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TECHNICAL EVALUATION REPORT FOR REVIEW OF FORT ST. VRAIN TECHNICAL SPECIFICATION UPGRADE PROGRAM

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Project: Selected Operating Reactors Issues - Technical Assistance for Fort St. Vrain (FSV) - Support for FSV Technical Specification Upgrade Review (FIN No. A9478, Project 1, Task 8-4)

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1. INTRODUCTION AND BACKGROUND

This Technical Evaluation Report (TER) documents the results of work performed by Oak Ridge National Laboratory (ORNL) in providing a technical evaluation of designated portions of the proposed Technical Specifications (TS)[1] for the Fort St. Vrain (FSV) Technical Specification Upgrade Program (TSUP). This effort has been performed under the direction of the U.S. Nuclear Regulatory Commission's (NRC's) Office of Nuclear Reactor Regulation (NRR) and has been coordinated with Idaho National Engineering Laboratory (INEL), which has prime responsibility for providing NRC with an integrated TER that documents the results of evaluations that are performed by both INEL and ORNL. Therefore, the format of this TER will follow that developed in cooperation with INEL and will contain all numbered sections to be included in the final integrated TER except for the TS evaluation subsections in Section 3 which will list only the ORNL contributions. The other sections and subsections to be written by INEL will be appropriately identified as such in this TER. This approach is being used to facilitate integration of the ORNL contributions into the final INEL integrated TER.

The ORNL contributions to the TSUP review and evaluation also include two separate TERs[2,3] that have respectively addressed in detail (1) the equipment redundancy requirements for accommodating the Rapid Depressurization Accident (RDA), or Design Basis Accident No. 2 (DBA-2), and (2) the acceptability of the TSUP proposals for accommodation of safety-related and important-to-safety cooling functions. As used for FSV in the TSUP review of cooling functions, important-to-safety refers to the use of equipment configurations that include nonsafety-related equipment and that provide the preferred or primary success path in responding to transients addressed in the FSV Updated Final Safety Analysis Report (FSAR)[4] but that, in each case, are also redundantly and diversely backed up by an alternate success path composed of seismically and environmentally qualified safety-related equipment. For the cooling functions that are addressed in the TSUP, the operability of both the safety-related and important-to-safety equipment (that is, equipment in both the primary and backup success paths) is assured by TS requirements. The conclusions of the ORNL TERs that addressed these separate issues will be summarized briefly in this TER.

The TSUP Sections or individual TSUP Specifications that were designated for ORNL review and evaluation include the following.

- o Specification 2.1, Safety Limits
- o Section 3/4.1, Reactivity Control Systems
- o Section 3/4.2, Core Irradiation, Temperature and Flow Limits
- o Specification 3/4.3.2.3, Seismic Instrumentation
- o Specification 3/4.3.2.4, Meteorological Instrumentation
- o Specification 3/4.3.2.7, Power-to-Flow Ratio Instrumentation System
- o Specification 3/4.3.2.8, Core Region Outlet Thermocouples
- o Section 3/4.4, Primary Coolant System
- o Section 3/4.5, Safe Shutdown Cooling Systems
- o Section 3/4.6, PCRV and Confinement Systems; that is, all of this section except for Specifications 3/4.6.1.5, Steam Generator Interspace Radiation Monitoring, and 3/4.6.5, Reactor Building Confinement
- o Specification 3/4.7.1.1, Boiler Feed Pumps
- o Specification 3/4.7.1.3, Pressure Relief Valves
- o Specification 3/4.7.1.5, Safety Valves - Operating
- o Specification 3/4.7.1.6, Safety Valves - Shutdown
- o Specification 3/4.7.1.7, Condensate Pumps
- o Specification 3/4.7.4, Service Water
- o Specification 3/4.7.5, Primary Coolant Depressurization
- o Specification 3/4.7.6.1, Spray and/or Sprinkler System
- o Specification 3/4.7.7, Fire Rated Barriers
- o Specification 3/4.10.1, Xenon Stability

In addition, this TER provides an alternatively worded evaluation for Specification 5.3.4, Reload Segment Design, since this specification is essentially a reformatting of the basis summary statement from an original FSV specification that has been upgraded in Section 3/4.1, Reactivity Control Systems. The subject specification represents an area of overlap in the ORNL and INEL evaluations.

1.1 Chronology of Major Steps in the Technical Evaluation Process

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1.2 Concerns of 1984 Assessment Report

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1.3 Ground Rules for Technical Specification Upgrade Program (TSUP)

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1.4 Redirection of Technical Specification Upgrade Program

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1.5 Highlights of Meetings and Correspondence on the TSUP

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1.6 Status of Out-of-Scope Items

[The July 1, 1988, draft by INEL of the introductory paragraph is rewritten as follows.]

Several TS items or issues have been classified to be outside the TSUP scope (Appendix A). These items or issues fall into two categories:

i) On-going or parallel efforts that have defined goals that will be incorporated into or otherwise modify the upgraded TS but as a separate final disposition.

ii) Issues or items that have been identified during the TSUP review and evaluation but that have been declared to be subject to potential future requests for additional information or to other action that is yet to be determined.

The areas that are being addressed in the first category of activities include: 1) plant protective system setpoints, 2) compliance with Generic Letters 83-36 and 83-37, 3) LCO 4.1.9, 4) Appendix R considerations on fire protection, and 5) the inservice inspection plan. Although these other issues involve changes to the TS, their evaluation is not restricted by the ground rules agreed to (see Section 1.3 of this report) for the TSUP. Therefore, even though these other areas involve TS changes, they are being pursued outside of but in parallel with the TSUP. In some cases, for example the Generic Letters 83-36 and 83-37 issues, the evaluation has been pursued in concert with the TS changes and upgrade material of the TSUP but the ground rules for the evaluation have been separate from the TSUP. Final disposition of these other areas will be separate from the TSUP. These other areas are only being described briefly here to clarify that they are not part of the TSUP and are being pursued under separate tasks. A summary status of each of these parallel efforts is given below.

The second category of out-of-scope items were generated in response to NRC review comments. These out-of-scope items are the Category F comments that were alluded to in Section 1.3 as being determined to be outside the ground rules of the TSUP but with further discussion possible. As indicated in Section 1.3, 50 comments were designated as being Category F from among the initial set of NRC comments on the TSUP final draft. Another 20 comments have been similarly designated from the review of other TS in efforts parallel to TSUP for a total of 70 Category F comments. The Category F comments were basically of two types. Most of these comments relate to the licensing bases for the existing TS either being unclear as g ven in the FSAR or being absent from both the FSAR and other referenceable design documentation. The other comments in the Category F classification relate to recent operating experience that is judged not to require the addition of TS at this time but may require reconsideration pending further experience and review. With regard to the first type of Category F comments, the scope of the TSUP did not specifically address the possibility that, when a comprehensive review of the FSAR and other design and licensing documentation is performed to establish completeness of the proposed TS, the result might be that some of the existing carryover TS would be found to lack an adequately documented licensing basis. Thus, the potential need for further FSAR revisions, which might involve additional analysis, was found to be beyond the intended scope of the TSUP. A summary of primary issues relating to the clarity and existence of the TS licensing bases is given below.

1.6.1 Plant Protective System Setpoints

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1.6.2 Generic Letters 83-36 and 83-37

This section is to be written by INEL.

1.6.3 LCO 4.1.9

This section is to be written by INEL.

1.6.4 Appendix R, Fire Protection

This section is to be written by INEL.

1.6.5 Inservice Inspection

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1.6.6 Category F Comments Relating to Technical Specification Bases

During the TSUP review, 70 comments were generated and subsequently designated as Category F, that is, defined as indicating further discussion possible (but not in the scope of the TSUP). The majority of these Category F comments relates to the lack of clarity in or absence of the detailed licensing bases in the FSAR as required to support the existing TS per 10 CFR Parts 50.34(a)(5) and (b), 50.36(b), and 50.71(e). Although there were four items in the TSUP scope (P-3, P-11, N-2, and N-4) that relate to assuring the clarity, completeness, and accuracy of the upgraded TS compared against the FSAR and other documentation forming the licensing and technical basis, the NRC judged that the reviewer comments that relate to the clarity, completeness, and accuracy of the bases of carryover Specifications are out-of-scope so as to facilitate completion of the TSUP in a timely manner. Thus, for purposes of the TSUP, the upgrading of the existing TS bases was restricted to format and structural issues of the "summary statement of the bases or reasons for such specifications" as required per 10 CFR Part 50.36(a) and was interpreted not to apply to the more substantive documentation requirements for the FSAR.

The issue of clarity, completeness, and accuracy of the detailed bases for the carryover Specifications stems from the TSUP-chartered review of two sources: namely, (1) the applicant's safety analyses supporting TS in the original FSV operating license that was approved by the U.S. Atomic Energy Commission (AEC) and (2) the licensee's safety analyses for subsequent license amendments that have been approved by either the AEC or the NRC. Both sources should be documented in the current Updated FSAR.

TS Carryover from Original FSV License. First, with regard to safety analysis deriving from the original FSAR and FSAR amendments, it is pertinent to recall that FSV licensing reviews for the construction permit and operating license were conducted during a period of time between 1966 and 1973 in which the current requirements for the correlation of TS with detailed analysis and evaluation in the FSAR were being formulated and initially applied to plant licensing by the AEC. As initially proposed in August 1966 (Ref. 5) and finally promulgated in December 1968 (Ref. 6), all applicants for plant licenses that were issued a construction permit prior to January 16, 1969, were given the choice of producing TS under the 1962 rule[7] or under the 1968 rule.[6] As indicated by the initially proposed FSV TS in Attachment F to Amendment No. 15 of the FSV Preliminary Safety Analysis Report (PSAR),[8] the applicant for FSV elected to comply with the 1968 rule per the provisions of 10 CFR Part 50.36(d)(2) as instituted in the 1968 rule change.

Per 10 CFR Part 50.34(a)(5) of the 1968 rule change, the applicant at FSV was exempted from updating the PSAR supporting the application for the construction permit. Under the 1968 rule, new PSARs were to provide "an identification and justification for the selection of those variables, conditions, or other items which are determined as the result of preliminary safety analysis and evaluation to be probable subjects of technical specifications for the facility."

In electing to comply with the provisions of 10 CFR Part 50.36(d)(2) with regard to adhering to the 1968 rule on TS, the applicant for FSV was obligated to comply with the 1968 issuance of 10 CFR Parts 50.36(b) and 50.34(b), (1) through (6), as interpreted from the AEC statements of considerations that accompanied the announcement of the 1968 rule changes.[6] However, the AEC review of the FSV operating license application did not specifically require the applicant for FSV to provide the additional information required by the cited regulations as evidenced by the AEC's findings in Section 7.0 and elsewhere of the 1972 Safety Evaluation.[9] This situation occurred because the docket record indicates that the AEC staff required substantial changes to the initially proposed TS[8], and thus the AEC staff is judged to have viewed the final TS proposal and approval as being subject to the statement in 10CFR Part 50.36(b) that stipulates that "the Commission may include such additional technical specifications as the Commission finds appropriate." This provision implies discretionary prerogatives on the part of the Commission with regard to the TS, and such prerogatives were more generally and freely interpreted during the earlier periods of plant licensing than is currently the case. Also, as evidenced in the wording of the AEC's safety evaluations of FSV, the AEC licensing staff recognized the enhanced inherent safety features of the FSV ceramic core, the coated fuel particles, and the prestressed concrete reactor vessel (PCRV).

Nevertheless, per 10 CFR Part 50.34(b) and consistent with the AEC statement of considerations accompanying the 1968 rule change, the applicant for FSV was to provide information in the FSAR to support the approval of the operating license and to include such TS items [that is, as identified in Part 50.36(c)] as: (1) the facility "design bases [defined per Part 50.2(u) in the 1968 regulation] and the limits on its operation," (2) "surveillance and periodic testing of structures, systems, and components," (3) "a description and analysis of structures, systems, and components of the facility...to show that safety functions will be accomplished...[and]...sufficient to permit understanding of system designs and their relationship to safety evaluations" [that is, interpreted to include primary success paths, assumed initial conditions for transients and accidents, and the final information on design features as required in the PSAR per Part 50.34(a)(3)(iii) to satisfy Part 50.36(b) and (c)(4)], and (4) "managerial and administrative controls." Also, per 10 CFR Part 50.36(b) and consistent with the AEC statement of considerations, the "analyses and evaluation included in the" FSAR and FSAR amendments for deriving TS and supporting approval of the operating license are to be substantial enough to provide the final "identification and justification for the selection of those variables, conditions, or other items which are determined as the result of...[final] safety analysis and evaluation to be...subjects of technical specifications for the facility" as required in the PSAR per Part 50.34(a)(5). These interpretations of the 1968 regulations are consistent with the accompanying AEC statement of considerations that indicated that "the analysis and evaluation of the facility under [Section] 50.34 must provide (1) the necessary information from which technical specifications will be derived, and (2) the detailed bases for the specifications derived."

In addition to announcing the 1968-final rule changes, the AEC issued guidelines for the content of TS[10] and for the organization and content of safety analysis reports.[11] The guidelines for safety analysis reports were initially announced concurrent with the 1966-proposed rule changes, and the applicant for FSV adhered to the organization guidelines in both the PSAR and FSAR submittals. The guidelines for content of TS also provided guidelines for the level of detail expected in the safety analysis report documentation of bases for TS (see pp. 11 and 26, Ref. 10). However, contrary to the regulatory requirements discussed above as well as to the regulatory guidelines for contents of the safety analysis reports as applicable at the time of FSV licensing, the applicant for FSV did not provide detailed documentation in the FSAR for the basis of every technical specification that was proposed and approved in the initial FSV license. As indicated above, this situation is judged to exist because of the AEC staff interpretation of the Commission's prerogatives that are implied to be discretionary per the last sentence in 10CFR Part 50.36(b). Although the cited regulations, statements of considerations, and guidelines promulgated by the AEC were reasonably explicit, it must be recognized that the FSV FSAR both was written and submitted by the applicant and was reviewed substantially by the AEC prior to the issuance of more detailed guidance as contained in (1) the November 1970 issuance of the "Safety Guide" series, (2) the February 1971 issuance of the current general design criteria (GDC), (3) the November 1971 issuance of the "Information Guide" series, (4) the November 1972 combining of the Safety and Information Guides into the Regulatory Guide series and the concurrent issuance of revised format and content guidance for safety analysis reports, and (5) the subsequent development of the standard review plans that are based on using the 1971 set of GDC and the Regulatory Guides to derive acceptance criteria for plant licensing. Because of the nascent stage of the licensing process under which the original FSV FSAR and TS were reviewed and given the acceptance and approval of the affected existing TS by the AEC's 1972 Safety Evaluation, the provisions of 10CFR Part 50.100 are judged not to be applicable to the Category F comments on the affected TS.

TS Carryover from Subsequent Amendments. Second, since the issuance of the FSV operating license, 59 amendments have been issued for that license. During the TSUP review, certain technical specifications modified by or originated in those amendments were also found to lack a documented basis in the FSV FSAR. Per the NRC statements of considerations accompanying the proposed and final rule changes affecting 10 CFR Part 50.71(e) (Refs. 12 and 13), the safety analyses and evaluation performed by the licensee to support license amendments are to be documented in the Updated FSAR, but no new analyses are to be required. In several cases for the FSV TS, referenceable documentation of the safety analyses supporting license amendments could not be found, so it is unclear as to whether a "new analysis" is at issue per the intent of the NRC statements of considerations or whether, for the affected TS, the AEC or the NRC staff made similar interpretations of the Commission's prerogatives that are implied to be discretionary per the last sentence in 10CFR Part 50.36(b).

