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Georgia Power
the southern electric system

February 25, 1994

Docket Nos. 50-321
50-366

HL-4495

TAC Nos. M87310
M87311

Edwin I. Hatch Nuclear Plant
Request to Revise Technical Specifications
Conversion to Improved Standard Technical Specifications
Consistent With NUREG 1433

Gentlemen:

In accordance with the provisions of 10 CFR 50.90, as required by 10 CFR 50.59(c)(1), Georgia Power Company (GPC) hereby proposes changes to the Plant Hatch Units 1 and 2 Technical Specifications, Appendix A to Operating Licenses DPR-57 and NPF-5.

This submittal proposes to revise the Units 1 and 2 Technical Specifications in their entirety to be consistent with NUREG 1433, "Standard Technical Specifications General Electric Plants, BWR/4." The proposed changes also include technically justified deviations from the NUREG and technically justified changes to the current licensing basis. The following documents are attached:

1. Application of NRC Final Policy Statement Selection Criteria (Units 1 and 2).
2. Proposed Hatch Technical Specifications (Units 1 and 2).
3. Proposed Hatch Technical Specifications Bases (Units 1 and 2).
4. Markup of the current Technical Specifications with a discussion of the proposed changes (Units 1 and 2).
5. No Significant Hazards Determination for the proposed changes.

In addition, a markup of NUREG 1433 and justifications for GPC deviations from the NUREG are provided as part of this submittal. Please note the industry and NRC proposed/accepted changes to the NUREG through January 31, 1994, are reflected in the NUREG markup. Any deviations from the proposed/accepted changes are justified on a plant-specific basis. The enclosure to this letter is a synopsis of the information contained in each attachment, including definition of pertinent designators, e.g., "A" for Administrative change.

A001 1/1 Limited List

Georgia Power

Implementation of the proposed amendments is tentatively scheduled for March 30, 1995. This date is based on the training schedules for both licensed and nonlicensed personnel, the timing of implementation with respect to refueling outages, licensed operator examination schedules, and the time required for procedure revisions and development of new programs. This date is also predicated on issuance of an SER in Fall 1994.

Although the new Standard Technical Specifications contain many improvements, they also impose a number of new surveillance requirements which have not been performed at Plant Hatch. GPC intends to treat these new surveillance requirements as "met" at the time of implementation of the new Technical Specifications, with the first test to be performed within the required frequency from the implementation date. Any revisions to the Emergency Plan or Final Safety Analysis Reports (FSAR) necessitated by the conversion to the Improved Standard Technical Specifications will be submitted in accordance with the requirements of 10 CFR 50.54.

Georgia Power Company requests to meet with you at your earliest convenience to discuss a review schedule and the contents of this submittal package.

In accordance with the requirements of 10 CFR 50.91, a copy of this letter and all applicable enclosures will be sent to the designated State official of the Environmental Protection Division of the Georgia Department of Natural Resources.

Mr. J. T. Beckham, Jr. states he is Vice President of Georgia Power Company and is authorized to execute this oath on behalf of Georgia Power Company, and to the best of his knowledge and belief, the facts set forth in this letter are true.

GEORGIA POWER COMPANY

BY: J. T. Beckham, Jr.
J. T. Beckham, Jr.

Sworn to and subscribed before me this 25 *day of* February, *1994.*

Sandra H. Morgan
Notary Public

SRM/cr

Enclosure: Improved Technical Specifications Submittal Synopsis

Attachments:

1. Application of Selection Criteria
2. Unit 1 Improved Technical Specifications
3. Unit 2 Improved Technical Specifications
4. Unit 1 Improved Bases
5. Unit 2 Improved Bases
6. Unit 1 Markup of Current Technical Specifications and Discussion of Changes
7. Unit 2 Markup of Current Technical Specifications and Discussion of Changes
8. Unit 1 No Significant Hazards Determination
9. Unit 2 No Significant Hazards Determination
10. NUREG 1433 Comparison Document - Specifications
11. NUREG 1433 Comparison Document - Bases
12. NUREG 1433 Comparison Document - Justification for Deviation

cc: *Georgia Power Company*
Mr. H. L. Sumner, Nuclear Plant General Manager
NORMS

