U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

15.2.6 LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES

REVIEW RESPONSIBILITIES

Primary - Organization responsible for review of transient and accidents analyses for PWRs/BWRs

Secondary - None

I. AREAS OF REVIEW

The loss of nonemergency alternating current (ac) power is assumed to result in the loss of all power to the station auxiliaries. This situation, which could be the result of a complete loss of either the external (offsite) grid or the onsite ac distribution system, is different from the loss of load condition considered in Standard Review Plan (SRP) Section 15.2.2 because, in the latter case, ac power remains available to operate the station auxiliaries. The major difference is that in the loss-of-ac-power transient, all the reactor coolant circulation pumps are tripped simultaneously by the initiating event, resulting in a flow coast-down as well as a decrease in heat removal by the secondary system.

Within a few seconds, the turbine trips, the reactor coolant system is isolated, and the pressure and temperature of the coolant increase. A reactor trip is initiated. The diesel generators start automatically and provide electric power to the vital loads. The sensible and decay heat loads are handled by actuation of the steam relief system, steam bypass to the condenser, reactor core isolation cooling system in a boiling-water reactor (BWR), emergency core cooling system, and auxiliary feedwater system in a pressurized-water reactor (PWR).

For the Westinghouse-AP1000 plants, a complete loss of the offsite grid accompanied by a turbine-generator trip may result in the loss of ac power, a moderate-frequency event. For removal of decay heat, this event is more severe than the turbine trip event because a reactor coolant system (RCS) flow coast-down accompanies the decrease in heat removal by the secondary system and further reduces the capacity of the primary coolant to remove heat from the core. The reactor trips upon reaching one of the reactor trip setpoints in the primary and secondary systems as a result of either the flow coast-down and decrease in secondary heat removal or the loss of power to the control rod drive mechanisms.

For the economic simplified BWR all scram signal paths, including valve position, high flux, high pressure, low level, and all manual attempts are assumed to fail.

The loss of ac power has the following effects:

- An immediate load rejection with fast closure of the turbine control valves.

- Due to the loss of power to the condensate and feedwater pumps, feedwater is lost.

- The reactor is isolated after loss of main condenser vacuum.

The review of the loss-of-ac-power transient includes the sequence of events, the analytical model, the values of parameters in the analytical model, and the predicted consequences of the transient. The specific areas of review are as follow:

1. The sequence of events described in the applicant's safety analysis report (SAR) is reviewed with concentration on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition.

2. The analytical methods are reviewed to ascertain whether the mathematical modeling and computer codes have been reviewed previously and accepted by the staff. If a referenced analytical method has not been reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by appropriate reviewers.

3. The predicted results of the transient analysis are reviewed for whether the consequences meet the acceptance criteria of subsection II of this SRP session. The results of the analysis are reviewed for whether the values of pertinent system parameters are within expected ranges for the type and class of reactor under review.

4. COL Action Items and Certification Requirements and Restrictions. COL action items may be identified in the NRC staff's final safety evaluation report (FSER) for each certified design to identify information that COL applicants must address in the application. Additionally, DC's contain requirements and restrictions (e.g., interface requirements) that COL applicants must address in the application. For COL applications referencing a DC, the review performed under this SRP section includes

information provided in response to COL action items and certification requirements and restrictions pertaining to this SRP section, as identified in the FSER for the referenced certified design.

Review Interfaces

The listed SRP sections interface with this section as follows:

1. Aspects of the sequence described in the SAR are reviewed for whether the reactor and plant protection and safeguards controls and instrumentation systems function as assumed in the safety analysis for automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems.

2. The design of the auxiliary feedwater system is reviewed for whether the requirements and guidance of NUREG-0737, II.E.1.2, and 10 CFR 50.34(f)(2)(xii) are met. The reviewers consult on whether the appropriate delay time for auxiliary feedwater initiation is assumed in the analysis.

3. Using fuel damage results, the reviewer evaluates the radiological consequences of fuel failure.

4. Sections 4.2 through 4.4: the values of the parameters in the analytical models of the reactor core are reviewed for compliance with plant design and specified operating conditions, acceptance criteria for fuel cladding damage limits are determined, and the core physics, fuel design, and core thermal-hydraulics data in the SAR analysis are reviewed.

5. Section 10.4.9: reliability of the auxiliary feedwater system is reviewed in accordance with the requirements and guidance of NUREG-0737, II.E.1.1, and NUREG-0660, II.K.2.(1) (item 1 of Table C.2), and 10 CFR 50.34(f)(1)(ii). The reviewers consult on whether the operational assumptions for the auxiliary feedwater system in the analysis is appropriate.

6. For COL reviews of operational programs, the review of the applicant's implementation plan is performed under Section 13.4, "Operational Review."