TS Information Needs. As indicated previously, the NRC staff overseeing the TSUP directed that the need for FSAR updating as identified in the Category F comments was beyond the viable scope of the TSUP and that such information needs may be addressed in the future as potential requests for additional information. Current operating experience at FSV indicates that the plant is operating safely, and the larger thermal margins and longer thermal response times inherent in the FSV ceramic core as compared to light water reactors provides confidence that the TSUP can be completed and the upgraded TS implemented without requiring prior resolution of the Category F comments. The upgraded specifications that lack a detailed, quality-assured, documented licensing basis in the FSAR or elsewhere are listed as follows with a summary statement of the needed additional information:

o Specification 2.1.1, Reactor Core Safety Limit, with respect to verification of assumed axial power distributions as a function of long term burnup effects. (The FSAR does not contain a description either of the analytical methods used for FSV nuclear design or of the experimental basis for the verification of those methods as expected per the guidelines of Section III.B(2)(b)1 of Ref. 11 and per the guidelines of Ref. 10 for detailed analysis and evaluation of TS bases. This situation applies to all specifications relating to nuclear design parameters, analyses, and assumptions as evaluated by ORNL.)

- Specification 3.0.5, Limiting Conditions for Operation, with respect to detailed documentation of the experimental verification of the accuracy of the Calculated Bulk Core Temperature calculation although the conservatism of this calculation is judged to be adequate.
- Specification 3/4.1.3, Shutdown Margin, with respect to the SHUTDOWN MARGIN assessment methodology for the initial and reload cycles as verified and applied in the determinations for meeting surveillance requirements.
- Specification 3/4.1.4.1, Control Rod Pair Position and Worth Requirements, with respect to control rod worth calculations and calculational methods for the initial and reload cycles.
- Specification 3/4.1.5, Reactivity Change with Temperature, with respect to calculations of and calculational methods for the temperature coefficient of reactivity in the initial and reload cycles, particularly, as affecting the use of such calculations to extrapolate beginning-of-cycle surveillance results to end-of-cycle core conditions.
- Specification 3/4.1.7, Reactivity Status, with respect to methods and data used in the generation, application and verification of the base reactivity curve to assess potential loss of SHUTDOWN MARGIN.
- Specification 3/4.2.3, Comparison Regions, with respect to the exact methodology for inferring "measured region peaking factors" and to the verification of calculations and calculational models used for generating the "calculated region peaking factors" that are used in the surveillances.
- Specification 3/4.2.1, Core Irradiation, with respect to the lack of

the documented basis for correlating FSAR Chapter 3 burnup limits (used in the specification) to the fluence and FIMA limits quoted in FSAR Section 3.8.1.2 and Appendix A.2. (This correlation should be clarified.)

- Specification 3/4.4.1, Primary Coolant Loops and Coolant Circulation, with respect to detailed quantitative analysis and evaluation of the alternate mechanisms for ensuring forced circulation cooling of the reactor and for which qualitative credit is taken in FSAR Sections 6.3 and 10.3 and FSAR Appendices C.41, C.44, C.46, and C.47 (for example, use of the auxiliary boiler feed pumps for backup supply of emergency condensate).
- Section 3/4.5, Safe Shutdown Cooling Systems, with respect to secondary coolant flow predictions that are based on a collection of inadequately documented, contractor-generated, project-quality calculational models with no known documented verification against experimental data. (There is a lack of licensee-controlled, production-quality methods and models with detailed documentation and verification for secondary coolant flow for safe shutdown cooling.)
- Specification 3/4.6.1.4, PCRV Closure Leakage, with respect to the expected leakage rates given in the FSAR being based on an incorrect extrapolation methodology to PCRV operating pressures using leak test data from weld inspections at near atmospheric pressure. (These should be clarified but are not expected to be safety significant.)
- Specification 3/4.10.1, Xenon Stability, with respect to calculational methods and their verification/validation and with respect to the applicability of Regulatory Guide 1.68 (Ref. 14) to establishing acceptance criteria for startup testing.
- Specification 5.3.4, Reload Segment Design, with respect to calculational methods and their verification/validation.
- Specification 5.4.1, Criticality (Fuel Storage), with respect to calculational methods and their verification/validation.

Rationale for Accepting TS. As discussed in the evaluation of the above-listed specifications as given in Section 3 of this TER, the implementation of specifications that utilize or rely upon nuclear design calculations that are not described and verified in the FSAR is subject to review, audit, and, in some cases, approval by the FSV Nuclear Facility Safety Committee (NFSC) under the provisions of TS Administrative Controls (ACs). Pending an NRC formal re~ iest for additional information to assure the adequacy of undocumented or inadequately documented TS bases per the requirements for detailed analysis and evaluation under 10CFR Parts 50.34(a)(5) and (b), 50.36(b), and 50.71(e), the NFSC activity per the upgraded TS is judged to be acceptable for assuring both quality in design control per Part 50, Appendix B, and adequate record keeping per Parts 50.59(b)(3) and 50.71(a) and also to be subject to inspection per Part 50.70(a). In addition to the larger thermal margins and longer thermal response times inherent in ensuring the safety of the FSV ceramic core, compliance with the latter regulations governing NFSC activity is judged to be an additional basis for accepting, at this time, the upgraded specifications that lack a clearly documented licensing basis; however, in the interim, it is also judged that NRC should ensure that the NFSC reviews, audits, and bases for approval adhere to the intent of the NRC and industry guidelines in Refs. 14 through 18 for good engineering practice in core reload design.

2. EVALUATION METHODOLOGY

2.1 Summary of Methodology

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2.1.1 New Versus Old Mode Definitions

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2.1.2 760°F Calculated Bulk Core Temperature Concept

This section is being written by INEL.

2.1.3 Reviews Done of FSAR Revisions

This section is being written by INEL.

2.1.4 Relationship of TSUP to TS Improvement Project

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2.1.5 Safety Related Cooling Functions

The cooling function technical specifications have been reviewed in a separate TER[3] prepared by ORNL. The acceptability of the proposed upgraded Technical Specifications for the FSV cooling functions is established using a methodology that is described in detail in the ORNL TER. Previous effort by the NRC staff had emphasized establishing the acceptability of the proposed upgraded Specifications based solely on showing the consistency of the proposed revisions with the FSV FSAR and with the existing FSV Specifications while using the Westinghouse Standard Technical Specifications (W-STS, Ref. 19) as general guidance especially with regard to format. However, the comprehensive review and evaluation methodology implemented by ORNL uses the W-STS to establish a more logical and focused framework for assessing and evaluating the completeness and adequacy of the proposed FSV Specifications. The need for focus is necessitated in part because the FSV FSAR (the licensing basis from which the TS are drawn) often lacks precision and clarity as to the functional significance of structures, systems, and components (SSCs) that perform cooling functions. This is because the original FSAR was written under early (1966) emerging guidelines for content. As discussed in Section 1.6.6 of this TER, the early guidelines portend but do not specifically reflect the level of consistency currently required between TS and the supporting safety analysis report. Thus, the W-STS has been used as a guide first to identify generic cooling functions and then to assess and evaluate how the FSV FSAR has addressed each function and whether the proposed FSV Specifications are consistent with the licensing basis in the FSV FSAR. As discussed in Sections 2 and 3 of the ORNL TER, the ORNL methodology is judged to be consistent with the intent and objectives reflected in the AEC's statements of considerations that accompanied the rulemaking for the regulations that governed the initial FSV licensing and the development of the existing FSV Specifications. However, as also discussed, the ORNL methodology executes the assessment using current regulatory guidelines while recognizing that the FSV license was, in most cases, formulated and approved prior to the development and implementation of the most current applicable regulations and regulatory guidelines. Key steps in the ORNL methodology are listed as follows:

- Identify a set of generic cooling functions that are cited as being important-to-safety in the General Design Criteria (GDC) for Nuclear Power Plants per Appendix A, Part 50 to Title 10 of the Code of Federal Regulations (that is, 10 CFR Part 50, Appendix A).
- Correlate the list of generic cooling functions with both:
  - the W-STS coverage of light water reactor (LWR) SSCs that are required to effect the generic cooling functions, and
  - the acceptance criteria for LWR SSCs that perform such cooling functions as discussed in the LWR Standard Review Plan (SRP, Ref. 20).
- Using the correlated list of generic cooling functions that are implemented in the W-STS, identify the proposed FSV Specifications that address the same cooling functions.
- Identify the similarities and the differences between the FSV Specifications and W-STS functional requirements including breadth and depth of coverage.
- Establish the technical and licensing basis for differences between the FSV Specifications and W-STS based on the FSV FSAR.
- Review the FSV FSAR against both the existing and the proposed FSV TS to identify the licensing basis for unique specifications and the need for additional cooling function specifications due to unique functional requirements at FSV.
- Compare and review both the existing and the proposed FSV TS to assure completeness and correctness.

Results of the ORNL evaluation are summarized in Section 3 of this TER with regard to each specification that addresses a cooling function. A detailed evaluation of the cooling function specifications is provided in Section 4 of the separate ORNL TER (Ref. 3).

2.1.6 Reevaluation of Design Basis Accident No. 2 (Rapid Depressurization)

A separate TER[2] has been prepared by ORNL reviewing the acceptability of helium circulator redundancy requirements to accommodate the Rapid Depressurization Accident (RDA) at FSV. This section summarizes the nature of the problem addressed and conclusions reached in the ORNL TER.

For a loss of normal shutdown cooling at FSV, the FSV reactor is assumed to remain pressurized with redundant trains of the seismically and environmentally qualified Safe Shutdown Cooling System available to restore forced circulation cooling. For the Safe Shutdown Cooling System, the proposed upgrade of the FSV TS requires no more than one of the two helium circulators in each of the two primary cooling loops to be OPERABLE on water turbine drive when the reactor is operating above 5% of rated thermal power or has significant levels of decay heat as determined by the magnitude (>760°F) of the Calculated Bulk Core Temperature. The Economizer-Evaporator Superheater (EES) section of each of the two steam generators is also required to be OPERABLE under the same conditions so that two OPERABLE cooling loops exist each consisting of at least one OPERABLE helium circulator and one OPERABLE EES Section. These requirements assure minimum redundancy in the structures and components of the Safe Shutdown Cooling System that interface with the primary coolant.

However, to assure adequate core cooling following a RDA, which is described in FSAR Section 14.11 as the Design Basis Accident No. 2 (or DBA-2), analysis has shown that two helium circulators operating at a speed of 8000 rpm are required to prevent fuel damage in the depressurized core following prolonged operation of the reactor at 105% of rated thermal power. If reactor-generated steam is not available to drive the helium circulators, achieving 8000 rpm on two circulators requires the use of high pressure feedwater that is provided by at least one boiler feed pump, two of which must be OPERABLE at all times per the existing and proposed upgraded Technical Specifications. The boiler feedpumps and supporting equipment such as the condenser and condensate pumps are normally operating equipment that are not part of the Safe Shutdown Cooling System. As described in FSAR Section 14.11.2.2, no fuel damage is predicted to occur as long as feedwater drive of two helium circulators can be initiated within 60 minutes of the reactor depressurization and assuming no other cooling takes place within that time. The licensee for FSV presented an analysis (Attachment No. 1 to Ref. 21) to support a position that the occurrence frequency for a RDA is sufficiently low ($>10^{-9}$ per reactor-year) as to be incredible so that TS are not needed to require both helium circulators in each loop to be OPERABLE on water turbine drive.

The ORNL TER addresses the adequacy of the Fort St. Vrain design to provide forced cooling in the event of a RDA or RDA-equivalent event. The subject TER also documents a review of the occurrence frequency for a RDA and concludes that the frequency of the event could be as high as 3×10^{-5} per reactor-year instead of 10^{-9} per reactor-year as concluded by the licensee in Attachment No. 1 to Ref 21. This higher estimate was derived by applying to the integrity of the FSV PCRV penetration closures the same rigorous logic that has been used by the Advisory Committee on Reactor Safeguards to derive PWR vessel rupture probabilities in 1974 as documented in the Reactor Safety Study (WASH-1400) and by the United Kingdom's Central Electricity Generating Board in reviewing the basis for judging pressurized water reactor vessel integrity in the Sizewell-B Inquiry. Although the estimate of the occurrence frequency for the RDA was projected to be possibly much higher than predicted by the licensee, the frequency of core damage at FSV given that a RDA occurs is further estimated to be 2.5×10^{-3} per event based on analysis performed by Science Applications International Corporation (SAIC) under contract to ORNL and documented in Attachment 1 to the ORNL TER. Thus, the occurrence frequency for core damage due to a RDA or equivalent event is estimated to be about 7.5×10^{-8} per reactor-year.

Since core damage due to the RDA is a low probability event but the equipment employed to accommodate the RDA is not seismically and environmentally qualified, the consequences of having to rely on Class I equipment is also addressed in the ORNL TER. As documented in FSAR Appendix D.4, previous analyses have shown that for extended FSV operation at 35% power, there would be no significant fuel damage expected for a complete loss of forced cooling accident given the operation of redundant Class I Systems. In this case, the cooldown is due entirely to heat losses to one train of the PCRV liner cooling system (LCS), which is Class I and can be cooled by either of two diverse and redundant Class I systems (service water or firewater). Considering that the current 82% limitation on FSV operating power level would reduce the cooling needed to prevent fuel damage as compared to the 105% power

FSAR case, an independent ORNL analysis was made of the potential for core damage for intermediate scenarios for equilibrium operation between 35% and 105% of rated reactor power. In the case considered in the ORNL TER, only one seismically and environmentally qualified circulator with a Class 1E driver (boosted firewater) is assumed to be available for the cooldown of the depressurized reactor. This is a highly reliable system not dependent on offsite power. Thus, relying only on a single train of the Class I equipment, it was determined that there would be very little fuel damage (about 1%) expected for the RDA occurring for <82% power scenarios.

Based on the ORNL TER, the redundancy requirements for the helium circulator on water turbine drive were judged to be acceptable as provided in the proposed TS for the Safe Shutdown Cooling System. The operability of both helium circulators in the same loop on water drive is not required in the upgraded TS to accommodate the potential for a RDA. The RDA was determined to be a low probability event that can be redundantly and diversely accommodated by the Safe Shutdown Cooling System with acceptable dose consequences under the current FSV power restrictions.

2.1.7 Electrical Section Review

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2.2 Generic Evaluation

This section is being written by INEL.

2.2.1 Specification Carryovers From Existing TS

This section is being written by INEL.

2.2.2 Specifications Adapted From STS

This section is being written by INEL.

2.2.3 Specifications Having Additional Safety Analysis/Justification

This section is being written by INEL.

2.2.4 Specifications Affected by Conversion of Interim Amendments to TSUP Format

Most of this section is being written by INEL. The following amendments are evaluated by ORNL as discussed herein.

Amendment No.: 49

Current Technical Specification: LCO 4.9.3

TSUP Specification: 3/4.10.1

Description: Added Xenon Stability Testing.

TSUP Impact: Slight wording changes were made to Specification 3/4.10.1 as it appeared in the November 1985 draft. In addition, AC 6.5.2.9.d was added to denote NFSC approval authority for the engineering evaluation of expected power perturbations during the test.

Amendment No.: 51

Current Technical Specification: SR 5.2.8

TSUP Specification: SR 4.5.2.1

Description:

a. Operate Normal Bearing Water Makeup Pump in recycle mode once per 92 days.

b. Perform functional tests of Emergency Bearing Water Makeup Pump every 92 days.

c. Perform functional tests of Bearing Water Pumps and controls at each shutdown, and calibrate instruments annually.

