U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

15.4.4 – 15.4.5 STARTUP OF AN INACTIVE LOOP OR RECIRCULATION LOOP AT AN INCORRECT TEMPERATURE, AND FLOW CONTROLLER MALFUNCTION CAUSING AN INCREASE IN BWR CORE FLOW RATE

REVIEW RESPONSIBILITIES

Primary - Organization responsible for review of reactor systems transient and accident analyses for PWRs/BWRs

Secondary - None

I. AREAS OF REVIEW

A number of anticipated operational occurrences (AOOs) may cause either increased core flow or introduction of cooler or de-borated water into the core. These AOOs result in an increase in core reactivity due to decreased moderator temperature, moderator boron concentration, or core void fraction. This Standard Review Plan (SRP) section is intended to be applicable to all such AOOs. Each of these AOOs should be discussed in individual sections of the applicant's Safety Analysis Report (SAR) or Design Control Document (DCD), as specified in NRC Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" and Draft Regulatory Guide DG-1145, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

The specific area of review are as follows:

1. Boiling water reactor (BWR): Startup of an idle recirculation pump.

2. BWR: Flow controller malfunction causing increased recirculation flow.

3. Pressurized water reactor (PWR) with loop isolation valves: Startup of a pump in an initially isolated inactive reactor coolant loop where the rate of flow increase is limited by the rate at which the isolation valves open.

4. PWR without loop isolation valves: Startup of a pump in an inactive loop.

5. ESBWR: Abnormal startup sequence for an ESBWR.

6. 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, DCs 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.

The review of the core flow increase AOOs considers the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the AOOs. The reviewer concentrates on the need for the reactor protection system and operator action to secure and maintain the reactor in a safe condition.

The analytical methods are reviewed to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer initiates a generic evaluation of the new analytical model. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

The predicted results of the AOOs are reviewed to ensure that the consequences meet the acceptance criteria given in Subsection II below. Furthermore, the results of the AOOs are reviewed to ascertain that the values of pertinent system parameters are within ranges expected for the type and class of reactor under review.

Review Interfaces

The listed SRP sections interface with this section as follows:

1. The organization responsible for reviewing reactor systems also performs the following reviews under the SRP sections indicated:

   A. The organization responsible for reviewing reactor systems performs generic reviews under SRP Sections 4.2 through 4.4 of the thermal-hydraulic computer models used for these AOOs and also performs, upon request, additional analyses related to these accidents for selected reactor types as part of its primary review responsibility.

2. In addition, the organization will coordinate other organizations' evaluations that interface with the overall review of the system, as follows:

   A. The organization responsible for the review of instrumentation and control systems reviews the instrumentation and control aspects of the sequences described in the SAR or DCD to confirm that reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis as part of its primary review responsibility for SRP Sections 7.2 through 7.5.

   B. The organization responsible for emergency preparedness and radiation protection is notified regarding the extent of the fuel failures that are predicted by the analyses. That organization then evaluates the radiological consequences of the events.

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 and GDC 20, as they relate to the reactor coolant system being designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations and AOOs.

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 and GDC 28, as they relate to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations and AOOs.

4. GDC 26, as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded during AOOs. This is accomplished by ensuring that appropriate margins for malfunctions, such as stuck rods, is accounted for.

5. The basic objectives of the review of the AOOs described above are:

   A. To identify which of the AOOs are the most limiting.

   B. To verify that, for the most limiting AOOs, the plant responds in such a way that the criteria regarding fuel damage and system pressure are satisfied.

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.

Using the ANS standards as guidance, the specific criteria necessary to meet the relevant requirements of the regulations identified above for incidents of moderate frequency are as follows:

A. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values [3].

B. Fuel-cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95% probability/95% confidence DNBR limit for PWRs and the critical power ratio (CPR) remains above the minimum critical power ratio (MCPR) safety limit for BWRs, based on acceptable correlations (see SRP Section 4.4).

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

D. The requirements stated in Regulatory Guide 1.105, "Instrument Spans and Setpoints," are used with regard to their impact on the plant response to the type of AOOs addressed in this SRP section.

E. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR 50, shall be identified and assumed in the analysis and should satisfy the guidance stated in Regulatory Guide 1.53.