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. K. Jabbour, Licensing Project Manager - Hatch
Mr. C. Grimes, Technical Specifications Branch

U.S. Nuclear Regulatory Commission, Region II
Mr. S. D. Ebneter, Regional Administrator
Mr. L. D. Wert, Senior Resident Inspector - Hatch

State of Georgia
Mr. J. D. Tanner, Commissioner - Department of Natural Resources

ENCLOSURE
IMPROVED TECHNICAL SPECIFICATIONS SUBMITTAL SYNOPSIS

The Edwin I. Hatch Nuclear Plant Conversion to Improved Technical Specifications (ITS) submittal consists of 12 documents. Below is a listing of the volumes and a brief description of the volume contents. In addition, a brief explanation of how the material was prepared and the designations utilized is included.

UNITS 1 AND 2 APPLICATION OF SELECTION CRITERIA

This volume provides, for each unit, a discussion of how the NRC Final Policy Statement was applied to the current Units 1 and 2 Technical Specifications. Also, included for each unit, is a matrix cross referencing the following documents: 1) the current Technical Specifications (CTS); 2) the Standard Technical Specifications, or STS Rev. 4, (Unit 1 only); 3) the proposed Technical Specifications, where applicable; and 4) the Final Policy Statement Selection criteria. For the current Technical Specifications that do not meet any of the criteria and are not retained in the ITS, an explanation of why each Specification does not meet the selection criteria is provided (Appendices A and B). These Specifications are proposed to be relocated to owner-controlled documents.

UNIT 1 SPECIFICATIONS
UNIT 2 SPECIFICATIONS

These volumes contain only the proposed Technical Specifications for Units 1 and 2 in the NUREG 1433 format. The current Specifications requirements that are to be relocated are not included in these volumes.

UNIT 1 BASES
UNIT 2 BASES

These volumes contain the proposed Technical Specifications Bases for Units 1 and 2. Information regarding the basis for each Specification, as well as details of what comprises OPERABLE subsystems, is provided.

UNIT 1 MARKUP OF CURRENT TECHNICAL SPECIFICATIONS AND DISCUSSION OF PROPOSED CHANGES (2 Volumes)
UNIT 2 MARKUP OF CURRENT TECHNICAL SPECIFICATIONS AND DISCUSSION OF PROPOSED CHANGES (2 Volumes)

These volumes contain annotated copies of the CTS to show the disposition of the existing requirements into the ITS. The pages are in ITS order, e.g., ITS 3.5.1 (ECCS - Operating) contains all appropriate pages from current ECCS requirements, as well as the LPCI inverter requirements in CTS Section 3.8, since the LPCI inverter requirements will now reside in ITS Section 3.5.

Each CTS page is annotated with the ITS Specification number at the top of the page, reflecting the ITS location of the CTS requirements. Items on the CTS page, which are located in ITS Specifications other than that referenced at the top of the page, are noted adjacent to the items. Where the ITS requirement differs from the CTS requirement, the individual details of the CTS being revised are annotated with alpha-numeric designators which relate to an appropriate Discussion of Change (DOC). The DOC provides a concise justification for the change outlined. The DOCs associated with each ITS section are located after the marked-up CTS pages. The alpha-numeric designators also relate to the no significant hazards determination (NSHD) evaluations contained in another volume of the submittal.

The current Plant Hatch Technical Specifications Bases pages are not individually annotated, since they are being replaced in their entirety. However, so that all current Technical Specifications pages can be accounted for, a page has been inserted behind the last DOC. This page details which CTS pages have not been included which are the following; CTS Bases pages and pages which indicate "left blank".

Current Technical Specifications pages containing requirements in more than one ITS Specification are repeated in the appropriate section(s). In this instance, portions of a single CTS Specification may relate to more than one ITS requirement, and as such, the requirement is contained within several ITS Specifications. For these CTS requirements, the CTS page is repeated for each ITS Specification, and may have differing alpha-numeric designators (with corresponding different DOCs), e.g., Unit 1 CTS Table 3.2.8, Functions 1 and 3, are located in three ITS LCOs (3.3.6.1, 3.3.6.2, and 3.10.1). Thus, the appropriate CTS pages are found in all three ITS sections of the CTS markups.