TSUP Impact: The amendment is reflected in Specifications SR 4.5.2.1.a and SR 4.5.2.1.c. The instrumentation calibrations have not been included since, consistent with the W-STS definition of OPERABLE, the operability of attendant instrumentation is to be ensured via an administratively controlled calibration program.

Amendment No.: 51

Current Technical Specification: SR 5.2.9

TSUP Specification: SR 4.5.2.1

Description: Perform functional tests of bearing water accumulators and controls every 92 days, and calibrate annually.

TSUP Impact: The amendment is reflected in Specification SR 4.5.2.1.a.1 except that, consistent with the W-STS definition of OPERABLE, the operability of attendant instrumentation and controls is to be ensured via an administratively controlled calibration program.

Amendment No.: 51

Current Technical Specification: SR 5.2.10

TSUP Specification: SR 4.5.5.1

Description: Verify each fire pump develops at least 1425 gpm at a discharge pressure no less than 119 psig.

TSUP Impact: The amendment is reflected in Specification SR 4.5.5.1.d.2.

Amendment No.: 51

Current Technical Specification: SR 5.2.16

TSUP Specification: SR 4.6.4.3

Description: Perform leak tests of helium purification cooler well closures once per Refueling cycle, and test/calibrate associated instrumentation.

TSUP Impact: The amendment, including leak detection instrumentation, is reflected in Specification SR 4.6.4.3.d.

Amendment No.: 51

Current Technical Specification: SR 5.2.21

TSUP Specifications: SR 4.5.5, SR 4.6.2.1, SR 4.8.4

Description: Perform testing of valves and transfer switches that must be manually positioned for actuation of the ACM mode of operation.

TSUP Impact: ACM loads are addressed in Specification SR 4.8.4.e.2, and this surveillance will test the manual transfer switches but at a reduced frequency consistent with ACM operability testing. The testing of ACM valves will be addressed in the administrative controls for Safe Shutdown Cooling Valves (of which the ACM valves are a subset) that will be implemented under Specifications SR 4.5.5 and SR 4.6.2.1 as appropriate. PSC has provided a list of the Safe Shutdown Cooling Valves correlated with the TSUP surveillances for those valves that will be under administrative controls.

Amendment No.: 51

Current Technical Specification: SR 5.2.24.b

TSUP Specification: SR 4.5.5

Description: Perform monthly functional tests of circulating water makeup pumps and controls, annual performance/mechanical condition test, and annual calibration of instruments.

TSUP Impact: The monthly test is included in Specification SR 4.5.5.1.b.2, except that instruments and controls are addressed via administrative controls consistent with the W-STS definition of OPERABLE.

The annual performance test and condition monitoring is included in Specification SR 4.5.5.1.c.2.

Amendment No.: 51

Current Technical Specification: SR 5.2.24.d

TSUP Specification: SR 4.5.5

Description: Verify alignment and settlement of the circulating water makeup pond embankments once per 5 years.

TSUP Impact: This amendment is reflected in Specification SR 4.5.5.1.e.

Amendment No.: 51

Current Technical Specification: SR 5.2.24.e

TSUP Specification: SR 4.7.4

Description: Perform monthly functional tests on each service water pump and associated controls. Verify performance/mechanical condition annually and calibrate instruments.

TSUP Impact: This amendment is reflected in Specifications SR 4.7.4.1.b.2 and SR 4.7.4.1.c. Instrumentation is to be addressed via administrative controls consistent with the W-STS definition of OPERABLE.

Amendment No.: 51

Current Technical Specification: SR 5.2.24.f

TSUP Specification: SR 4.6.2.1

Description: Perform monthly functional tests on each reactor plant cooling water pump and controls. Annually verify performance and mechanical condition and calibrate instruments.

TSUP Impact: This amendment is reflected in Specifications SR 4.6.2.1.b.2 and SR 4.6.2.1.c. Instrumentation and controls are to be addressed via administrative controls consistent with the W-STS definition of OPERABLE.

Amendment No.: 51

Current Technical Specification: SR 5.2.24.g

TSUP Specification: SR 4.7.5

Description: Perform monthly functional tests of each purification cooling water pump and controls. Annually verify performance and calibrate instrumentation.

TSUP Impact: This amendment is reflected in Specifications SR 4.7.5.b and SR 4.7.5.c except that the operability of instrumentation and controls is to be ensured via administrative controls consistent with the W-STS definition of OPERABLE.

Amendment No.: 51

Current Technical Specification: SR 5.2.24.h

TSUP Specification: LCO/SR 3/4.6.4.3

Description: Perform testing of valves that are used for automatic isolation of purification cooling water system and reactor plant cooling water system to assure confinement integrity of PCRV interfacing structures (cooling tubes).

TSUP Impact: This amendment, which has been expanded to include a limiting condition for operation on PCRV integrity in terms of assuring confinement integrity afforded in interfacing structures, is reflected in Specifications LCO 3.6.4.3.b, SR 4.6.4.3.b, and SR 4.6.4.3.c.

Amendment No.: 51

Current Technical Specification: SR 5.3.4

TSUP Specifications: SR 4.5.1, SR 4.5.2, SR 4.5.3, SR 4.5.4, SR 4.5.5, SR

4.7.8

Description: Test Safe Shutdown Cooling Valves.

TSUP Impact: Consistent with the W-STS treatment of manual valves, the operability of manually actuated valves will be demonstrated via administrative controls for the associated systems. PSC has provided a correlation of the manual valves with the implementing surveillance. Safe Shutdown Cooling Valves actuated automatically by SLRDIS will be demonstrated operable via SR 4.7.8.

Amendment No.: 51

Current Technical Specification: SR 5.4.5

TSUP Specification: SR 4.6.2.1

Description: Perform functional check of the PCRV cooling system flow scanner, alarms, and flowmeters.

TSUP Impact: None. The provisions of SR 4.6.2.1 have been accepted as demonstrating an adequate basis for assuring PCRV liner cooling tube flow without reliance on the flow scanner.

Amendment No.: 55

Current Technical Specification: LCO 4.3.1

TSUP Specification: LCO/SR 3/4.5.3

Description: Deleted the reheater section of each steam generator from Safe Shutdown Cooling Equipment.

TSUP Impact: This amendment is reflected in Specification LCO/SR 3/4.5.3.

Amendment No.: 57

Current Technical Specifications: LCO 4.0.4, LCO 4.1.9, SR 5.1.8

TSUP Specifications: LCO 3.0.5, LCO/SR 3/4.2.4

Description: Revised minimum helium flow requirements as a function of the new concept of Calculated Bulk Core Temperature.

TSUP Impact: This amendment is reflected in Specifications LCO 3.0.5 and LCO/SR 3/4.2.4 but is also expanded to use the Calculated Bulk Core Temperature to demarcate the transition in redundancy requirements for safety-related and important-to-safety equipment.

2.3 Plant Specific Technical Specifications Evaluation Methodology

This section is being written by INEL.

3. EVALUATIONS

The introduction to this section is being written by INEL.

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

2.1.1 Reactor Core Safety Limit. (Specification carryover from existing TS) Although SL 2.1.1 is a carryover from existing Specification 3.1, the formulation of the safety limit has been modified (1) to be consistent with the manner in which the margin to the safety limit is tracked by plant personnel and (2) to place the required operator responses to transient-induced core power-to-flow imbalances in an LCO rather than in the safety limit specification. The latter modification is consistent with the format and structure of the STS and enhances the clarity of the safety limit itself as compared to the existing Specification that does not adequately distinguish the safety limit on the integral effects of single or multiple transients from the required operator response to deviations in observed plant parameters during a specific transient. The new separate Specification, LCO/SR 3/4.2.6, combines the implied limiting condition for operation that was included in existing SL 3.1 and the surveillance requirements for establishing the safety limit margin as given in existing Specification SR 5.1.6.

The thrust of the safety limit is to assure that fuel particles in each segment of fuel do not experience in a single transient or in multiple transients a limiting combination of high temperatures and prolonged positive thermal gradients (heat fluxes) that can be induced by core power-to-flow imbalances. If the core power-to-flow imbalance is large and lasts for too long a period of time, fuel element heat fluxes may be of sufficient magnitude and of sufficient integral time duration to cause the fuel kernel inside the fuel particle coating to migrate along the direction of the thermal gradient and through both the inner carbon buffer coating and the inner pyrolytic carbon shell surrounding the fuel kernel. Migration of the kernel through the first two coating layers of the particle would allow the kernel to chemically attack in succession the silicon carbide coating and the outer pyrolytic carbon coating and ultimately to fail the fuel particle coatings leading to fission product release. Since the effects of separate instances of kernel migration are cumulative, multiple migration-inducing transients of different power-to-flow ratios and of different time duration can lead to violation of the safety limit.

The analysis (FSAR Sections 3.6.7.6 and 3.6.8) supporting the safety limit and the LCO/SR has two principal results: namely, (1) the determination of the power level and power-to-flow ratio envelope (Figure 3.2.6-1) in which no kernel migration occurs even for indefinite periods of operation and (2) the determination of the time limits (Figure 3.2.6-2) for extended operation as a function of the power-to-flow ratios that induce kernel migration up to the point of penetrating the first two particle coating layers in a nominally manufactured fuel particle. These determinations have been made with conservative/bounding assumptions about steady state core conditions, such as incore power peaking and temperatures, and about the kernel migration rate as inferred from experimental data at the 95% confidence level.

The mechanism for establishing compliance with the safety limit is that the sum of the ratios of the time interval that each fuel segment experiences an operating condition that can induce kernel migration divided by the time it would take to violate the safety limit under that operating condition must be less than 1 for that fuel segment. The integral fraction of time is used since the time limit varies as a function of the power-to-flow ratio, which can differ with each transient. The ACTION for LCO 3.2.6 prescribes (1) the mechanism for identifying a potential migration-inducing transient, (2) the technique for calculating the ratio of the time in the transient divided by the time it would take the kernel to penetrate both inner particle coating layers during the transient, and (3) the requirement for summing the fractions from the current and previous transients to compare to the safety limit integral fraction of unity. By using this technique, the integral effect of all migration-inducing power-to-flow transients on the fuel particles in any segment of fuel is assured not to exceed that effect, which, using bounding assumptions on core conditions, can cause the kernel in the average particles to penetrate both the inner buffer layer and inner pyrolytic carbon layer of the particle coatings.

If a violation of the safety limit occurs, the reactor must be shut down within 24 hours, and the event reported to NRC which will determine if restart is allowed. The upgraded specification is judged to be acceptable.

2.1.2 Reactor Vessel Pressure. (Specification carryover from existing TS) This specification has not been modified significantly in the carryover except for restructuring to reflect the W-STS format. The basis statement has been rewritten to reflect the FSAR design bases for the PCRV and the PCRV penetrations, and a footnote has been added to the basis statement citing the more stringent pressure limits per Specification LCO 3.9.1 when the primary coolant boundary includes that of the PCRV-mounted fuel handling machines being used during incore fuel handling and refueling. The specification is judged to be acceptable.

3. LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.1 Reactivity Control Systems. (Section carryover from existing interim TS except for upgraded Specification 3/4.1.7.) The partial failure-to-scram event[22] of June 23, 1984, and the subsequent NRC audit report[23] of October 16, 1984, led to the development and implementation of an interim set of technical specifications for the FSV reactivity control system.[24-27] These Interim Specifications have been in effect since the summer of 1985, and one interim specification on reactivity change with temperature has also been revised on an interim basis.[28]

Because the FSV reactivity control system more closely resembles that of a boiling water reactor (BWR) as opposed to that of the pressurized water reactor (PWR), the provisions of the interim and upgraded FSV specifications often more closely resemble those of the General Electric BWR-STS (Ref. 29) as opposed to the W-STS (Ref. 19). Some of these parallels are addressed in the following evaluations.

3/4.1.1 Control Rod Pair Operability. (Specification carryover from existing Interim TS) This specification is a revised and upgraded carryover of the Interim Specification 3/4.1.1. The operability requirements for control rod pairs have been further clarified with respect to the Interim TS to specify that (1) the 152 second scram time for demonstrating operability is from the fully withdrawn position, (2) the helium purge flow shall not be carrying condensed water, and (3) there shall be an absence of a slack cable alarm. The ACTION statements for the Interim TS have been clarified as follows: (1) to require a SHUTDOWN MARGIN assessment whenever a shutdown is effected due to identifying immovable control rod pairs, (2) to allow continued operation with the temperatures of up to four control rod motors exceeding 250°F as long as periodic (24-hour) surveillances (partial scram tests) are performed to establish an acceptable scram time for operability, (3) to specify appropriate action in response to finding significant amounts of condensed moisture in the knock-out pot on the clean helium purge line to the control rod drives, and (4) to specify the maximum operator response times allowed for diagnosing the cause of a slack cable alarm (FSAR Sections 3.2.2.6, 3.8.1.1.1, and 3.8.2).

In addition to surveillance modifications that reflect changes made to the Interim TS LCO, the surveillances have also been modified slightly from the Interim TS to account for operating experience. Due to mechanical limitations, partial scram tests for fully inserted rod pairs can only be used to demonstrate that the rod pair is capable of being scrammed but not for showing an acceptable scram time. Once a rod pair is withdrawn from the fully inserted position, the operability requirement on scram time is to be demonstrated within 7 days per the SR.

In Attachment 1 to Ref. 30, the licensee has presented the rationale for not specifying a minimum purge flow for the surveillance at this time. The licensee indicates that assuring a positive value of net purge flow is sufficient at this time and that purge flow requirements will be included in the future resolution of outstanding commitments to NRC of integrated control rod drive operability issues. These issues include planned temperature requalifications of the control rod drive mechanism that will directly impact the magnitude of required purge flow.

The control rod operability specification is judged to be acceptable at this time.

3/4.1.2 Control Rod Pair Position Indication Systems

3/4.1.2.1 Control Rod Pair Position Indication System - Operating. (Specification carryover from existing interim TS) This specification is a reformatted version of Interim Specification LCO/SR 3/4.1.2 that has been revised to quantify the number of redundant and/or diverse indication mechanisms that are required to be operable at different control rod positions (FSAR Sections 3.8.1.1 and 7.2.2). Both the system requirements in the condition statement and the ACTIONs as given in the Interim TS have been reformatted into Specification Table 3.1.2-1 of the upgrade. The table provides the quantification of the required operable indication mechanisms. The surveillance has been reformatted slightly from the Interim TS with the rationale for certain surveillance actions to prevent potentiometer damage moved to the basis summary statements. The upgraded specification is judged to have an improved clarity in comparison to the Interim TS and to be acceptable.

3/4.1.2.2 Control Rod Pair Position Indication System - Shutdown. (Specification carryover from existing interim TS) This specification is a reformatted and revised version of Interim Specification LCO/SR 3/4.1.3. The revision quantifies the required operable rod position indication mechanisms for SHUTDOWN and REFUELING. The surveillance has been reformatted similar to that of the upgraded LCO/SR 3/4.1.2.1 discussed previously. The upgraded specification is judged to have an improved clarity in comparison to the Interim TS and to be acceptable.

3/4.1.3 Shutdown Margin. (Specification carryover from existing interim TS) This specification is a slight revision and reformatting of Interim Specification LCO/SR 3/4.1.4. The most significant revision is that, in the upgraded TS, the surveillance and the basis summary statement refer to the "assessment" of SHUTDOWN MARGIN as opposed to "verification" as used in the Interim TS. This change properly reflects the fact that, to comply with the surveillance, the operator must obtain the results of calculational analyses made off site and has no control over the verification process for these analyses (FSAR Sections 3.5.3.1 and 3.5.7.4 and Table 3.5-6, although the assessment methodology is not described in the FSAR). During SHUTDOWN and REFUELING, the operator is required to verify subcriticality from the startup detector count rates when control rod pairs are withdrawn to a position that is calculated to be worth .01 delta-k of rod withdrawal.