F. The guidance provided in SECY 77-439, SECY 94-084 and DG-1145 with respect to the consideration of the performance of non-safety related systems during transients and accidents, as well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems), must be evaluated and verified.

The applicant's analysis of the most limiting AOOs should be performed using an acceptable model. If analytical methods that have not been approved are proposed by the applicant, they are evaluated by the staff for acceptability.

The values of parameters used in the analytical model are to be suitably conservative. The following values are considered acceptable:

1. Initial power level is rated output (licensed core thermal power) for the number of loops initially assumed to be operating, plus an allowance of 2% to account for power measurement uncertainty, unless (a) a lower number can be justified through the

measurement uncertainty methodology and evaluation, or (b) unless the uncertainty has otherwise been accounted for (see SRP Section 4.4). An analysis to determine the effects of a flow increase must be made for each allowed mode of operation (i.e., one, two or three loops initially operating) or the effects referenced to a limiting case.

2. Conservative scram characteristics are assumed, e.g., maximum time delay with the most reactive rod held out of the core for a PWR and a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate for a BWR, unless (a) a different conservatism factor can be justified through the uncertainty methodology and evaluation, or (b) unless the uncertainty has otherwise been accounted for (see SRP Section 4.4).

3. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile and radial power distribution.

4. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with Regulatory Guide 1.105 as determined by the organization responsible for instrumentation and controls.

The reviewer shall verify that the protection system (1) automatically initiates the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded for this event, and (2) senses the plant conditions and initiates the operation of SSCs important to safety.

For BWR plants where flow control is part of the reactivity control system, GDCs 26 and 28 must be satisfied for this event; otherwise, GDCs 26 and 28 are not applicable. Where applicable, GDCs 26 and 28 are satisfied if compliance with GDCs 10 and 15 is demonstrated.

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. Compliance with GDC 10 requires that the reactor core and associated coolant, control and protection systems be designed with appropriate margin to ensure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs.

   GDC 10 is applicable to this SRP section because the reviewer evaluates the consequences of events associated with startup of an inactive loop or recirculation loop at an incorrect temperature and with a flow controller malfunction causing an increase in BWR core flow rate. This section, SRP Sections 4.2 through 4.4, and 7.2 through 7.5, and Regulatory Guides 1.53 and 1.105, provide guidance for ensuring that the reactor core, coolant, control and protection systems are designed with appropriate margin.

   Meeting the requirements of GDC 10 provides assurance that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs.

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 sequences of events, including automatic actuations or 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. Compliance with GDC 15 requires that the reactor coolant system and associated auxiliary, control and protection systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including AOOs.

   GDC 15 is applicable to this SRP section because the reviewer evaluates the consequences of events associated with startup of an inactive loop or recirculation loop at an incorrect temperature and with a flow controller malfunction causing an increase in BWR core flow rate. This section, SRP Sections 4.2 through 4.4, and 7.2 through 7.5, and Regulatory Guides 1.53 and 1.105, provide guidance ensuring that the reactor coolant system and associated auxiliary, control and protection systems are designed with appropriate margin.

   Meeting the requirements of GDC 15 provides assurance that the reactor coolant pressure boundary will not be breached during any condition of normal operation, including the effects of AOOs.

4. Compliance with GDC 20 requires that the protection system be designed (a) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs, and (b) to sense accident conditions and to initiate the operation of SSCs important to safety.

   GDC 20 is applicable to this section because the reviewer evaluates the consequences of the events associated with startup of an inactive loop or a recirculation loop at an incorrect temperature and with a flow controller malfunction causing an increase in BWR core flow rate. This section, SRP Sections 4.2 through 4.4, and 7.2 through 7.5, and Regulatory Guides 1.53 and 1.105, provide guidance for ensuring that the reactor coolant system is designed with appropriate margin. Thus, when the reactor protection system senses an accident condition, it will initiate the operation of SSCs important to safety to ensure that SAFDLs are not exceeded.

   Meeting the requirements of GDC 20 provides assurance that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs.

5. Compliance with GDC 26 requires that one of the reactivity control systems shall use control rods capable of reliably controlling reactivity changes to ensure that under conditions of normal operation including AOOs, and with appropriate margin for malfunctions such as stuck rods, SAFDLs are not exceeded.