7. Section 16.0: review of technical specifications.

The specific acceptance criteria and review procedures are contained in the referenced SRP sections.

II. ACCEPTANCE CRITERIA

Requirements

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

1. General Design Criterion (GDC) 10, as to RCS design with appropriate margin so specified acceptable fuel design limits are not exceeded during normal operation including anticipated operational occurrences.

2. GDC 13, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 15, as to design of the RCS and its auxiliaries with appropriate margin so the pressure boundary is not breached during normal operation including anticipated operational occurrences.

4. GDC 26, as to reliable control of reactivity changes so specified acceptable fuel design limits are not exceeded in anticipated operational occurrences. This control is accomplished by appropriate margin for malfunctions like stuck rods.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for review described in Subsection I of this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

Specific criteria necessary to meet the relevant requirements of GDCs 10, 13, 15, and 26 for events of moderate frequency (see definitions of design and plant process conditions in References 18 and 19) are as follow:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values (Reference 19).

2. Fuel cladding integrity should be maintained by keeping the minimum departure from nucleate boiling ratio (DNBR) above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) above the minimum critical power ratio safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).

3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

4. For the requirements of GDCs 10 and 15, the positions of Regulatory Guide (RG) 1.105, "Instrument Setpoints for Safety-Related Systems," have impact on the plant response to the type of transient addressed in this SRP section.

5. The most limiting plant system single failure, as defined in the "Definitions and Explanations" of 10 CFR Part 50, Appendix A, must be assumed in the analysis and must satisfy the positions of RG 1.53.

The applicant's analysis of the loss of ac power transient should be based on an acceptable and NRC-approved model. If the applicant proposes analytical methods not approved, these are evaluated by the staff for acceptability and approval. For new generic methods, the reviewer requests an appropriate evaluation.

The parameter values in the analytical model should be suitably conservative. The following values are acceptable:

A. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2 percent to account for power measurement uncertainties unless the applicant can justify a lower power level. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.

B. Conservative scram characteristics are assumed (*i.e.*, for a PWR maximum time delay with the most reactive rod held out of the core and for a BWR a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate).

C. The core burn-up is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, power profile, and radial power distribution.

D. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with RG 1.105. Compliance with RG 1.105 is determined.

Technical Rationale

The technical rationale for application of these requirements and/or SRP acceptance criteria to the areas of review addressed by this SRP section is discussed in the following paragraphs:

1. GDC 10 requires that the reactor core and its coolant, control, and protection systems be designed with appropriate margin so specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

   GDC 10 applies to this section because the reviewer evaluates the consequences of the loss of nonemergency ac power to the station auxiliaries. This anticipated operational occurrence creates a potential for specified acceptable fuel design limits to be exceeded. Within seconds after the loss of power, the turbine and the reactor both trip, and the pressure and temperature of the reactor coolant increase. RG 1.53 provides guidance for application of the single-failure criterion to the design and analysis of nuclear power plant protection systems. RG 1.105 describes a method acceptable to the staff for keeping instrument setpoints within the technical specification limits.

   GDC 10 requirements provide assurance that specified acceptable fuel design limits are exceeded and that fuel cladding integrity is maintained in loss of nonemergency ac power to the station auxiliaries.

2. GDC 13 requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges to assure adequate safety, and of controls that can maintain these variables and systems within prescribed operating ranges.

   GDC 13 applies to this section because the reviewer evaluates the sequence of events, including automatic actuations of protection systems, and manual actions, and determines whether the sequence of events is justified, based upon the expected values of the relevant monitored parameters and instrument indications.

3. GDC 15 requires that the RCS and its auxiliary, control, and protection systems be designed with sufficient margin so design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. GDC 15 applies to this section because the reviewer evaluates the consequences of the loss of nonemergency ac power to the station auxiliaries. This transient is an anticipated operational occurrence, and the reactor coolant pressure must be analyzed to confirm whether the pressure acceptance criterion is satisfied.

   GDC 15 requirements provide assurance that the design conditions of the reactor coolant pressure boundary are not exceeded in the loss of nonemergency ac power to the station auxiliaries.

4 . GDC 26 requires that one of the reactivity control systems consist of control rods capable of reliably controlling reactivity changes with appropriate margin for malfunctions like stuck rods so that specified acceptable fuel design limits are not exceeded under conditions of normal operation, including anticipated operational occurrences.

   GDC 26 applies because the transient analyzed in this section involves the movement of control rods in response to the loss of nonemergency ac power and because rod misalignment, including stuck rods, can produce thermal-hydraulic conditions more severe than otherwise. GDC 26 requires a thermal margin sufficient to accommodate these conditions. Under SRP Section 15.2.6 these margins are examined where applicable for whether the thermal criteria remain satisfied.