The ACTION statements in both the interim and upgraded versions of this TS are formatted after the example given by Specification 3/4.1.1 in the BWR-STS, but the surveillance for SHUTDOWN MARGIN determination more closely follows the example in SR 4.1.1.1 and SR 4.1.1.2 in the W-STS.

The assessment methodology for SHUTDOWN MARGIN is understood to be related to the methodology used for determining the base reactivity curve as applied in upgraded Specification LCO 3.1.7, Reactivity Status, which is discussed below. Further, the assessment methodology is understood to be related to that used for SHUTDOWN MARGIN analysis in the fuel segment reload design per Specification Design Feature (DF) 5.3.4. The reload segment design methodology is also used to establish the control rod pair withdrawal sequence per Specification DF 5.3.4. Both the base reactivity curve used in Specification LCO 3.1.7 and the control rod pair withdrawal sequence are approved by the FSV Nuclear Facility Safety Committee (NFSC) per Specification AC 6.5.2.9.a. The reload segment design, including associated SHUTDOWN MARGIN, is a safety significant change to the reactor core system that is subject to NFSC review per Specification AC 6.5.2.8.a and record keeping per 10 CFR Part 50.59(b)(3). Compliance with the "assessed" SHUTDOWN MARGIN as a TS limit is subject to NFSC audit per Specification AC 6.5.2.10.a as are associated quality assurance activities per Specification AC 6.5.2.10.d. Thus, the upgraded Specifications are judged to place the responsibility for tracking and assuring the validity of the SHUTDOWN MARGIN assessments made per Specification SR 4.1.3 directly under the cognizance of the NFSC.

The upgraded TS is judged to be explicit and thereby a significant improvement over the indirect coverage of SHUTDOWN MARGIN as implied in the existing former LCO 4.1.2 and existing LCO 4.1.8 and SR 5.1.4. The NFSC is judged to be directly responsible for the quality assurance of the SHUTDOWN MARGIN assessment. The TS is judged to be acceptable.

3/4.1.4 Control Rod Worth and Position Requirements

3/4.1.4.1 Control Rod Worth and Position Requirements - Operating. (Specification carryover from existing interim TS) This specification is essentially a reformatting of Interim Specification LCO/SR 3/4.1.5 and, with regard to control rod position requirements, is equivalent in intent to W-STS LCO/SR 3.1.3.6, Control Rod Insertion Limits. At FSV, unlike the Westinghouse PWR in which the rod bank withdrawal sequence remains unchanged from cycle to cycle, the control rod pair withdrawal sequence is specified uniquely as part of the nuclear design for each fuel cycle and is hard-wired into the control circuitry before startup of each fuel cycle (FSAR Sections 3.5.3.1 and 3.5.3.4 and Tables 3.5-6 through 3.5-8). Thus, there is no need for similar specifications as those for operability of the BWR rod sequence control system per BWR-STS LCO 3.1.4.2 since FSV cannot operate in STARTUP or at higher power levels without the equivalent system being both operable and operating. Also, because of the predetermination of the control rod pair withdrawal sequence and because the position limits imposed by FSV upgraded Specification LCO 3.1.4.1, there is no need for a rod worth minimizer as used at BWRs and as subjected to operability requirements per BWR-STS LCO/SR 3.1.4.1.

As discussed in the basis summary statements for FSV upgraded Specification LCO/SR 3/4.1.4.1, the control rod pair withdrawal sequence is determined in accordance with Specification DF 5.3.4, Reload Segment Design, and this determination is reviewed and approved by the FSV NFSC per Specifications AC 6.5.2.8.a and AC 6.5.2.9.a prior to startup of the core reload. The control rod pair position limits of upgraded LCO 3/4.1.4.1 are factored into the reload design to establish by analysis that power peaking, the temperature coefficient of reactivity and maximum control rod pair worth are acceptable and within the bounds of assumptions in the safety analysis. As indicated, the NFSC has cognizance over reload segment design review.

During startup, Specification SR 4.1.4.1.2 is performed to measure control rod pair group worths to verify that the calculated results are valid for assuring the maximum calculated control rod pair worths per the limits of LCO 3.1.4.1.c. The NFSC audits these comparisons per both AC 6.5.2.10.a and AC 6.5.2.10.d and is judged to be responsible for assuring this information is factored into the basis for calculating and approving the control rod pair withdrawal sequence for the next cycle. Given the NFSC oversight, the upgraded specification is judged to be acceptable.

<u>3/4.1.4.2 Control Rod Worth and Position Requirements - Shutdown</u>. (Specification carryover from existing interim TS) This specification is essentially a reformatting and slight rewording of Interim Specification LCO/SR 3/4.1.6. The major word changes reflect that SHUTDOWN MARGIN is "assessed" as opposed to "verified." The specification assures that during SHUTDOWN and REFUELING, control rod pairs are either fully inserted or, if not fully inserted, are positioned so as to maintain SHUTDOWN MARGIN. If SHUTDOWN MARGIN is not maintained by the positioning of control rod pairs, reserve shutdown material is to be inserted to restore SHUTDOWN MARGIN. The provisions of this specification supplement and complement those of upgraded LCO/SR 3/4.1.3, Shutdown Margin. The combination of these two specifications account for the unique reactivity control system configuration at FSV and provide a logical and comprehensive functional equivalent to W-STS LCO/SR 3/4.1.1.2, Shutdown Margin-Tavg $\leq 200^{o}F$, and BWR-STS LCO/SR 3/4.1.1, Shutdown Margin. Given the NFSC oversight of the SHUTDOWN MARGIN "assessment" methodology as discussed under the evaluation of upgraded Specification 3/4.1.3, the upgraded specification is judged to be acceptable.

<u>3/4.1.5 Reactivity Change with Temperature</u>. (Specification carryover from existing interim TS) This specification reflects a slight rewording of the Interim Specification LCO/SR 3/4.1.7. The Interim TS added the beginning of cycle limit on the magnitude of the temperature reactivity defect that was not addressed in the original FSV Specifications LCO 4.1.5 and SR 5.1.3. This is an important consideration in evaluating cold SHUTDOWN MARGIN at beginning of cycle when the positive reactivity contribution from reactor core cooldown

is greatest due to the large temperature reactivity defect.

Evaluation of the measured temperature coefficient of reactivity and its integral over temperature (that is, the reactivity defect) at beginning of cycle is also the basis for assuring that the temperature coefficient at end of cycle is acceptable (that is, at least as negative as the minimum power feedback coefficient assumed in the accident analysis). As indicated in the basis summary statement, the evaluation compares the measured and calculated values of the coefficient and defect at beginning of cycle and infers the effect on values at end of cycle. Per the evaluation of the plant general design criterion given in FSAR Appendix C.8, the power coefficient of reactivity, which is dominated by the temperature coefficient of reactivity (FSAR Section 3.5.5.1), must remain negative throughout core life. Per FSAR Tables 3.5-1, 3.5-4, and 3.5-9, the FSV temperature coefficient of reactivity is least negative at end-of-cycle, although the calculational methods and input data for producing these results are not described nor evaluated in the FSAR.

As indicated per Specification DF 5.3.4, Reload Segment Design, the sign and magnitude of the temperature coefficient is directly related to the allowable maximum control rod pair worth for the control rod pair withdrawal sequence of the reload cycle. The responsibilities and authority of the NFSC with regard to SHUTDOWN MARGIN, which is affected by the magnitude of the temperature defect as noted above, were discussed previously with regard to upgraded Specification SR 4.1.3. Per Specification AC 6.5.2.9.a, the NFSC approves the control rod pair withdrawal sequence and thus is expected to exercise direct cognizance over establishing the acceptability of the evaluation of the measured and calculated values of the temperature coefficient in assuring the acceptability of the withdrawal sequence throughout the reload cycle. Further, NFSC responsibilities with regard to verifying the adequacy of the calculation of the temperature coefficient and to establishing the traceability of current analytical methods to those used in the FSAR (Section 3.5.5.1, although not provided in detail) are evidenced in the review requirements per Specifications AC 6.5.2.8.a and AC 6.5.2.8.h and the audit requirements per Specifications AC 6.5.2.10.a and AC 6.5.2.10.d.

Given the understanding of the cognizance exercised by the NFSC in assuring the acceptability both of the calculated values of the temperature defect as affecting SHUTDOWN MARGIN and of the calculated values of the temperature coefficient as affecting the safety analysis of the reload segment design, upgraded FSV Specification LCO/SR 3/4.1.5 is judged to be acceptable.

<u>3/4.1.6 Reserve Shutdown System</u>

<u>3/4.1.6.1 Reserve Shutdown System - Operating</u>. (Specification carryover from existing interim TS) This specification is a slight reformatting and revision of Interim Specification SR/LCO 3/4.1.8, which is a substantial revision to the existing FSV Specifications LCO [illegible] and SR [illegible] that have been superseded by the Interim TS. During operation, operability is demonstrated by assuring the capabilities to burst the reserve shutdown hopper rupture disc, which is an action necessary to release the shutdown material into the core, and to provide a diverse backup mechanism to actuate the hopper pressurization valves in the event of loss of electrical power (FSAR Section 3.8.3). The reactivity worth of the reserve shutdown material is given in FSAR Section 3.5.3.3 and Table 3.5-1, although the analysis methods for nuclear design performance are not presented nor evaluated in the FSAR. Operability of the reserve shutdown system does not depend on the absence of condensed water in the purge gas because that is controlled per Specification LCO 3.1.1 for the operability of the control rod drives, each of which occupies the same penetration as the associated reserve shutdown hopper. However, since long-term exposure to condensed water may affect the insertability of the reserve shutdown material into the core, a surveillance of potentially affected hopper material is provided to test the dropout capability of hopper contents in an excore facility and this surveillance is performed in response to ACTION b.2 in Specification LCO 3.1.1.

FSV upgraded Specification LCO/SR 3/4.1.6.1 is judged to be functionally equivalent to the BWR-STS Specification LCO/SR 3/4.1.5, Standby Liquid Control System. The FSV reserve shutdown system has greater redundancy than the subject BWR system such that total system inoperability is not expected at FSV. The ACTION times are judged to be appropriate to the FSV system configuration. The specification is judged to be acceptable.

3/4.1.6.2 Reserve Shutdown System - Shutdown. (Specification carryover from existing interim TS) This specification is a slight reformatting and revision of Interim Specification SR/LCO 3/4.1.9. An important wording change is that the wording in the ACTION has been changed to "assess" SHUTDOWN MARGIN consistent with changes made to upgraded Specification SR 4.1.3 from the Interim TS. Given NFSC oversight of SHUTDOWN MARGIN "assessments," the specification is judged to be acceptable for the same reasons as given above for Specification LCO/SR 3/4.1.6.1.

3/4.1.7 Reactivity Status. (Specification carryover from existing TS) This specification is a reformatting of existing Specifications LCO 4.1.8 and SR 5.1.4. The specification tracks the difference between the observed and expected (calculated) core reactivity by use of the cycle-dependent base reactivity curve. The reactivity difference cannot exceed 0.01 delta-k, which is the minimum required cold SHUTDOWN MARGIN per Specification LCO 3.1.3. Although not described in the FSAR, the cycle-dependent base reactivity curve and changes thereto are reviewed and approved by the NFSC per Specification AC 6.5.2.9.b. Other NFSC responsibilities with regard to assuring the adequacy of the base reactivity curve for each reload cycle are also indicated with regard to the review of safety significant design changes per Specifications AC 6.5.2.8.a and AC 6.5.2.8.h and with regard to audits per Specifications AC 6.5.2.10.a, AC 6.5.2.10.c, and AC 6.5.2.10.d.

The carryover specification is judged to be functionally equivalent in intent to BWR-STS Specification LCO/SR 3/4.1.2, Reactivity Anomalies, in that both specifications use observed core conditions to indirectly assess the potential effect on SHUTDOWN MARGIN as indicated in the basis summary statements of the respective specifications. At FSV, the periodic (once per 7 days) tracking of the difference between observed and expected reactivity can also be indicative of other long term burnup effects such as changes in control rod worth, temperature coefficient of reactivity, and axial power distribution. Since none of these effects are otherwise observed directly throughout the reload cycle burnup, the reactor operations staff and the NFSC are expected to examine trends in the reactivity deviations as part of good engineering practice[14-18] in effecting the safe operation of the facility. This expectation occurs independent of the basis summary statement for this TS since the NFSC is judged to have this responsibility per the provisions of the above-cited Specification ACs.

Given the NFSC approval authority and the expected degree of NFSC oversight, the specification is judged to be acceptable.

3/4.2 Core Irradiation, Temperature, and Flow Limits

3/4.2.1 Core Irradiation. (Specification carryover from existing TS) This specification is a substantial revision of existing Specification LCO 4.1.1. A surveillance requirement has been added and an exposure limit on control rods has been included in the LCO (FSAR Section 3.8.1.2). During NRC review of the preliminary draft, the NRC requested deletion of the least limiting of the dual limits on fuel elements as given in the existing specification. The 1800 EFPD limit was retained, and the fuel average burnup limit of 110,000 MWD per tonne of heavy metal (uranium plus thorium) was deleted. Per FSAR Section 3.2.1, the limit of 110,000 MWD per tonne of heavy metal is that to which the fuel is designed for the equilibrium cycle. However, at full power (842 MW-thermal) and given the reference uranium and thorium loadings of the equilibrium cycle reload segments per FSAR Section 3.5.2.1, the average burnup for a fully irradiated (1800 EFPD) reload segment corresponds to only 96,412 MWD per tonne of heavy metal, assuming uniform exposure on a per-segment basis. The upgraded specification is judged to be conservative and acceptable.

3/4.2.2 Core Inlet Orifice Valves/Region Outlet Temperature Limits. (Specification carryover from existing TS) The upgraded specification is a revision of existing Specification LCO 4.1.7. The title has been changed to reflect that the limits apply to the region outlet temperature as well as the inlet flow orifice position (FSAR Sections 3.6.2, 3.6.7, and 3.9). A surveillance requirement has been added, and the limits on comparison regions (FSAR Section 3.6.6.3) have been placed in a separate upgraded specification. The comparison regions are used to establish flow requirements (that is, inlet orifice valve positions) for the regions lacking trustworthy coolant outlet temperature indication.

Existing Specification Figure 4.1.7-1 has been revised as upgraded Specification Figure 3.2.2-1 to more clearly illustrate the prohibited and allowable operating limits for region outlet temperature mismatch under different operating conditions of core average inlet temperature and core average temperature rise. These limits in effect restrict region outlet temperature mismatch to within the values assumed in Specification SL 2.1.1. When the core average outlet temperature is less than 950°F, the further provisions of upgraded Specification LCO 3.2.4 are invoked to cover low flow conditions that can occur in the LOW POWER operating mode. The specification is unique to prismatic HTGRs and is not supplemented by incore power maps because instrumentation did not exist for reliably performing such maps in a high temperature environment at the time FSV was licensed. Other than for neutron distribution measurements to infer incore power distributions, there are no functionally equivalent specifications for PWRs or BWRs. The upgraded specification is judged to be acceptable.

<u>3/4.2.3 Core Inlet Orifice Valves/Comparison Regions</u>. (Specification carryover from existing TS) This specification is a reformatting, revision, and combination of the portion of existing Specification LCO 4.1.7 that relates to comparison regions and of existing Specification SR 5.1.7 with regard to comparison region surveillances (FSAR Section 3.6.6.3). Comparison regions are those similarly fueled and rodded regions that are in symmetric locations to the grouping of regions that have known inadequacies in the measurement of outlet gas flow temperature. These measurements inadequacies are due to intermittent bypass gas flows that can overcool the thermocouples used to measure the outlet temperature.