   GDC 26 is applicable to this section because the reviewer evaluates the consequences of the events associated with startup of an inactive loop or a recirculation loop at an incorrect temperature and with a flow controller malfunction causing an increase in BWR core flow rate. This section, SRP Sections 15.4.4, 4.2 through 4.4, and 7.2 through 7.5, and Regulatory Guides 1.53 and 1.105, provide guidance for ensuring that the reactivity control system (control rods) is capable of reliably controlling reactivity changes with appropriate margin for malfunctions such as stuck rods.

Meeting the requirements of GDC 26 provides assurance that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs.

6. Compliance with GDC 28 requires that reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (a) result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor (b) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor temperature and pressure, and cold water addition. In addition, the introduction of hot water into the core of a PWR with a positive moderator temperature coefficient should be considered.

   GDC 28 is applicable to this section because the reviewer evaluates the consequences of the events associated with startup of an inactive loop or a recirculation loop at an incorrect temperature and with a flow controller malfunction causing an increase in BWR core flow rate. This section, SRP Sections 4.2 through 4.4, and 7.2 through 7.5, and Regulatory Guides 1.53 and 1.105, provide guidance for ensuring that the reactor coolant system and associated auxiliary, control and protection systems are designed with appropriate margin to ensure that the reactor coolant pressure boundary will not be breached.

   Meeting the requirements of GDC 28 provides assurance that the reactor coolant pressure boundary will not be breached during any condition of normal operation, including the effects of AOOs.

III. REVIEW PROCEDURES

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

The sequence of core flow increase events from initiation until a stabilized condition is reached is reviewed to ascertain:

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

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

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

4. The operation of engineered safety systems that are required.

5. The extent to which operator actions are required.

6. That appropriate margin for malfunctions, such as stuck rods, is accounted for.

7. That instrumentation uncertainties of system and operating parameters are appropriately accounted for.

If the SAR or DCD states that a particular core flow AOO is not as limiting as some other similar AOO, the reviewer evaluates the justification presented by the applicant. The applicant should present a quantitative analysis in the SAR or DCD of the increase in flow AOO that is determined to be most limiting. For this AOO, the reactor systems reviewer, with the aid of the instrumentation and controls reviewer, reviews the timing of the initiation of protection, engineered safety feature, and other systems needed to limit the consequences of the core flow increase AOO to acceptable levels. The reviewer compares the predicted variation of system parameters with various trip setpoints. The review of Chapter 7 of the SAR or the corresponding chapter of the DCD confirms that the instrumentation and control system design is consistent with the requirements for safety system actions for these events.

To the extent required, the reviewer evaluates the effect of single active failures of safety systems and components that may alter the course of the AOO. This phase of the review uses the system review procedures described in the SRP sections for Chapters 5, 6, 7, and 8 of the SAR or the corresponding chapters of the DCD. The reviewer considers and evaluates the possibility of a single failure that would permit the loop isolation valves to open prior to startup of a pump in an idle loop (for those plants with loop isolation valves). If this could occur, the core flow rate increase would not be limited by the rate at which the valve opens, and the resulting rate of reactivity insertion could be greater than for other AOOs of this group.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed to determine if these models have been previously reviewed and found acceptable by the staff. If not, a generic review of the model proposed by the applicant is initiated.

The values of system parameters and initial core conditions, including fuel data and system conditions used as input to the model, are reviewed. Of particular importance are the reactivity coefficients and control rod worths used in the applicant's analysis, and the variation of moderator temperature, void and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that the selected core burnup yields the minimum margins is evaluated.

The reviewer should review SAR Section 4.4 or the corresponding chapter of the DCD to evaluate how uncertainties of the input parameters are applied in the CPR and DNBR analyses.

The results of the analysis are reviewed and compared with the acceptance criteria presented in Subsection II of this SRP section regarding the maximum pressure in the reactor coolant and main steam systems, as well as minimum DNBR (PWR) or MCPR (BWR, if applicable). Time-related variations of the following parameters should be reviewed for consistency:

- reactor power;
- heat fluxes (average and maximum);
- reactor coolant system pressure;
- core and recirculation loop coolant flow rates (BWR, if applicable);
- 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 (BWR);
- pressure relief valve flow rate and quality; and
- flow rate from the reactor coolant system to the containment system (if applicable).

