux monitor trips, insertion of control rods, attainment of primary coolant safety valve set points, opening and closing of safety valves, generation of containment isolation signals, and containment isolation) and

identification of all operator actions credited in the transient and accident analyses for consequence mitigation

ii. Extent to which normally operating plant instrumentation and controls are assumed to function

iii. Extent to which plant and reactor protection systems are required to function

iv. Credit taken for the functioning of normally operating plant systems

v. Credited operation of engineered safety systems

vi. Assurance of consistency between the safety analyses and the emergency response guidelines/emergency procedure guidelines or emergency operating procedures with respect to the operator response (including action time) and available instrumentation

The reviewer verifies that the applicant has specified only safety-related systems or components for use in mitigating AOO and postulated accident conditions, and has included the effects of single active failures in those systems and components. If nonsafety-related systems are operational during a response to an initiating event and negatively impact the response, then applicant would be expected to include this in analysis. The reviewer may consider the licensee's technical justifications for the operation of nonsafety-related systems or components (e.g., when they are used as backup protection and when they are not disabled, except by a detectable, random, and independent failure).

The reviewer ascertains that the applicant has evaluated the effects of single active failures and operator errors and that the licensee's application contains sufficient detail to permit independent evaluation of the adequacy of systems, as they relate to the subject events.

C. Core, System, and Barrier Performance

i. Evaluation Model. The reviewer ensures that the applicant has discussed the evaluation model used and any simplifications or approximations introduced to perform the analyses and identified digital computer codes used in the analysis. If the analysis uses more than one computer code, the applicant should describe the method used to connect the codes. The reviewer verifies that the applicant has discussed the important output of the codes under "results" with emphasis on the input data and the extent or range of variables investigated and that the applicant has included detailed descriptions of evaluation models and digital computer codes or listings by referencing documents that are available to the NRC.

The reviewer ensures that the applicant has provided a table listing the titles of topical reports (TRs) that describe models or computer codes used in transient and accident analyses and listed the associated NRC safety evaluation reports approving those TRs. The reviewer checks that implementations of NRC-approved models or codes are within the applicable ranges and conditions and that the applicant has demonstrated

compliance with each of the conditions and limitations imposed by the NRC staff in its safety evaluation reports that approve the TRs.

ii. Input Parameters and Initial Conditions. The reviewer verifies that the applicant has (1) identified the major input parameters and initial conditions used in the analyses; (2) included the initial values of other variables and parameters in the application if they are used in the analyses of the particular event under study; (3) ensured that the parameters and initial conditions used in the analyses are suitably conservative; and (4) discussed the bases (including the degree of conservatism) used to select the numerical values of the input parameters.

iii. Results. The reviewer ensures that the applicant has presented the results of the analyses, including key parameters as a function of time during the course of the transient or accident. The following are examples of parameters that should be included:

- Neutron power
- Thermal power
- Heat fluxes, average and maximum
- RCS pressure
- Departure from nucleate boiling ratio, DNBR
- Coolant conditions, including inlet temperature, core average temperature, average exit and hot channel exit temperatures, and steam volume fractions
- Temperatures, including maximum fuel centerline temperature, maximum clad temperature, or maximum fuel enthalpy
- Reactor coolant inventory, including total inventory and coolant level in various locations in the RCS
- Secondary (power conversion) system parameters, including steam flow rate, steam pressure and temperature, feedwater flow rate, feedwater temperature, and steam generator inventory
- ECCS flow rates and pressure differentials across the core, as applicable
- Containment pressure
- Relief and/or safety valve flow rate
- Automatic depressurization valves flow rates

- Flow rate from the RCS to the containment system, if applicable

- Pressurizer water volume

In addition, the discussion of the results should emphasize the margins between the predicted values of various core parameters, as well as the values of those parameters that would represent limiting acceptable conditions.

Review Interfaces

Other DSRS sections interface with this section as follows:

1. Design basis radiological consequence analyses associated with design basis accidents are reviewed under DSRS Section 15.0.3.

II. ACCEPTANCE CRITERIA

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. 10 CFR Part 20, “Standards for Protection Against Radiation “

2. 10 CFR Part 50,”Domestic Licensing of Production and Utilization Facilities “ (especially 10 CFR 50.46 and Appendix A)

3. 10 CFR Part 100,”Reactor Site Criteria “

4. 10 CFR Part 52,”Early Site Permits; Standard Design Certification; and Combined Licenses for Nuclear Power Plants “

The following GDC from Appendix A to 10 CFR Part 50 are relevant to DSRS Section 15:

1. GDC 2, as it relates to the seismic design of structures, systems, and components (SSCs) whose failure could cause an unacceptable reduction in the capability of the residual heat removal system.

2. GDC 4, as it relates to the requirement that SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accident conditions, including such effects as pipe whip and jet impingement.

3. GDC 5, as it relates to the requirement that any sharing among nuclear power units of SSCs important to safety will not significantly impair their safety function.