The objective of the upgraded specification is to assure that the measured relative region power density (called the region peaking factor or RPF) is not less than 10% smaller than the calculated region relative power density for those core regions that are used as comparison regions. The RPF is the ratio of region average power density to core average power density (hence, the term relative power density is used commonly throughout most of the rest of the industry). At FSV, the measured RPF is inferred from the product of the inferred region coolant flow temperature increase (that is, the difference between normalized thermocouple readings of core inlet and region outlet coolant temperatures) and the inferred region coolant mass flow (that is, as inferred both from the orifice setting that dominates changes in regionwise flow resistance and from total core flow that can be inferred from either circulator rotational speeds or circulator outlet pressure-differential flow meters). The specification assures that the controlled region that relies on a comparison region to infer region gas flow temperature rise will have an acceptable basis for establishing flow requirements for that region under upgraded Specification LCO 3.2.[illegible]. The acceptable basis is that the comparison of calculated and measured RPF in the comparison region either agrees within 10%, yields a measured value that is higher than the calculated value, or provides for quantifying the percent measured-to-calculated discrepancy so that a correction can be made to the measured value when used for comparison.

As indicated in the basis summary statement for upgraded Specification LCO 3.1.4.1, Control Rod Pair Position and Worth Requirements, and in Specification DF 5.3.4, Reload Segment Design, one of the objectives in the selection of the control rod pair withdrawal sequence is to "satisfy the criteria for reactor power distribution" wherein the criteria for acceptable RPFs (that is, relative power densities) are provided in Specification DF 5.3.4. Per the NFSC's responsibilities for review, approval and audit as given, respectively, in Specifications AC 6.5.2.8.a, AC 6.5.2.8.h, AC 6.5.2.9.a, AC 6.5.2.10.a, AC 6.5.2.10.c, and AC 6.5.2.10.d, the determination and application of the calculated region RPFs and the trending in evaluated discrepancies between measured and calculated RPFs are judged to be appropriate items to be followed by the NFSC to assure the quality level of the methodology employed in selecting the control rod pair withdrawal sequence and thus to assure also the safety of core operation.

Given NFSC oversight for assuring the accuracy and conservatism of the methods employed, this specification is judged to be acceptable.

<u>3/4.2.4 Core Inlet Orifice Valves/Minimum Helium Flow and Maximum Core Temperature Rise</u>. (Specification carryover from existing TS) This specification is a reformatting of the recently approved (License Amendment N

57) revisions to existing Specifications LCO 4.1.9 and SR 5.1.8. This specification applies (1) when the Core Average Outlet Temperature is less than 950°F per upgraded Specification LCO 3.2.2.b, (2) when the core thermal power level falls below 25% of rated by heat balance, and (3) when, in SHUTDOWN and REFUELING, the Calculated Bulk Core Temperature exceeds 760°F. The specification assures adequate helium cooling flow through the downflow core at low heat loads and pressure drops where the buoyancy effects of heated helium could cause channel flow stagnation and local overheating of the fuel if the pressure drop due to the inlet orifice setting were allowed to be too large. The specification of the allowable region inlet orifice settings has been conservatively developed to avoid core channel flow instabilities for the pressurized and partially depressurized reactor for a range of core heat loads that accommodates fission heating below 25% of rated thermal power as well as decay and residual (stored) heat loads down to a core average temperature of 760°F. Below a core average temperature of 760°F, the combination of decay and residual (stored) heat loads could not cause local overheating of the fuel for many hours in the absence of all flow. Otherwise, if reactor core cooling flow is lost for conditions under which this specification applies, the reactor is to be depressurized per upgraded Specification LCO 3.4.1 to facilitate the PCRV containment function in retaining radioactive fission products that may be released from the core if the fuel were to be damaged from over-heating.

The upgraded specification is judged to be acceptable.

<u>3/4.2.5 Region Constraint Devices</u> (Specification carryover from existing TS) This specification is a revision of existing Specification SR 5.2.26 to include an LCO. As described in FSAR Section 3.3.1.1, the Region Constraint Devices (RCDs), which are located on top of plenum elements of generally three adjacent fuel regions, restrain region movements in relation to one another by means of centering pins inserted in the handling hole of the upper plenum elements. The RCDs are used to limit horizontal movement of the fuel columns which mitigates temperature fluctuations in the primary coolant circuit at the individual core region outlets as discussed in FSAR Section 3.6.6.

Visually inspecting the RCDs will ensure that they are performing their design function by restraining the fuel columns. Monitoring the lifting force required to remove the RCDs with the fuel handling machine will provide early indications of degradation in function. Monitoring the vertical location coordinate ensures that the RCDs have been properly installed with the pins engaged in the plenum elements.

The specification is judged to be acceptable.

<u>3/4.2.6 Power-to-Flow Ratio</u>. (Specification carryover from existing TS) This specification is a revision and combination of that portion of existing Specification SL 3.1 that addressed immediate operator responses to observed plant transients involving a power-to-flow ratio imbalance and of existing Specification SR 5.1.6 with regard to the mechanism for assessing the degree of challenge to the safety limit. The challenge results from a power-to-flow imbalance that exceeds a specified threshold (upgraded Specification Figure 3.2.6-1). The carryover of existing Specification SL 3.1 into upgraded Specification SL 2.1.1 has been discussed previously. FSAR Section 3.6.7 describes the instrumentation systems that alert the operator to the potential existence of a core power-to-flow ratio imbalance, and FSAR Sections 3.6.7.6 and 3.6.8 describe the technical bases for the allowed response times to such imbalances and for determining the degree of challenge posed to the safety limit. Very rapid transients involving a high power-to-flow ratio imbalance have a lag time since the high (conservative) temperatures that are assumed in the calculations of fuel particle migration (see discussion in previous evaluation of Specification SL 2.1.1) take time to develop because of the high heat capacity of the graphite core. Thus, in rapid transients, the core temperature lags behind the steady state assumptions used to quantify the safety limit. Such transients, which should be readily detected and responded to by the Plant Protection System (PPS), require a prompt response from the operator but, if quickly terminated, may not contribute to the challenge to the safety limit because of the thermal sluggishness of the core.

The specification is judged to be acceptable.

<u>3/4.3 Instrumentation</u>

3/4.3.2 Monitoring Instrumentation

3/4.3.2.3 Seismic Instrumentation. (Specification carryover from existing TS) This specification is a revision and expansion of existing Specifications LCO 4.4.4 and SR 5.4.10 to more closely follow the provisions of W-STS LCO/SR 3/4.3.3.3. All triaxial time-history accelerographs and vertical seismic triggers as previously installed at FSV are required to be operable (FSAR Section 7.3.9), but, consistent with the TSUP scope guidelines for limiting the backfitting of W-STS requirements (see Appendix A to this TER), no new or additional seismic instrumentation has been installed to equate numerically to that required in the W-STS. However, the FSV seismoscopes, which are noncalibratable smoked glass motion indicators (FSAR Section 7.3.9), have been included in the upgraded specifications to assure diversity and redundancy in seismic event detection.

Channel calibration for all out-of-calibration seismic instruments found following a seismic event is to be performed within 30 days following the seismic event as opposed to within 10 days for all seismic instruments per the W-STS. Because FSV cannot perform the channel calibration on site, a channel functional test is performed within 5 days followed by offsite channel calibration for those found out-of-calibration. This allows time to keep the instrumentation in use on site to record potential aftershocks. The Special Report of seismic effects is required within 14 days as opposed to 10 days to allow adequate time for recording and evaluation of aftershocks.

The specification is judged to be acceptable.

3/4.3.2.4 Meteorological Instrumentation. (Specification adapted from STS) This specification is equivalently worded to, if not exactly worded as, W-STS LCO/SR 3/4.3.3.4, Meterological Instrumentation. The TS reflects the current guidance given in proposed Revision 1 to Regulatory Guide 1.23 (Ref. 33). This specification is judged to be acceptable.

3/4.3.2.7 Power-to-Flow Ratio Instrumentation System. (Specification carryover from existing TS) This specification is an upgrade and expansion of existing Specification SR 5.4.8 to include an LCO that was merely implied and not quantified in terms of acceptable ACTION statements by the existing SR. The system is briefly described in FSAR Sections 3.6.7.1 and 7.3.11. The system operates continuously and has no automatic actuation function. Backup instrumentation can be used if the system is inoperable.

The upgraded TS most closely resembles, in terms of functional equivalence, W-STS LCO/SR 3/4.3.3.6, Accident Monitoring Instrumentation; however, the condition monitored and recorded at FSV is the profile of a power-to-flow imbalance or transient to allow post-transient assessment of the degree of challenge to Specification SL 2.1.1, which is discussed previourly in this TER. In another sense, the instrumentation system is somewhat functionally analogous to the recording and analysis devices that would be addressed in the non-TS implementing procedures for W-STS LCO/SR 3/4.3.3.2, Movable Incore Detectors, that are used for assessing compliance with the associated TS in W-STS Section 3/4.2, Power Distribution Limits. However, the condition monitored and recorded at FSV is the result of a transient compared to a safety limit and not solely the continued assurance of initial conditions for design basis accidents or transients as in the PWR. Thus, the function of the instrumentation system is unique to FSV but can be described as being addressed in the TS consistent with that of virtually analogous functions in the W-STS.

The specification is judged to be acceptable.

<u>3/4.3.2.8 Core Region Outlet Thermocouples</u>. (Specification adapted from STS) This specification is for instrumentation that is used to support assessing compliance with core region power-to-flow limits per Specifications LCO/SR 3/4.2.2, LCO/SR 3/4.2.3, and LCO/SR 3/4.2.4 as well as backup instrumentation for core power-to-flow transient assessment per LCO/SR 3/4.2.6. The specification is functionally analogous to W-STS LCO/SR 3/4.3.3.2, Movable Incore Detectors, but is written to reflect the unique need in the large prismatic HTGR to infer core and region power-to-flow ratio as opposed to fission power distribution in the PWR. The specification accounts for the redundancy in the thermocouples used in FSV (FSAR Section 3.6.7). The specification is judged to be acceptable.

3/4.4 Primary Coolant System

3/4.4.1 Primary Coolant Loops and Coolant Circulation

3/4.4.1.1 Primary Coolant Loops and Coolant Circulation -- above 5% Power. (Specification adapted from STS) This specification is new. The specification reflects the allowable forced circulation cooling configurations for FSV power operation per FSAR Section 4.3 and provides functional consistency with the intent and provisions of W-STS LCO/SR 3/4.4.1.1, Startup and Power Operation. The new specification provides for distinguishing between the helium circulator functional requirements for being in operation on steam or water turbine drive for normal process cooling and being operable on water turbine drive for safe shutdown cooling. The specification also incorporates the appropriate ACTIONs to effect shutdown and reactor depressurization if all forced circulation cooling is lost for an extended period of time; the requirement and timing for depressurization were carried over from existing Specification LCO 4.2.18, Primary Coolant Depressurization. The upgraded specification complements the TS limits for assuring adequate core and regionwise power-to-flow ratio as stipulated in Section 3/4.2 of the upgraded TS. This is done by specifying the required operating configuration of the primary coolant system equipment as a function of core power level.

A detailed functional evaluation of this specification is provided in Section 4.2 of the ORNL TER (Ref. 3) on the upgraded TS for FSV cooling functions. This specification is judged to be acceptable.

3/4.4.1.2 Primary Coolant Loops and Coolant Circulation - Below 5% Power. (Specification adapted from STS) This specification is new and supplements LCO/SR 3/4.4.1.1 that is discussed above. The specification is functionally consistent with the intent and provisions of W-STS LCO/SR 3/4.4.1.1, LCO/SR 3/4.4.1.2, and LCO/SR 3/4.4.1.3 with regard to providing an adequate primary coolant system equipment configuration during the FSV conditions of reactor startup and equivalent hot shutdown. During shutdown at FSV, the functional equivalent to hot shutdown has been established as effectively existing when the core average temperature (that is, the Calculated Bulk Core Temperature) exceeds 760°F. Below 760°F, forced circulation cooling is not required because there are significant thermal margins to temperatures at which the primary coolant system structures and the fuel fission product barriers would begin to be damaged. Otherwise above 760°F, one operating primary coolant loop consisting of at least one operating helium circulator and one operating steam generator section is required for the conditions for which Specification LCO 3.4.1.2 applies. If forced circulation cooling is lost for an extended period, the specification provides for depressurization of the reactor to mitigate the challenge to PCRV containment by core heatup.

This specification is evaluated in detail in Sections 4.2 and 4.3.4 of the ORNL cooling function TER (Ref. 3) with regard to the functions of normal process cooling and residual heat removal. The specification is judged to be acceptable.

<u>3/4.4.2 Primary Coolant Activity</u>. (Specification carryover from existing TS) This specification is a combination and upgrade of existing FSV Specifications LCO 4.2.8, Primary Coolant Activity Limits, SR 5.2.6, Plateout Probe, and SR 5.2.11, Primary Reactor Coolant Radioactivity. The upgraded TS is functionally analogous to W-STS LCO/SR 3/4.4.8, Specific Activity, but is structured to retain the limits of the existing specifications that reflect the unique features and operating conditions of a HTGR as compared to the LWR. The upgraded TS includes limits on both circulating and plateout activity (FSAR Sections 3.7, 9.4, and 14.12). In HTGRs, plateout activity is important in the assessment of blowdown source terms due to plateout liftoff during a postulated rapid depressurization accident (FSAR Sections 3.7 and 14.11). The carryover surveillance on the plateout probe is the basis for assessing the continued conservatism in FSAR assumptions for accident analysis. The definitions for Dose Equivalent I-131 and E-BAR (the average beta-gamma decay or disintegration energy) have been added to Section 1 of the upgraded TS consistent with the format of the W-STS but reflecting both the accident analysis assumptions and the limit assessment techniques that are employed at FSV. FSV currently operates well below limits on the coolant activity and plateout activity.

This specification complements the provisions of upgraded

Specification LCO/SR 3/4.7.5, Primary Coolant Depressurization, that provides for the operability of the Helium Purification System (HPS). The HPS is normally operating to assure meeting primary coolant limits on both radioactivity and impurity levels but is also relied upon as a Class I safety-related system to assure containment integrity by providing for a filtered depressurization during the permanent loss of forced cooling accident (FSAR Section 14.10). LCO 3.4.2 assures that circulating activity is within the initial condition assumed in the accident analyses.

The upgraded specification is judged to be acceptable.

3/4.4.3 Primary Coolant Impurity Levels - High Temperature. (Specification carryover from existing TS) The specification is an upgrading and combination of the existing Specification LCO 4.2.10 and associated SR 5.2.12 as well as existing surveillances for potential impurity-induced structural effects as provided in existing Specifications SR 5.2.22, PGX Graphite, and SR 5.2.25, Core Support. A fuel surveillance program that addresses the detection of impurity-induced effects on the fuel particles is effected outside the TS. The specification LCO is applicable when Core Average Outlet Temperature is greater than 1200°F. The TS provides limits on primary coolant oxidant (O_2, CO_2, and CO) levels that are imposed to mitigate the effects of graphite oxidation and carbon mass transport from the core to cooler surfaces such as the steam generator tubes and PCRV liner surface (FSAR Sections 4.2.1 and 9.4.2). Surveillances of the PGX graphite specimens and of the core support blocks, which are manufactured from PGX graphite, provide assurance that safety-related graphite structures are maintaining adequate mass and strength and are not subject to unanticipated preferential degradation under allowed oxidant impurity levels (FSAR Section 3.3.2.2 and Appendix A.12). The basis summary statement identifies the instrumentation and alarms, including the PPS moisture monitors, that are available to alert the operator of conditions that indicate a potential for increased oxidant levels in the coolant. The specification is judged to be acceptable.