   GDC 26 requirements provide assurance by appropriate margin for malfunctions of the reactivity control system, including stuck rods, that specified acceptable fuel design limits are not exceeded.

III. <u>REVIEW PROCEDURES</u>

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For each area of review specified in subsection I of this SRP section, the review procedure is identified below. These review procedures are based on the identified SRP acceptance criteria. For deviations from these specific acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the relevant NRC requirements identified in subsection II.

The procedures are for review of construction permit (CP), COL, and operating license (OL) applications. During the CP review the values of system parameters and setpoints in the analysis are preliminary in nature and subject to change. At the COL or OL review stage, final values should be in the analysis and the reviewer should compare these to the limiting safety system settings in the proposed technical specifications.

1. The description of the loss of ac power transient presented by the applicant in the SAR is reviewed by the organization responsible for review of transients and accidents analyses regarding the occurrences leading to the initiating event. The sequence of events from initiation until condition stabilization is reviewed to ascertain:

   A. The extent to which normally operating plant instrumentation and controls are assumed to function.

   B. The extent to which plant and reactor protection systems are required to function.

   C. The credit taken for the functioning of normally operating plant systems.

   D. The operation of engineered safety systems required.

   E. The extent to which operator actions are required.

   F. The operation of standby diesel generators required.

   G. Whether the description accounts for appropriate margin for malfunctions like stuck rods (per subsection II.B of this SRP section).

2. If the SAR states that the loss of ac power transient is not as limiting as some other similar transient, the reviewer evaluates the applicant's justification. If the SAR presents a quantitative analysis of the loss of ac power transient, the timing of the initiation of those protection, engineered safety, standby diesel generator, and other systems needed to limit transient consequences acceptably level is reviewed. The reviewer compares the predicted variation of system parameters to various trip and system initiation setpoints. The review of SAR Chapter 7 confirms whether the instrumentation and control systems design is consistent with the requirements for safety system actions for these events. To the extent necessary, the reviewer evaluates the effects of single, active system and component failures which may affect the course of the transient. This aspect of the review uses the procedures described in SRP sections for SAR Chapters 4, 5, 6, 7, 8, and 9.

3. The applicant's mathematical models for evaluating core performance and predicting system pressure in the RCS and main steam lines are reviewed by for whether these models have been reviewed and accepted by the staff. If not, the reviewer initiates a generic review of the applicant's proposed model.

4. System parameter values and initial core and system conditions as input to the model are reviewed. Of particular importance are the reactivity coefficients and control rod worths in the applicant's analysis and the variations of moderator temperature, void, and

Doppler coefficients of reactivity with core life. The applicant's justification to show that it has selected the core burn-up that yields minimum margins is evaluated. The values of the reactivity parameters in the applicant's analysis are reviewed.

5. The results of the analysis are reviewed and compared to the acceptance criteria of subsection II of this SRP section as to the maximum pressure in the reactor coolant and main steam systems. The variations during the transient of neutron power, heat fluxes (average and maximum), RCS pressure, minimum DNBR (PWR) or CPR (BWR), core and recirculation loop coolant flow rates (BWR), coolant conditions (inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam fractions), steam line pressure, containment pressure, pressure relief valve flow rate, and flow rate from the RCS to the containment system (if applicable) are reviewed. Time-related variations of the following parameters are reviewed:

    A. reactor power;

    B. heat fluxes (average and maximum);

    C. RCS pressure;

    D. minimum DNBR (PWR) or CPR (BWR);

    E. core and recirculation loop coolant flow rates (BWR);

    F. coolant conditions (inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam fractions);

    G. steam line pressure;

    H. containment pressure;

    I. pressure relief valve flow rate; and

    J. flow rate from the RCS to the containment system (if applicable).

    The more important of these parameters for the loss of ac power transient are compared to those predicted for other similar plants to confirm that they are within the expected range.

6. For reviews of DC and COL applications under 10 CFR Part 52, the reviewer should follow the above procedures to verify that the design set forth in the safety analysis report, and if applicable, site interface requirements meet the acceptance criteria. For DC applications, the reviewer should identify necessary COL action items. With respect to COL applications, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit, or other NRC-approved material, applications, and/or reports.

After this review, SRP Section 14.3 should be followed for the review of Tier I information for the design, including the postulated site parameters, interface criteria, and ITAAC.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) 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.

The staff concludes that the plant design as to transients expected to occur with moderate frequency and to result in the loss of all power to the station auxiliaries is acceptable and meets the relevant requirements of GDCs 10, 13, 15, and 26 and the applicable TMI Action Plan items. This conclusion is based on the following findings:

1. The applicant meets the requirements of GDCs 10 and 26 by demonstrating that resultant fuel integrity is maintained because the specified acceptable fuel design limits were not exceeded for the event.