4. GDC 10, as it relates to the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations including AOOs.

5. GDC 13, as it relates to instrumentation and controls provided to monitor variables over anticipated ranges for normal operations, for AOOs, and for accident conditions.

6. GDC 15, as it relates to the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations, including AOOs.

7. GDC 17, as it relates to the requirement that an onsite and offsite electric power system be provided to permit the functioning of SSCs important to safety. The safety function for each system (assuming the other system is not working) shall be to provide sufficient capacity and capability to ensure that the acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded during an AOO and that core cooling, containment integrity, and other vital functions are maintained in the event of an accident.

8. GDC 19, as it relates to the requirement that a control room be provided from which personnel can operate the nuclear power unit during both normal operating and accident conditions, including a LOCA.

9. GDC 20, as it relates to the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that the plant does not exceed specified acceptable fuel design limits during any condition of normal operation, including AOOs.

10. GDC 25, as it relates to the requirement that the reactor protection system be designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control system, such as accidental withdrawal of control rods.

11. GDC 26, as it relates to the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded even during AOOs. This is accomplished by ensuring that the applicant has allowed an appropriate margin for malfunctions such as stuck rods.

12. GDC 27 and 28, as they relate to the RCS being designed with an appropriate margin to ensure that acceptable fuel design limits are not exceeded and that the capability to cool the core is maintained.

13. GDC 29, as it relates to the design of the protection and reactivity control systems and their performance (i.e., to accomplish their intended safety functions) during AOOs.

14. GDC 31, as it relates to the RCS being designed with sufficient margin to ensure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized.

15. GDC 34, as it relates to the capability to transfer decay heat and other residual heat from the reactor so that fuel and pressure boundary design limits are not exceeded.

16. GDC 35, as it relates to the RCS and associated auxiliaries being designed to provide abundant emergency core cooling.

17. GDC 55, as it relates to the isolation requirements of small-diameter lines connected to the primary system.

18. GDC 60, as it relates to the radioactive waste management systems being designed to control releases of radioactive materials to the environment.

19. GDC 61, as it relates to the requirement that the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions.

DSRS Acceptance Criteria

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the U.S Nuclear Regulatory Commission's (NRC's) regulations identified above are set forth below. The DSRS is not a substitute for the NRC's regulations, and compliance with it is not required. Identifying the differences between this DSRS section and the design features, analytical techniques, and procedural measures proposed for the facility, and discussing how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria, is sufficient to meet the intent of 10 CFR 52.47(a)(9), "Contents of applications; technical information." The same approach may be used to meet the requirements of 10 CFR 52.79(a)(41) for COL applications.

Subsection I.2 of this DSRS section discusses general acceptance criteria, and DSRS Chapter 15 subsections discuss specific acceptance criteria for transients or accidents.

III. REVIEW PROCEDURES

The review procedures are based on the identified DSRS acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

1. Programmatic Requirements - In accordance with the guidance in NUREG-0800 "Introduction," Part 2 as applied to this DSRS Section, the staff will review the programs proposed by the applicant to satisfy the following programmatic requirements. If any of the proposed programs satisfies the acceptance criteria described in Subsection II, it can be used to augment or replace some of the review procedures. It should be noted that the wording of "to augment or replace" applies to nonsafety-related risk-significant SSCs, but "to replace" applies to nonsafety-related nonrisk-significant SSCs according to the "graded approach" discussion in NUREG-0800 "Introduction," Part 2. Commission regulations and policy mandate programs applicable to SSCs that include:

    A. Maintenance Rule SRP Section 17.6 (DSRS Section 13.4, Table 13.4, Item 17, Regulatory Guides 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." and RG 1.182; "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants".

    B. Quality Assurance Program SRP Sections 17.3 and 17.5 (DSRS Section 13.4, Table 13.4, Item 16).

    C. Technical Specifications (DSRS Section 16.0 and SRP Section 16.1) – including brackets value for DC and COL. Brackets are used to identify information or characteristics that are plant specific or are based on preliminary design information.

D. Reliability Assurance Program (SRP Section 17.4).

E. Initial Plant Test Program (Regulatory Guide 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants, ”DSRS Section 14.2, and DSRS Section 13.4, Table 13.4, Item 19).

F. ITAAC (DSRS Chapter 14).

2. In accordance with 10 CFR 52.47(a)(8),(21), and (22), for new reactor license applications submitted under Part 52, the applicant is required to (1) address the proposed technical resolution of unresolved safety issues (USIs) and medium- and high-priority generic safety issues (GSIs) that are identified in the version of NUREG-0933 current on the date 6 months before application and that are technically relevant to the design; (2) demonstrate how the operating experience insights have been incorporated into the plant design; and, (3) provide 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). Reference: 10 CFR 52.47(a)(21), 10 CFR 52.47(a)(22) , and 10 CFR 52.47(a)(8), respectively. These cross-cutting review areas should be addressed by the reviewer for each technical subsection and relevant conclusions documented in the corresponding SER section.