3/4.4.4 - Primary Coolant Impurity Levels - Low Temperature. (Specification carryover from existing TS. This specification is an upgrade of existing Specifications LCO 4.2.11 and SR 5.2.12 as applicable when the reactor is generating fission energy and the Core Average Outlet Temperature is less than 1200°F. The oxidant (H_2O, CO_2, and CO) limits under these conditions are based on preventing the corrosion of metallic components and the oxidation of the boron carbide in burnable poisons (FSAR Sections 4.2.1 and 9.4.2). The specification is judged to be acceptable.

3/4.5 Safe Shutdown Cooling Systems

3/4.5.1 Helium Circulators

3/4.5.1.1 Helium Circulators -- CBCT above 760. (Specification carryover from existing TS) This specification is an upgrade of existing Specifications LCO 4.2.1 and SR 5.2.7. As in the existing specification, one helium circulator in each of the two primary coolant loops is required to be operable on its associated water turbine drive for purposes of performing safe shutdown cooling (FSAR Sections 10.3.9 and 14.4.2.2). Safe shutdown cooling is the functional equivalent at FSV of emergency core cooling at a PWR. The equipment supporting the capability for water turbine drive for safe shutdown cooling is Class I and is seismically and environmentally qualified. A detailed functional evaluation of this specification is provided in Sections 4.1 and 4.4 of Ref. 3.

This specification is applicable whenever the reactor is generating fission heat at a rate exceeding 5% of rated reactor power and otherwise when the Calculated Bulk Core Temperature exceeds 760°F. The W-STS restoration time of 72 hours has been adopted for the inoperability of a single train; one hour for both trains. The time period to achieve shutdown given failure to restore the inoperable equipment is 24 hours based on the period allowed in the existing specifications. This allowance is consistent with the larger thermal margins and longer thermal response times of the FSV ceramic graphite core, and the low doses calculated for the permanent loss of forced cooling accident (FSAR Section 14.10 and Appendix D). Operability of the helium circulators on steam drive is assured for normal operation by upgraded Specification LCO 3.4.1, which has been discussed previously in this TER.

This specification is judged to be acceptable.

3/4.5.1.2 Helium Circulators -- CBCT below 760. (Specification adapted from STS) This specification is an adapted upgrade of existing Specifications LCO 4.2.1 and SR 5.2.7 to be applicable when only one train of the Safe Shutdown Cooling System is required to be operable, similar to the W-STS provisions for the emergency core cooling system. Previously, existing Specification LCO 4.2.1 required one helium circulator in each loop to be operable on its associated water turbine drive in power operation (greater than 2% of rated thermal power) The upgraded specification requires one helium circulator in only one primary coolant loop to be operable whenever the Calculated Bulk Core Temperature is less than or equal to 760°F, including operation at up to 5% of the rated reactor power. In combination with upgraded Specification LCO 3.5.1.1 as discussed above, this specification is more comprehensive than the existing specification. The specification is judged to be acceptable.

3/4.5.2 Helium Circulator Auxiliaries

3/4.5.2.1 Helium Circulator Auxiliaries -- CBCT above 760. (Specification carryover from existing TS) This specification is an upgrade, combination, and simplification of existing Specifications LCO 4.2.2, LCO 4.2.3, SR 5.2.8, SR 5.2.9, SR 5.2.23, and SR 5.2.27. In comparison to the existing specifications, the upgraded specifications distinguishes between the operability of the helium circulator on water turbine drive per upgraded Specification LCO 3.5.1 and the operability of the multiplicity of normally operating auxiliaries that must operate in order for the circulator to operate on either water turbine drive or steam turbine drive (FSAR Section 4.2.2.3). The upgraded specification assures the operability of both trains (Loops 1 and 2) of the helium circulator auxiliary equipment that is required for safe shutdown cooling (FSAR Sections 10.3.9 and 14.4.2.2). The specification also assures the operability of the bearing water accumulator system which is needed to prevent circulator damage during certain circulator trips and thus to facilitate restart for safe shutdown cooling if required. The specification also assures the capability of automatic water turbine startup above 30% of rated reactor power (FSAR Section 4.2.2.3.5). The water turbine auto-start is a mitigative feature for responding to the trip of both helium circulators on steam drive during one loop operation or in the untripped loop during two loop trouble (FSAR Sections 4.3.2, 4.3.3, 7.1.2.4, and 14.4.3). The specification is applicable for the same plant conditions as Specification LCO 3.5.1.1 for operability of the helium circulator water turbine drives for safe shutdown cooling. Restoration time for any single component is 72 hours except for the bearing water accumulators which have a more restrictive 24-hour restoration time since this component is relied upon to prevent circulator damage upon trip with loss of normal bearing water. A functional analysis of this specification is provided in Section 4.4 of Ref. 3.

This specification is judged to be acceptable.

3/4.5.2.2 Helium Circulator Auxiliaries -- CBCT below 760. (Specification adapted from STS) This specification requires an operable train of auxiliaries for the helium circulator required to be operable per Specification LCO 3.5.1.2. This equipment is the same as that for each single train addressed in Specification LCO 3.5.2.1 except that the water turbine automatic start feature is not relied upon below 30% of rated reactor power. Loss of operability of any single component requires being in at least SHUTDOWN within 12 hours.

This specification is judged to be acceptable.

3/4.5.3 Steam Generators

3/4.5.3.1 Steam Generators -- CBCT above 760. (Specification carryover from existing TS) This specification is an upgrade and combination of existing Specifications LCO 4.3.1 and SR 5.3.10 with SR 5.3.11 and SR 5.3.12. The latter two existing surveillances relate to required inspections, respectively, of bimetallic welds and of steam generator tube leaks; these surveillances have been carried over into the upgrade.

The existing Specification LCO 4.3.1 required both the reheater section and the economizer-evaporator-superheater (EES) section of each steam generator to be operable during power operation for the purpose of removing decay heat (FSAR Sections 4.3.4 and 14.4.); the current specification interprets operable as being the capability of each steam generator section to receive and dispell the flow of Safe Shutdown Cooling Water (that is, firewater) from the seismically and environmentally qualified sources (FSAR Sections 4.2.4.3.3, 10.3.9, and 14.4.2.2). The specification requires the operability of each steam generator section when the plant is operating above 5% of rated reactor power and otherwise when the Calculated Bulk Core Temperature exceeds 760°F. With the provision of firewater after a 90-minute delay in restoring forced circulation cooling of the reactor, either EES section is capable of removing decay heat without fuel damage (that is, safe shutdown cooling) following equilibrium operation at or below 82% of rated reactor power. Under the same conditions, the reheater sections can be used for safe shutdown cooling following equilibrium operation at or below 39% of rated reactor power. The reheaters are also relied upon for responding to other transients as described in FSAR Sections 14.4 and 14.5.

The restoration time for one inoperable section is 72 hours followed by 24 hours to be in at least SHUTDOWN; the restoration time for any two inoperable sections is one hour plus 24 hours to be in at least SHUTDOWN. The specification is judged to be acceptable.

3/4.5.3.1 Steam Generators -- CBCT below 760. (Specification adapted from STS) This specification provides for the operability of one Safe Shutdown Cooling System train by assuring the operability of one steam generator section in the loop with an operable helium circulator when the Calculated Bulk Core Temperature is less than 760°F. The specification is judged to be acceptable.

3/4.5.4 Emergency Condensate and Emergency Feedwater Headers

3/4.5.4.1 Emergency Condensate and Emergency Feedwater Headers -- CBCT above 760. (Specification carryover from existing TS) This specification is an upgrade of existing Specifications LCO 4.3.4 and SR 5.2.7. The specification requires both headers to be operable for operation above 5% of rated reactor power and otherwise when the Calculated Bulk Core Temperature exceeds 760°F. This assures redundancy in the seismically and environmentally qualified flow paths from the Safe Shutdown Cooling Water Supply System to the helium circulator water turbine drives and the steam generator sections for effecting safe shutdown cooling (FSAR Sections 10.3.9 and 10.3.10). Above 30% of rated reactor power, the emergency feedwater header is operating so that, in combination with the auto-start feature of the water turbines, high pressure feedwater drive (and therefore high speed drive) of the helium circulators can be provided if steam drive is lost to the circulators.

This specification is judged to be acceptable.

<u>3/4.5.4.2 Emergency Condensate and Emergency Feedwater Headers -- CBCT below 760</u>. (Specification adapted from STS) This specification requires either the emergency condensate header or the emergency feedwater header to be operable to provide a single equivalent train operable for safe shutdown cooling when the Calculated Bulk Core Temperature is less than 760°F. This specification is judged to be acceptable.

<u>3/4.5.5 Safe Shutdown Cooling Water Supply System</u>. (Specification carryover from existing TS) This specification upgrades and combines the Safe Shutdown Cooling Water Supply functions covered in existing Specifications LCO 4.2.6, SR 5.2.10, and SR 5.2.24, but deletes the fire suppression requirements of the first two existing specifications. Fire suppression requirements are included in upgraded Specification LCO/SR 3/4.7.6.1, Spray and/or Sprinkler Systems. The portions of existing Specification SR 5.2.24 that relate to equipment in other systems is also broken out into other upgraded specifications as appropriate. The Safe Shutdown Cooling Water Supply System is constituted of the circulating water makeup ponds and the pumps, valves, and flowpaths from the ponds to the firewater supply header that can feed the emergency condensate and emergency feedwater headers through diverse and redundant flowpaths. The upgraded specification requires the operability of two equivalent trains of Safe Shutdown Cooling Water Supply; the terminology "equivalent trains" is used since there are redundant cross connections that provide flexibility in constituting a single train. The cross connections are single-failure proof (FSAR Section 10.3.10).

For plant conditions above a Calculated Bulk Core Temperature of 760°F, if one component of the system is inoperable, the restoration time is 72 hours. If more than one component is inoperable but if one equivalent operable train exists, the restoration time for all components is 48 hours. The existance of no operable flowpath requires restoration of at least one operable flowpath within one hour. After any restoration time is expired without the required restoration, the reactor is to be in at least SHUTDOWN within 24 hours. For plant conditions below a Calculated Bulk Core Temperature of 760°F, two trains are also required but 14 days is allowed for restoration of an inoperable train or the provision of a backup.

The specification is judged to be acceptable.

3/4.6 PCRV and Confinement Systems

3/4.6.1 PCRV Pressurization

3/4.6.1.1 PCRV Safety Valves. (Specification carryover from existing TS) This specification is an upgrade of existing Specifications LCO 4.2.7 and SR 5.2.1 except that existing provisions for the PCRV penetration overpressure protection systems and the penetration minimum pressurization requirements have been broken out into separate upgrade specifications. The PCRV safety valves are an engineered safety feature described in FSAR Section 6.8. The specification is applicable whenever PCRV pressure exceeds 100 psia and requires two operable pressure relief paths, either of which can relieve PCRV overpressure prior to the PCRV reaching the safety limit reference pressure of 845 psig.

Since the primary coolant is a single phase gas and not a two phase fluid, credible overpower and overheating transients are not expected to lift the PCRV safety valves. For a water ingress event to challenge the lifting of the safety valves, several levels of protection that are also subject to TS would also have to fail as described in FSAR Sections 4.3.6 and 6.8. Thus, lifting of the safety valves is not expected.

This specification is judged to be acceptable.

<u>3/4.6.1.2 Steam Generator/Circulator Penetration Overpressure Protection</u>. (Specification carryover from existing TS) This specification is an upgrade of those portions of existing Specifications LCO 4.2.7 and SR 5.2.1 that relate to PCRV penetration overpressure protection in those PCRV penetrations that have piping with high pressure and/or high energy fluids passing through them: These penetrations include those for the steam generators (FSAR Section 5.8.2.5.4) and the helium circulators (FSAR Section 5.8.2.5.5). The pressures of helium circulator bearing water, feedwater, superheat steam, and reheat steam can exceed PCRV reference pressure during normal operation. Leaks or partial ruptures of high pressure/high energy fluid piping in the penetration interspace are accommodated by safety valves on the penetrations to preclude overpressure challenges to the integrity of the PCRV and of the seismically qualified penetration closures. The specification is judged to be acceptable.

<u>3/4.6.1.3 Interspace Minimum Pressurization</u>. (Specification carryover from existing TS) This specification is an upgrade and combination of that portion of existing Specification LCO 4.2.7 that relates to PCRV penetration interspace minimum pressurization and of existing Specification 5.2.15. Penetration interspace pressurization is accomplished with clean purified helium. The specification requires penetration interspace pressure to be maintained higher than reactor coolant pressure when PCRV pressure exceeds 100 psia. An exception allowed by the specification is for the interspace pressure in steam generator penetrations with known leak paths from the penetration interspace to reheat steam For this exception, penetration interspace pressure is maintained just above cold reheat steam pressure to minimize the leakage of clean helium into the reheat steam (FSAR Section 5.8.2.5.4). In all other penetrations, interspace pressure is held higher than reactor pressure to assure purified helium leakage into the reactor, rather than contaminated helium leakage out if the seismically qualified primary closure were to leak. Potential contaminated helium leakage into or through the steam generator penetration interspaces would be detected by the PPS radiation detectors on the reheat steam lines (LCO 3.3.1), the non-PPS monitors on the interspaces (LCO 3.6.1.5), and the non-PPS monitor on the condensor air-ejector (Environmental LCO 8.1.1.g.7).

This specification is judged to be acceptable.

3/4.6.1.4 PCRV Closure Leakage. (Specification carryover from existing TS) This specification is an upgrade of existing Specification LCO 4.2.9 and related portions of Specifications SR 5.2.16.a and SR 5.2.16.b except that surveillances of instrumentation without automatic actuation functions have been deleted from the upgrade consistent with the W-STS definition of OPERABLE as adapted for defining FSV equipment operability (Attachment 1 to Ref. 30). This specification limits PCRV penetration interspace leakage of purified helium to 400 lb/day except for the steam generator reheater header leakage that is limited to 700 lb/day of purified helium. The 400 lb/day leak rate equates to a 1145 lb/hr leak through the seismically qualified primary closure with the PCRV at full pressure (688 psig) and the penetration interspace depressurized (that is, an assumed total failure of the seismically qualified secondary closure with the penetration at the same pressure as the reactor building confinement). The resulting release in this low probability situation is reported to be on the same order as the maximum credible accident (FSAR Section 14.8), which assumes a rupture in the worst location of the helium purification train.

This specification is judged to be acceptable.

3/4.6.2 Reactor Plant Cooling Water/PCRV Liner Cooling System

3/4.6.2.1 Reactor Plant Cooling Water/PCRV Liner Cooling System -- CBCT above 760. (Specification carryover from existing TS) This specification is an upgrade and combination of existing Specifications LCO 4.2.13, LCO 4.2.14, SR 5.4.4, and SR 5.4.11. Consistent with the W-STS definition of OPERABLE as adapted for FSV, the instrumentation surveillances in the latter two existing specifications have been deleted from the upgraded specifications and will be addressed in the licensee's implementing procedures for assuring system operability (Attachment 1 to Ref. 30). This is an acceptable simplification of the level of detail addressed in the TSUP consistent with the W-STS treatment of instrumentation with no automatic actuation functions.