2. The applicant meets GDC 13 requirements by demonstrating that all credited instrumentation was available, and that actuations of protection systems, automatic and manual, occurred at values of monitored parameters that were within the instruments' prescribed operating ranges.

3. The applicant meet GDC 15 requirements by demonstrating that the reactor coolant pressure boundary limits were not exceeded by this event and that resultant leakage is within acceptable limits. This requirements is met because the maximum pressure within the reactor coolant and main steam systems did not exceed 110 percent of the design pressure.

4. The applicant meets GDC 26 requirements for the capability of the reactivity control system to control reactivity adequately during this event with appropriate margin for stuck rods because the specified acceptable fuel design limits were not exceeded.

For DC and COL reviews, the findings will also summarize (to the extent that the review is not discussed in other SER sections) the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable, and interface requirements and combined license action items relevant to this SRP section.

V. IMPLEMENTATION

The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section, unless superceded by a later revision.

VI. REFERENCES

1. GDC 10, "Reactor Design."

2. GDC 13, “Instrumentation and Control.”

3. GDC 15, "Reactor Coolant System Design."

4. GDC 26, "Reactivity Control System Redundancy and Capability."

5. RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems."

6. RG 1.105, "Instrument Spans and Setpoints for Safety-Related Systems."

7. NUREG-75/103, Safety Evaluation Report, "RESAR-41 Standard Reference System," Westinghouse Electric Corporation, December 1975.

8. NUREG-75/112, Safety Evaluation Report, "CESSAR System 80, Standard Reference System," Combustion Engineering Incorporated, December 1975.

9. NUREG-0104, Safety Evaluation Report, "RESAR-35, Standard Reference System," Westinghouse Electric Corporation, December 1976.

10. NUREG-0151, Safety Evaluation Report, "GESSAR-251, Nuclear Steam Supply System Standard Design," General Electric Company, March 1977.

11. NUREG-0152, Safety Evaluation Report, "GESSAR-238, Nuclear Steam Supply System Standard Design," General Electric Company, March 1977.

12. NUREG-0433, Safety Evaluation Report, "B-SAR-205, Nuclear Steam Supply System," Babcock & Wilcox Company, May 1978.

13. NUREG-0491, Safety Evaluation Report, "RESAR-414 Standard Reference System," Westinghouse Electric Corporation, November 1978.

14. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses."

15. NUREG-0737, "Clarification of TMI Action Plan Requirements."

16. AP1000 Final Safety Evaluation Report, Chapter 15, Transient and Accident Analyses.
17. ESBWR Design Control Document, Chapter 15 - Safety Analyses.

18. NRC Inspection Manual Chapter IMC-2504, “Construction Inspection Program - Non-ITAAC Inspections,” issued April 25, 2006.

19. ANS Trial Use Standard N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," American Nuclear Society (1974).

20. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1974).

SRP Section 15.2.6
Description of Changes

This SRP section affirms the technical accuracy and adequacy of the guidance previously provided in Draft Revision 2, dated April 2006, of this SRP section. See ADAMS accession number ML061990647.

In addition, this SRP section was administratively updated in accordance with NRR Office Instruction LIC-200, Revision 1, "Standard Review Plan (SRP) Process." The revision also adds standard paragraphs to extend application of this updated SRP section to prospective applicant submissions pursuant to 10 CFR Part 52.

The technical changes are incorporated in Revision 2, dated [Month] 2007:

Review Responsibilities - Reflects changes in review branches resulting from reorganization and branch consolidation. Change is reflected throughout the SRP.

I. AREAS OF REVIEW

1. Reformatted the section with new numbering system. Incorporated reference to 10 CFR 52 from draft revision 1 - April 1996. Incorporated generic paragraphs relating to certified designs, ESPs, and COLs.

2. Added additional review scenario for Westinghouse-AP1000 plants, on a complete loss of the offsite power.

3. Reordered the items numbers with the section "Review Interfaces."

II. ACCEPTANCE CRITERIA

1. Reformatted the section with new numbering system. Incorporated reference to 10 CFR 52 from draft revision 1 - April 1996. Incorporated generic paragraphs relating to certified designs, ESPs, and COLs.

III. REVIEW PROCEDURES

1. Reformatted the section with new numbering system. Incorporated reference to 10 CFR 52 from draft revision 1 - April 1996. Incorporated generic paragraphs relating to certified designs, ESPs, and COLs.

2. Reformatted a few selective areas under this section time-Related variations of parameters monitored.

IV. EVALUATION FINDINGS

None.

V. IMPLEMENTATION

None.

VI. REFERENCES

References were updated and modified.