The values of the more important of these parameters for the core flow increase AOOs are compared with those predicted for other similar plants to see that they are within the range expected.

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 combined license action items. With respect to COL applications, the scope of the review is dependent on whether the COL applicant references a DC, an ESP 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.

A number of plant AOOs can result in a core flow increase. Those that might be expected to occur with moderate frequency are the startup of an idle recirculation pump (BWR, if applicable), flow controller malfunction causing increasing core flow (BWR), startup of a pump in an inactive reactor coolant loop (PWR), and startup of a pump in an initially isolated inactive reactor coolant pump loop. All these postulated AOOs have been reviewed. It was found that the most limiting with regard to core thermal margins and pressure within the reactor coolant and main steam systems was the AOO. This AOO was analyzed by the applicant using a mathematical model that has been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative.

The staff concludes that the plant design with regard to AOOs that result in an increase in coolant flow through the reactor core is acceptable and meets the relevant requirements of GDCs 10, 15, 20, 26 and 28. This conclusion is based on the following:

1. The applicant has met the requirements of GDCs 10, 20 and 26 with respect to demonstrating that SAFDLs are not exceeded for this 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 has met the requirements of GDCs 15 and 28 with respect to ensuring that the design conditions of the reactor coolant pressure boundary are not exceeded because the protection system operates to maintain the maximum pressure within the reactor coolant and main steam system pressures below 110% of the design values.

4. The applicant has met the positions of Regulatory Guide 1.53, SECY 77-439, SECY 94-084 and DG-1145 as related to the single-failure criterion and Regulatory Guide 1.105 as related to instrument actuations of SSCs important to safety.

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 (DAC), as applicable, and interface requirements and combined license action items relevant to this SRP section.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

The staff will use this SRP section in performing safety evaluations of design certifications and license applications submitted by applicants pursuant to 10 CFR 50 or 52. Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of 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.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides.

VI. REFERENCES

1. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

2. Draft Regulatory Guide DG-1145, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

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

4. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

5. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

6. General Design Criterion 10, "Reactor Design."

7. General Design Criterion 13, “Instrumentation and Control.”

8. General Design Criterion 15, "Reactor Coolant System Design."

9. General Design Criterion 20, "Protection System Functions."

10. General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

11. General Design Criterion 28, "Reactivity Limits."

12. Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems."

13. Regulatory Guide 1.105, "Instrument Spans and Setpoints."

14. ANSI/ANS 51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," (replaced ANSI N18.2-1974, reaffirmed 1988, withdrawn 1998).

15. ANSI/ANS 52.1-1983, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," (replaced ANS Trial Use Standard N212-1974, reaffirmed 1988, withdrawn 1998).

16. NUREG-1801, "Generic Aging Lessons Learned Report," Rev. 1, v.1-2.

17. SECY-77-439, "Single Failure Criterion."

18. SECY-94-084, "Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs."

19. NUREG-0933, "A Prioritization of Generic Safety Issues."

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

SRP Section 15.4.4 - 15.4.5
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 ML061930469.

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 Part 52 from draft revision 1 - April 1996. Incorporated generic paragraphs relating to certified designs, ESPs, and COLs.

2. Added additional stipulation to follow review guidance contained in DCDs such RG 1.70 or DG-1145.

3. Added additional area of review for abnormal startup sequence for an ESBWR

II. ACCEPTANCE CRITERIA

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

2. Added stipulation to follow review guidance contained in DCDs such RG 1.70 to account for induce fuel failures above those expected from normal operational occurrences (item 7)

3. Under “Specific Acceptance Criteria“ added additional stipulation to follow review guidance provided in SECY and DG-1145 (item G)

4. Under “Specific Acceptance Criteria“ Item 1 & 2 revised to include uncertainty methodology.

III. REVIEW PROCEDURES

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

2. Added Instrument uncertainties (item 7) .

IV. EVALUATION FINDINGS

Added a new paragraph for COL review staff evaluation of ITAAC.

V. IMPLEMENTATION

None.

VI. REFERENCES

References were updated and modified.