The upgraded specification requires two operable trains of the Reactor Plant Cooling Water (RPCW) and the PCRV Liner Cooling System (LCS) including redundancy in the PCRV liner cooling tubes. Every other PCRV liner cooling tube is supplied by a different (redundant) train of the RPCW/PCRV LCS. Only one train supplying half the total number of tubes on the PCRV liner is required to assure primary coolant boundary and containment integrity during the design basis accident of permanent loss of forced circulation cooling (FSAR Section 14.10.3.1 and Appendices D.1.2.1.5, D.2.2, and D.2.3). A detailed functional analysis of the containment heat removal functions performed by the RPCW/PCRV LCS and assured by the upgraded TS is provided in Section 4.5 of Ref. 3. In addition, operability of the RPCW/PCR LCS is also conditional upon acceptable cooling water temperature rise in locations on the PCRV that have known hot spots.

During plant operation above 5% of rated reactor power and during other plant conditions with the Calculated Bulk Core Temperature exceeding 760°F, both trains of the RPCW/PCRV LCS are to be operating with specified tube redundancy being maintained. Loss of one train requires restoration within 48 hours and being in at least SHUTDOWN within the following 12 hours, and loss of tube redundancy for reasons other than the inoperability of a single train requires restoration of tube redundancy within 24 hours plus another 24 hours to be in SHUTDOWN if restoration is not accomplished. Loss of operability in both trains requires immediate shutdown. Increase in known PCRV liner hot spot temperatures (FSAR Section 5.9.2.8) must be compensated for within 7 days or a corrective action report submitted to NRC within 14 days. The known hot spots have significant margin to temperatures at which PCRV concrete strength would be challenged (Attachment 2 to Ref. 1). The specification is judged to be acceptable.

<u>3/4 6.2.2 Reactor Plant Cooling Water/PCRV Liner Cooling System -- CBCT below 760</u>. (Specification adapted from STS) The provisions of existing Specifications LCO 4.2.13 and LCO 4.2.14 required two operable and operating trains above 2% of rated reactor power. The upgraded specifications are more comprehensive in providing for at least one operable and operating train whenever the Calculated Bulk Core Temperature is less than 760°F. The

specification has liberal restoration times (that is, prior to reaching a Calculated Bulk Core Temperature of 760°F) because of the very large thermal margins inherent in the reactor at temperatures below a core average temperature of 760°F; however, elsewhere[31,32] the NRC has indicated that residual heat removal from the reactor should be actively accomplished and available either by forced circulation cooling or through the PCRV LCS under cold core conditions as demarcated by the 760°F limit. Upgraded Specification LCO 3.6.2.2 is the mechanism for assuring the continuation of an effective residual heat removal function as well as the assurance of the containment heat removal function during equivalent cold shutdown conditions at FSV. A detailed functional analysis of this specification is provided in Sections 4.3.4 and 4.5 of Ref. 3. This specification is judged to be acceptable.

<u>3/4.6.3 Reactor Plant Cooling Water/PCRV Liner Cooling System Temperature</u>. (Specification carryover from existing TS) This specification is an upgrade and combination of existing Specifications LCO 4.2.15, SR 5.4.4, and SR 5.4.5. Consistent with the W-STS definition of OPERABLE as adapted at FSV, the instrumentation surveillances of the latter two existing specifications have been deleted from the upgraded TS and placed under the licensee's implementing procedures for assuring monitoring capability for the monitored parameters (Attachment 1 to Ref. 30). The affected instrumentation has no automatic actuation function. The specification assures that LCS water and PCRV concrete temperatures are maintained within acceptable operating limits (initial conditions) as specified in FSAR Sections 5.7, 5.9, 5.12, and 9.7. This specification is judged to be acceptable.

<u>3/4.6.4 PCRV Integrity</u>

<u>3/4.6.4.1 Structural Integrity</u>. (Specification carryover from existing TS) This specification is an upgrade that involves adding an LCO and combining existing Specifications SR 5.2.2, SR 5.2.3, SR 5.2.4, and SR 5.2.13 to address jointly PCRV tendon and concrete integrity. PCRV tendon integrity is based on meeting tendon load requirements (lift-off tests) and corrosion limits on the tendon anchor assemblies including surrounding-concrete

degradation. The projected effects of tendon wire corrosion as evidenced by failed wires are compared against the actual test results for meeting tendon load requirements using the remaining available steel cross-sectional area when failed tendon wires are accounted for as the basis for comparison. Evaluations and reporting of results to the NRC are required annually. Structural integrity of PCRV concrete (other than visual examination in the vicinity of tendon anchor assemblies) is demonstrated through surveillance of deformations and deflections, helium permeability tests on PCRV-installed test configurations, crack mapping, and other periodic visual examinations for evidence of deterioration. FSAR Sections 5.2, 5.3, 5.4, 5.5, 5.6, 5.12, and 5.13 provide detailed technical bases for the PCRV structural integrity determination. This specification is judged to be acceptable for assuring PCRV, and therefore containment, structural integrity.

<u>3/4.6.4.2 Liner</u>. (Specification carryover from existing TS) This specification is an upgrade that consists of adding an LCO and combining existing Specifications SR 5.2.5 and SR 5.2.14 for surveillance of irradiation and corrosion effects. FSAR Section 5.2, 5.7, 5.8, and 5.13 provide the technical bases for the specification limits. This specification is judged to be acceptable. Evidence of the continued integrity of the liner insulation and thermal barriers is assured by upgraded Specifications LCO 3.6.2 and LCO 3.6.3 that have been discussed above.

<u>3/4.6.4.3 Penetrations, Wells, and Isolation Valves</u>. (Specification carryover from existing TS) This specification is an upgrade that consists of adding an LCO and combining existing Specifications SR 5.2.28 with SR 5.2.16.a through SR 5.2.16.g and SR 5.2.24.h. Consistent with the W-STS definition of OPERABLE as adapted for defining operability of FSV equipment, surveillances on instrumentation, except for instrumentation used to actuate an automatic isolation function, have been deleted from the upgraded TS and are to be addressed in the implementing procedures (Attachment 1 to Ref. 30). This specification addresses (1) containment integrity as provided by the structural and mechanical integrity of the PCRV penetrations and wells, and (2) confinement integrity as provided by automatic isolation valves in the

respective purified helium and cooling water supply lines to and discharge lines from the penetrations, the PCRV LCS and the purification cooling water system. Potential leaks paths through the PCRV via the purified helium or the LCS cooling water lines could compromise confinement integrity in an accident condition and are thus protected by automatic isolation valves. The bases for the surveillances of the subject components are described in FSAR Sections 5.3, 5.8, 5.9, 5.12, 5.13, 9.4, and 9.7. The specifications are judged to be acceptable.

3/4.7 Plant and Safe Shutdown Cooling Support Systems

3/4.7.1 Turbine Cycle

3/4.7.1.1 Boiler Feed Pumps. (Specification carryover from existing TS) This specification is an upgrade and combination of existing Specifications LCO 4.3.2 and SR 5.2.7. As with the existing TS, the upgraded TS is based on maintaining redundant capability in the boiler feed pumps to supply high pressure feedwater for helium circulator water turbine drive and steam generator cooling in the event of a reactor depressurization accident that could be accompanied by the simultaneous loss of reactor-generated steam. A detailed functional analysis of the upgraded TS is provided in Sections 4.1 and 4.3.3 of Ref. 3, and the redundancy requirements for helium circulators on feedwater drive are supported by the evaluation in Ref. 2.

During power operation above 33% of rated reactor power, both steam-driven boiler feed pumps (33% capacity each) are normally in operation to provide normal feedwater supply to the steam generators. Above 30% of rated reactor power, one of the steam-driven feed pumps is used to maintain water supply to and pressure in the emergency feedwater header. Upon loss of steam drive to the helium circulators during either one loop operation (FSAR Sections 4.3.1 and 4.3.2) or a two loop trouble event (FSAR Section 7.1.2.4), the emergency feedwater header is used to automatically supply the helium circulator water turbine drives upon actuation of the automatic start function that is required to be active per upgraded Specification LCO 3.5.2.1. Above 65% of rated reactor power, the fixed-speed electric-motor-driven feed pump is placed in operation. The turbine-driven boiler feed pumps are used for normal startup and shutdown maneuvers because of their speed control capability whereas the motor-driven pump is maintained in operable standby during such maneuvers because its fixed-speed characteristics make it difficult to control by use of the throttle valve and bypass throttle valve; however, the motor-driven pump can be used to execute its important-to-safety function and this is demonstrated periodically by surveillance. The electric power requirements for the motor-driven pump exceed the capability of the on-site diesel generators. Therefore, the motor-driven feed pump requires off-site power; whereas the two turbine driven feedpumps can be operated using steam from the reactor or from either the auxiliary boiler or the backup auxiliary boiler that are diesel-fired.

For purposes of accommodating long-term cooling without fuel damage following reactor shutdown in the depressurization accident, two helium circulators are required to be operating within 60 minutes of a rapid depressurization accident (RDA) as described in FSAR Sections 4.3.3 and 14.11.2.2, and at least one helium circulator is assumed to operate continuously in the case of the maximum expected depressurization rate (FSAR Sections 4.3.3, 14.4.3.2 [Case C2], and 14.8). In both cases of the rapid and slow depressurization accidents, condensate is assumed to be available with or without offsite power since the small (12 1/2% capacity) condensate pumps can be powered off the emergency diesel generators. Part of the condensate can also be routed to either section of either steam generators via the emergency condensate header during a depressurization accident so that feedwater may be supplied only to the circulator drives if required. As discussed in Ref. 2, a slow depressurization accident may have the equivalent or nearly equivalent effect as the RDA if the primary coolant leak from the PCRV actuates the Steam Line Rupture Detection and Isolation System (SLRDIS) and the operators have to expend time to restore primary and secondary coolant flow that is isolated by SLRDIS. The actuation of SLRDIS precludes long-term adverse environmental conditions within the reactor building due to rupture of high energy fluid piping and thus contributes to the success path for restoring forced cooling in the event that a reactor depressurization PCRV occurs due to a penetration failure in which a high energy fluid line also fails. Ref. 2 also demonstrates that the RDA following equilibrium operation at 82% of rated reactor power can be accommodated with the operation of one train of the seismically and environmentally qualified Safe Shutdown Cooling System with a 60-minute delay in startup and leading to only about 1% of fuel damage in the worst case. As discussed in Ref. 2, the offsite doses would be less than the 10CFR Part 100 guidelines for this case.

The upgraded TS requires that both the motor-driven feed pump and one of the two steam turbine-driven feed pumps be operable to provide diversity and redundancy or, if the motor-driven feed pump is inoperable, that both steam turbine-driven feed pumps be operable to provide redundancy with one of the auxiliary boilers operable to provide assured diversity in steam supply within 60 minutes of the onset of a RDA (FSAR Section 14.11.2.2). To satisfy requirements stemming from the recent environmental qualification of FSV per 10CFR Part 50.49, the auxiliary boilers are no longer allowed to be operating on intermittent firing (FSAR Sections 6.3 and 10.2.6) when reactor power exceeds 35% of rated and the motor-driven feed pump is operating but not classified as operable per this specification. Operability of the feed pumps is to be demonstrated at least once per refueling cycle by operation of two helium circulators at 8000 rpm on feedwater supply through the emergency feedwater header from each pump. The specification is applicable whenever the reactor is operating above 5% of rated reactor power and otherwise when the Calculated Bulk Core Temperature exceeds 760°F. As discussed in Ref. 3, this requirement is judged to be conservative for equilibrium operation below 35% of rated reactor power because the PCRV LCS can accommodate the pressurized core heatup event with little or no fuel damage (FSAR Appendix D.4).

This specification is judged to be acceptable.

<u>3/4.7.1.3 Pressure Relief Valves</u>. (Specification carryover from existing TS) This specification is an upgrade of existing Specification SR 5.3.3 to which has been added an LCO for those valves that are not used normally and are relied upon to accommodate anticipated transients as described in the FSAR. Consistent with the W-STS, normally-used steam bypass valves (FSAR Sections 10.1.1, 10.1.3, 10.2.5.1, and 10.2.5.3) have been deleted from the specification, and the main steam power-operated pressure relief valves

(FSAR Section 10.2.5.3) have also been deleted since these valves are not relied upon in any FSAR accident analysis. The reheat steam power-operated pressure relief valves (FSAR Figure 10.1-1) that are addressed in the upgraded TS are relied upon to accommodate loss-of-offsite power transients as described in FSAR Sections 10.3.1 and 10.3.2. The specification is judged to be acceptable.

3/4.7.1.5 Safety Valves -- Operating. (Specification carryover from existing TS) This specification[30] is a clarifying upgrade of existing Specification SR 5.3.9 to which has been added an LCO. Consistent with the W-STS, surveillance requirements have been modified to reflect ASME code requirements. A safety valve on each steam generator superheater section outlet is required to be operable for each operating boiler feed pump; so all three superheater safety valves must be operable when the three feed pumps are in operation. The single low-pressure safety valve on each steam generator reheater section outlet is required to be operable at any time that fission heat is being generated. A restoration time of 72 hours is allowed for an inoperable valve. Power operation is allowed to continue at or below about 66% of rated for an inoperable superheater safety valve and at or below about 33% for two inoperable superheater safety valves on an operating steam generator. A reheater safety valve, which has no redundancy, must be restored or the reactor shut down. A brief functional analysis of this specification in comparison to W-STS requirements is provided in Section 4.8 of Ref. 3. This specification is judged to be acceptable.

3/4.7.1.6 Safety Valves -- Shutdown. (Specification carryover from existing TS) This specification[30] is a clarifying upgrade of existing Specification SR 5.3.9 to which has been added an LCO. One operable safety valve is required for each operating section of the steam generator. A brief functional analysis is provided in Section 4.8 of Ref. 3. The specification is judged to be acceptable.

3/4.7.1.7 Condensate Pumps. (Specification adapted from STS) There are four condensate pumps at FSV. Two are 60% capacity; two are 12 1/2%

capacity. Only the small pumps are addressed in the TS upgrade. The large condensate pumps require electrical power from the main generator or offsite power and can only draw water from the condenser hot well. The small (12 1/2% capacity) condensate pumps can be aligned to obtain water from the condenser hot well, the two condensate storage tanks, the (helium circulator) turbine water drain tank via the hot well, or the decay heat removal exchanger and can provide flow directly to either the main condensate line or the emergency condensate header. The large pumps can also provide flow to the emergency condensate header but through a more circuitous route than that of the small pumps. The small condensate pumps are an automatically sequenced essential load of the emergency diesel generators. The small condensate pumps are relied upon during loss of offsite power with main turbine-generator trip to provide either condensate flow and net positive suction head for at least one turbine-driven boiler feed pump operating on reactor-generated steam (FSAR Sections 10.3.1, 14.3.6.5, and 14.4.2) or both direct steam generator cooling and helium circulator water turbine drive via the emergency condensate header (FSAR Sections 10.3.2, 14.3.6.6, and 14.4.2.1 [Case B1]). Only one small condensate pump is required to be operable above 5% power since the Safe Shutdown Cooling System provides an environmentally and seismically qualified alternate success path. The role of the condensate pumps is addressed functionally in Ref. 3. The specification is judged to be acceptable.