To evaluate the LAR, DC, or COL application, the reviewer verifies that the applicant has performed the applicable transient and accident analyses needed to demonstrate conformance to the regulations.

DSRS Chapter 15 subsections discuss specific review procedures for transients or accidents.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the staff’s technical review and analysis, as augmented by the application of programmatic requirements in accordance with the staff’s technical review approach in the DSRS Introduction, support conclusions of the following type to be included in the staff’s safety evaluation report. The reviewer also states the bases for those conclusions.

DSRS Chapter 15 subsections discuss the statements and conclusions of evaluation findings for transients or accidents.

V. IMPLEMENTATION

The staff will use this DSRS section in performing safety evaluations of mPower™-specific DC, or combined license (COL), applications submitted by applicants pursuant to 10 CFR Part 52. The staff will use the method described herein to evaluate conformance with Commission regulations.

Because of the numerous design differences between the mPower™ and large light-water nuclear reactor power plants, and in accordance with the direction given by the Commission in SRM- COMGBJ-10-0004/COMGEA-10-0001, “Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews,” dated August 31, 2010 (ML102510405), to develop risk-informed licensing review plans for each of the small modular reactor (SMR) reviews including the associated pre-application activities, the staff has developed the content of this DSRS section as an alternative method for mPower™ -specific DC, or COL submitted pursuant to

10 CFR Part 52 to comply with 10 CFR 52.47(a)(9), "Contents of applications; technical information."

This regulation states, in part, that the application must contain "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 content of this DSRS section has been accepted as an alternative method for complying with 10 CFR 52.47(a)(9) as long as the mPower™ DCD final safety analysis report does not deviate significantly from the design assumptions made by the NRC staff while preparing this DSRS section. The application must identify and describe all differences between the standard plant design and this DSRS section, and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria. If the design assumptions in the DC application deviate significantly from the DSRS, the staff will use the SRP as specified in 10 CFR 52.47(a)(9). Alternatively, the staff may supplement the DSRS section by adding appropriate criteria in order to address new design assumptions. The same approach may be used to meet the requirements of 10 CFR 52.79(a)(41), and COL applications.

VI. DEFINITIONS

Term	**Definition**
anticipated operational occurrences (AOOs)	Conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power. AOOs are also known as Condition II and III events.
anticipated transient without scram (ATWS)	AOO followed by the failure of the reactor trip portion of the protection system specified in GDC 20, because of common-mode failure.
common-mode failure	The result of an event which, because of dependencies, causes a coincidence of failure states of components in two or more separate channels of a redundancy system, leading to the failure of the defined system to perform its intended function.
departure from nucleate boiling (DNB)	The DNB acceptance criterion for an AOO is met when there is a 95 percent probability at a 95 percent confidence level (the 95/95 DNB criterion) that DNB will not occur, and the fuel centerline temperature stays below the melting temperature.
design basis	Information that identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted state of the art practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a

Term	Definition
	structure, system, or component must meet its functional goals.
design-basis accidents	Postulated accidents that are used to set design criteria and limits for the design and sizing of safety-related systems and components.
design-basis events	Conditions of normal operation, including AOOs, design-basis accidents, external events, and natural phenomena, for which the plant must be designed to ensure functions of safety-related electric equipment that ensures the integrity of the reactor coolant pressure boundary; the capability to shut down the reactor and maintain it in a safe shutdown condition; or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures.
general design criteria (GDC)	Reference 1 lists the GDC. The GDC that mention AOOs are 10, 13, 15, 17, 20, 26, 29, 60, and 64. The GDC that mention postulated accidents are 4, 16, 17, 22, 27, 28, 31, 41, 51, 61, and 64.
loss-of-coolant accident (LOCA)	A postulated accident that results in the loss of reactor coolant at a rate in excess of the replacement capability of the reactor coolant makeup system.
overpressurization	The condition that occurs when pressure exceeds the design pressure of the component of interest by more than 10 percent, in accordance with the ASME Code.
postulated accidents	Unanticipated conditions of operation (i.e., not expected to occur during the life of the nuclear power unit). Postulated accidents are also known as Condition IV events.
protection system	The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety. (GDC 20)
single failure	An occurrence that results in a component's loss of capability to perform its intended safety functions.

VII. REFERENCES

1. Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Plants."

2. RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

3. RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

4. ANS 51.1, “Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants“ (replaces ANSI N18.2), 1983 (withdrawn in 1998).

5. ANSI/ANS-52.1-1978, “Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants“ (withdrawn in 1998).

6. 10 CFR 50.62, “Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water Cooled Nuclear Power Plants.“

7. ASME Boiler and Pressure Vessel Code, Section III, “Nuclear Power Plant Components,“ Article NB-7000, “Protection Against Overpressure,“ American Society of Mechanical Engineers.

8. SECY-77-439, “Single-Failure Criterion,“ August 1977 (ADAMS Accession No. ML060260236).

9. 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities.“

10. 10 CFR Part 52, “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants.“