3/4.7.4 Service Water System

3/4.7.4.1 Service Water System -- Operating. (Specification carryover from existing TS) This specification is an upgrade, combination, and expansion of existing Specifications LCO 4.2.4 and SR 5.2.24.e. The upgraded TS addresses not only the redundant service water pumps that were covered in the existing TS, but also the pump inlet supply flow paths from the circulating water makeup system and the pump outlet supply flow paths to essential service water users. The action statements also account for providing backup capability from the diverse and redundant seismically qualified firewater system. All service water components that are addressed in the existing and upgraded TS for the service water system are seismically qualified Class I components. With the exception of certain nonseismically qualified safety-related equipment (service water cooling tower fans and return pumps) that is part of the Alternate Cooling Method (ACM) and is surveilled per upgraded Specification SR 4.8.4.e.2 and Table 4.8.4-2, the non-Class I portions of the normal service water system are not addressed in the TS. When the reactor is operating above 5% of rated reactor power or otherwise when the Calculated Bulk Core Temperature exceeds 760°F, two of the three service water pumps must be operable, and each flowpath must be operable or have a backup flow path operable within 72 hours. A brief functional analysis is provided in Section 4.6.3 of Ref. 3. The functional analysis addresses the functions of both the Class I and non-Class I configurations of the service water system. The specification is judged to be acceptable.

<u>3/4.7.4.2 Service Water System - Shutdown</u>. (Specification adapted from STS) Although there is no equivalent specification in the W-STS for service water under equivalent (cold) shutdown conditions, the FSV service water system is the Class I primary success path for assuring PCRV cooling during shutdown with firewater backup as the alternate Class I success path. Also normally, below 5% fission power when the Calculated Bulk Core Temperature is less than 760°F, service water is the heat sink for residual heat removal through the PCRV LCS. One pump with operable flow paths is required to be operable. The specification is judged to be acceptable.

<u>3/4.7.5 Primary Coolant Depressurization</u>. (Specification carryover from existing TS) This specification is an upgrade and expansion of existing Specifications LCO 4.2.12, LCO 4.2.18, and SR 5.2.24.g. The specification requires that two operable flow paths for reactor depressurization and primary coolant filtration be available through the Helium Purification System (HPS) to accommodate the permanent loss of forced cooling accident (FSAR Sections 9.4.3.3.2, 9.6.6, 9.7.3.4, and 14.10.2 and Appendix D.1). The HPS, normally operates to maintain (1) primary coolant radioactivity within the limits of upgraded Specification LCO 3.4.2 and (2) primary coolant impurities within the limits of upgraded Specifications LCO 3.4.3 or LCO 3.4.4. During the loss of forced cooling accident, the HPS also provides for a filtered depressurization

within the time allowed and under the conditions specified in upgrade Specifications LCO 3.2.4 and upgraded LCO 3.4.1. The filtered depressurization mitigates the dose consequences of the permanent loss of forced cooling accident (FSAR Section 14.10 and Appendix D) by (1) removing circulating activity that existed in the primary coolant before the accident, (2) eliminating the pressure differential that could drive released fission products out of the PCRV if a leak developed in the pressurized PCRV after core heatup and fuel damage had occurred as a result of the accident, and (3) eliminating convective heat transfer as a source of potentially excessive and damaging heat loads on the upper PCRV thermal barrier and liner during the core heatup event. The upgraded specification provides for two operable trains of the HPS whenever the core is generating fission heat or otherwise when the Calculated Bulk Core Temperature exceeds 760°F. This is conservative for accidents initiated below an equilibrium power level of 35% of rated reactor power because recent analysis (FSAR Appendix D.4) has shown that very little fuel damage occurs for the permanent loss of forced cooling under this condition even if the PCRV remains pressurized. The specification is judged to be acceptable.

3/4.7.6 Fire Suppression Systems

3/4.7.6.1 Spray and/or Sprinkler Systems. (Specification carryover from existing TS) This specification is an upgrade of existing Specifications LCO 4.10.5 and SR 5.10.6 and includes an updated complete listing of areas serviced by the spray/sprinkler systems consistent with FSAR Section 9.12 (Revision 5). Operability requirements have been expanded to apply at all times instead of merely during power operation as provided in the existing TS. The upgraded TS does not require special reports as an action as provided in W-STS LCO/SR 3/4.7.11.2 for sprinkler/spray component inoperabilities, but a continuous fire watch with backup equipment is required for sprinkler/spray inoperabilities in areas containing safety-related equipment. If an evaluation determines that no safety-related equipment is affected by the inoperability, no fire watch is required. If redundant equipment is not also affected based on evaluation, an hourly fire watch is permitted. W-STS requirements have been adapted for the upgraded surveillances. The assurance of firewater supply to the spray/sprinkler system is addressed in upgraded Specification LCO 3.5.5, Safe Shutdown Cooling Water Supply System, because of the redundant roles of firewater as the Class I source for emergency forced circulation cooling of the reactor, for PCRV (containment) heat removal and for backup service water for cooling of essential equipment such as the emergency diesel generators. This specification is judged to be acceptable.

<u>3/4.7.7 Fire Rated Barriers</u>. (Specification carryover from existing TS) This specification is an upgrade of existing Specifications LCO 4.10.4 and SR 5.10.4 to more closely resemble W-STS LCO/SR 3/4.7.12, Fire Rated Assemblies. The upgraded TS clarifies applicability at all times. Unlike the W-STS equivalent, no special reporting is required as an action. Loss of integrity in a fire barrier requires either establishing a continuous fire watch or verifying the operability of local fire detectors, both of which actions are equivalent to the W-STS. At FSV, an alternative is that an evaluation and determination can be made that the affected equipment separated by the barrier is not required to be operable. The surveillance, although reformatted, has been adapted from and is equivalent to that of the W-STS. This specification is judged to be acceptable.

<u>3/4.10 Special Test Exception</u>

<u>3/4.10.1 Xenon Stability</u>. (Specification carryover from existing TS) The specification is an upgrade for the recently approved (License Amendment No. 47) existing Specification LCO 4.9.3. The engineering evaluation performed prior to xenon stability testing is to be reviewed and approved by the NFSC under upgraded Specification AC 6.5.2.9.d. Results of previous analysis and tests are summarized in FSAR Sections 3.5.4.4, 13.4.2.10, and 13.4.2.11. Although the B-series startup test procedures have not been submitted for NRC review, given the NFSC oversight responsibilities, the upgraded specification is judged to be acceptable.

5. DESIGN FEATURES

5.3 Reactor Core

5.3.4 Reload Segment Design. (Specification carryover from existing TS) The specification is an upgrade that adapts into the W-STS design features format much of the information that was formally contained in the basis summary statement of existing Specification LCO 4.1.3. Existing Specification LCO 4.1.3, Rod Sequence, has been suspended during recent operation under the Interim Specifications for Reactivity Control Systems; however, the Interim Specifications did not address the analysis and evaluation performed outside the scope of the specification to establish an acceptable control rod pair withdrawal sequence. Specification DF 5.3.4 of the FSV upgraded TS reiterates the acceptance criteria for the control rod pair withdrawal sequence as formerly given in the cited basis summary statement from the existing TS and adds the general requirements for new fuel and burnable poison loadings that were not specifically addressed in the existing TS but are provided in FSAR Sections 3.5.1 through 3.5.6.

In the proposed upgraded TS for FSV, the cycle-dependent control rod withdrawal sequence (which is hard-wired and checked before reload cycle startup), the reload segment fuel loading and distribution and the reload segment burnable poison loading and distribution are broadly defined to be among "those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety" as used to define design features per 10CFR Part 50.36(c)(4). Under the provisions of 10CFR Part 50.59, review, audit, and approval of the reload segment nuclear performance characteristics against the acceptance criteria and requirements for the reload segment design feature have been vested in the NFSC per the provisions of upgraded Specifications AC 6.5.2.8.a, AC 6.5.2.8.f, AC 6.5.2.9.a, AC 6.5.2.9.b, AC 6.5.2.9.d, AC 6.5.2.10.a, AC 6.5.2.10.c, and AC 6.5.2.10.d. Records of the NFSC review, audit, and approval process are maintained for NRC inspection per upgraded Specification AC 6.5.2.12.

As indicated above, the reload segment design section is new to the Design Features section, and the format has been adapted from the W-STS DF 5.3.1. It identifies acceptance criteria for rod pair reactivity worths, shutdown margin, and power peaking limits such that reload configurations are designed to be compatible with existing FSAR safety analysis. The only relation to W-STS DF 5.3.1 is that both relate to reload fuel; otherwise, the FSV TSUP information is unique to FSV. The power peaking information presented is somewhat analogous to the information required on F_{xy} in W-STS AC 6.9.1.6, Radial Peaking Factor Limits Report. However, the FSV fuel damage limits have only been related to the gross peaking factors of Region Peaking Factor, Axial Peaking Factor (actually top and bottom zone peaking, see FSAR Section 3.5.1.2), and Intra-Region Peaking Factor rather than the more detailed F_{xy} specifications on core region outlet temperature, in upgraded Specification 3/4.2.4, and power-to-flow ratio limits, in upgraded Specification 3/4.2.6. These gross peaking factor design limits are as stated to ensure the validity of power peaking assumptions in the safety analysis of FSAR Sections 3.2.3.1, 3.5.4.1, 3.5.4.2, 3.5.4.3, 3.6.4.6, and 3.6.4.7. Controlling on the more global power distribution limits and region outlet temperatures, and power-to-flow ratio limits has been possible because of the large margin between the nominal maximum fuel temperature of 2372°F and that of the 2900°F which must be experienced for long periods before the onset of rapid deterioration of the fuel particle fission product barrier (FSAR Section 3.2.3.3).

Given the NFSC oversight responsibility discussed above, the specification is judged to be acceptable.

4. CONCLUSIONS

As discussed in detail in Section 3.0 of this TER with regard to each specification that has been reviewed by ORNL, the upgraded TS have been found to be technically acceptable. A summary of the major findings of the integrated reviews by INEL and ORNL is being written by INEL.

5. REFERENCES

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3. D. L. Moses, Technical Evaluation Report: The Acceptability of Fort St. Vrain Upgraded Technical Specifications for Structures, Systems, and Components with Safety-Related and Important-to-Safety Cooling Functions, Oak Ridge National Laboratory, July 1988.

4. Fort St. Vrain Nuclear Generating Station Updated Final Safety Analysis Report, Docket No. 50-267, Public Service Company of Colorado, updated through Revision 5, July 22, 1987.

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Appendix A

Scope and Guidelines for the Fort St. Vrain (FSV) Technical Specification Upgrade Program (TSUP)

I. Commitments by Public Service Company of Colorado (PSC) plus clarifying comments in brackets.

P-1 Limiting Conditions for Operation (LCOs) will be revised to identify the applicable operating modes, limiting condition and action statement defining remedial actions to be taken if the limiting condition is exceeded.

P-2 All applicable operating modes will be clearly identified for each LCO.

P-3 Limiting conditions and action statements will agree with Final Safety Analysis Report (FSAR) accident and safety analyses. [However, for purposes of the TSUP, the interpretation has been made that licensing bases will not be required to be added to the UFSAR at this time for those Technical Specifications that lack such documentation. See 10CFR Parts 50.34(a)(5) and (b), 50.36(b), and 50.71(e).]

P-4 LCOs will cross reference the applicable Surveillance Requirement (SR) and SRs will cross reference the applicable LCO. All LCOs will have associated with them one or more SRs and all SRs will have associated with them one or more LCOs.

P-5 Surveillances will be specified as necessary and sufficient to verify compliance with the associated LCO(s).

P-6 SRs will describe the associated acceptance criteria.

P-7 Technical Specification statements will be unambiguous with a singular interpretation.

P-8 Terminology used in the Technical Specifications will be clearly defined.

P-9 Technical Specifications will be simplified if possible.

P-10 LCO format will be revised to include the LCO number and title, applicability statement, LCO statement, action statement(s), and cross reference to the associated SR(s). [Format has actually adhered more closely to that of the Standard Technical Specifications than implied by this scope item.]

P-11 Technical Specifications will be reviewed and expanded as necessary to assure accuracy, completeness, and consistency with existing design and safety analysis documentation. [However, for purposes of the TSUP, the interpretation has been made that licensing bases will not be required to be added to the UFSAR at this time for those Technical Specifications that lack such documentation. See 10CFR Parts 50.34(a)(5) and (b), 50.36(b), and 50.71(e).]

P-12 The Technical Specifications will account for and utilize existing plant equipment and safety systems. [This includes equipment changes made during the TSUP reviews such as the 6-inch vent lines installed on the main steam lines to meet Environmental Qualification requirements.]

P-13 The initial draft of the upgraded Technical Specifications will be submitted to the Nuclear Regulatory Commission (NRC) by April 1, 1985. [Completed.]

P-14 A schedule for the Technical Specification Upgrade Program will be submitted to the NRC for information by December 14, 1984. [Completed.]

II. NRC Changes and Clarifications Regarding the FSV TSUP Scope.

N-1 ANSI/ANS Standard 58.4 (1979 Edition), Criteria for Technical Specifications for Nuclear Power Station, will be used for guidance regarding the content of the Technical Specifications.

N-2 The bases for the Technical Specifications will be included in the upgrade effort. [However, for purposes of the TSUP, this has been interpreted by NRC to refer restrictively to the "summary statement of the bases or reasons for such specifications" per 10CFR Part 50.36(a) but not to the FSAR "analyses and evaluations" from which the Technical Specifications are to be derived per 10CFR Parts 50.34(a)(5) and (b), 50.36(b), and 50.71(e)].

N-3 The Standard Technical Specifications for Westinghouse pressurized water reactors (PWRs) will be used as guidance, where applicable, in performing the upgrade.

N-4 A thorough review of the FSV FSAR and other relevant design documentation will be done to ensure the Technical Specifications are complete and correct. ["Correctness" of the Technical Specifications was interpreted by NRC not to mean that the FSV UFSAR had to be updated at this time to support the carryover provisions of existing Specification that lack a formally documented licensing basis. See 10CFR Parts 50.34(a)(5) and (b), 50.36(b), and 50.71(e).]

N-5 Operating experience to date will be considered and factored into the upgrade effort.

N-6 The need for additional instrumentation or other hardware to ensure compliance with the upgraded Technical Specifications will be considered on a case-by-case basis.

N-7 Any hardware change, analytical effort, or developmental work, which may be proposed by PSC as a result of the upgrade effort, will be scheduled for completion at a later date if it cannot be done by July 1, 1985.

III. PSC Guidelines for Use of the Standard Technical Specifications (STS).

G-1 Plant modifications and backfits would not be undertaken to permit the adoption of any Standard Technical Specification requirement.

G-2 It is outside the scope of the Fort St. Vrain Technical Specification Upgrade Program to utilize or consider any Standard Technical Specification requirement which opens the licensing basis of the Fort St. Vrain plant for further justification or analysis.

G-3 Significant research and development efforts or analytical investigations would not be undertaken to determine how to utilize, or whether or not a Standard Technical Specification requirement can be utilized at Fort St. Vrain. Questionable Standard Technical Specifications requiring such efforts and investigations would not be utilized or given further consideration.

G-4 The numbering system of the Standard Technical Specifications and the Standard Technical Specification format, whereby each LCO is juxtaposed to its associated SR, would not be utilized for the Fort St. Vrain Technical Specification Upgrade Program. [Withdrawn, or at least not adhered to.]

G-5 The Fort St. Vrain Technical Specification Upgrade Program will adopt relevant Standard Technical Specification definitions where the definitions are consistent with existing plant features and the licensing basis of the Fort St. Vrain plant, i.e., FSAR terminology and analyses.

G-6 The Fort St. Vrain Technical Specification Upgrade Program will adopt relevant Standard Technical Specification requirements, including limiting conditions for operation, surveillance requirements, and surveillance frequencies, which are consistent with existing plant features and the licensing basis of the Fort St. Vrain plant as embodied in the FSAR.

G-7 Each Fort St. Vrain Upgraded Technical Specification requirement need only be supported and justified relative to the licensing basis of the Fort St. Vrain plant as embodied in the FSAR, and justification would not be required regarding the Standard Technical Specification treatment of the same or similar requirements for light water reactor plan