The alpha-numeric designators relating to the CTS changes are numbered sequentially within each letter category and within each ITS Specification. The changes for each CTS are separated into the following categories:

Designator	Category
A	ADMINISTRATIVE - associated with restructuring, interpretation, and complex rearranging of requirements, and other changes not substantially revising an existing requirement.
R	RELOCATED - specific requirements that do not meet the NRC Final Policy Statement selection criteria.

Designator	Category
M	MORE RESTRICTIVE - changes to the CTS being proposed in converting to the ITS, resulting in added restrictions or eliminating flexibility.
LA, LB, LC	LESS RESTRICTIVE - justified with a single No Significant Hazards Determination (NSHD). The "LA" changes consist of elimination of detail from the CTS. Typically, this involves details of system design and function, or procedural detail on methods of conducting a surveillance. "LB" changes are related to the extension of allowed outage times (AOTs) and surveillance test intervals (STIs) conducted in accordance with GE NRC approved Topical Reports. "LC" changes reflect elimination of various instrumentation requirements, where the instrument is an alarm or indication-only instrument that does not otherwise meet the NRC Final Policy Statement selection criteria.
L	LESS RESTRICTIVE - those in which requirements are relaxed or eliminated, or new flexibility is provided. Each Less Restrictive change has a unique corresponding NSHD (while there is a single NSHD for each of the other categories).

The DOCs associated with the LA, LB and LC Less Restrictive changes are labeled "Generic" and are discussed first after the Administrative, Relocated, and More Restrictive changes. The plant specific changes are labeled "Specific" and follow the "Generic" changes.

UNIT 1 NO SIGNIFICANT HAZARDS DETERMINATION
UNIT 2 NO SIGNIFICANT HAZARDS DETERMINATION

These volumes, containing the 10 CFR 50.92 required NSHDs, show that the proposed changes do not contain any significant hazard considerations that would indicate the change should not be made. Many of the proposed changes are grouped together based on similarities; e.g., requirements that are administrative, certain types of less restrictive changes, more restrictive changes, and requirements proposed to be relocated. "Generic" NSHD evaluations are presented for these categories.

The first section of the NSHD volume contains an NSHD for each of the "generic" categories of changes (generic categories are all categories except category "L"). A single NSHD applies to all numbered changes of that type (e.g., a single "LA" NSHD is provided for all "LA.x" labeled changes). These generic NSHDs are ordered alphabetically (i.e., A, LA, LB, LC, M, and R). Additionally, the generic Environmental Assessment is located at the end of this section.

The NSHD sections following the generic NSHDs contain separate NSHDs corresponding to each "L" DOC, and are provided in ITS order, consistent with the CTS markup and DOCs. For instance, an "L.1" change noted in the CTS markup for ITS LCO 3.5.1 has a unique NSHD labeled "L.1" in the section for ITS LCO 3.5.1 NSHDs.

UNIT 1 AND UNIT 2 NUREG 1433 COMPARISON DOCUMENT (3 Volumes)

These volumes contain a copy of NUREG 1433 (Specifications and Bases) annotated to compare the NUREG to the proposed Plant Hatch Improved Technical Specifications. The annotations include any deviations made in either Unit's proposed Specifications. The justifications for each deviation from the NUREG are found in the third volume.

A deviation affecting only one Unit is indicated beside the deviation (e.g., Unit 1 only). For Secondary Containment Section 3.6, a Unit 1 markup followed by a Unit 2 markup is provided, because the two Units differ in design.

For the purposes of determining deviations from NUREG 1433, the NUREG is considered to be NUREG 1433, dated 9/28/92, as modified by all generic changes submitted as of 1/31/94, whether or not they have been accepted, consistent with the generic change Matrix Rev. 7, dated 1/31/94.

There are two types of deviations identified -- Plant Specific and Generic. The plant-specific deviations are annotated by a "P," and the deviations due to generic changes in the NUREG are annotated with either a "GA" for generic accepted or "GP" for generic pending. The plant-specific deviations cover all deviations that have not been submitted as a generic change. Both types of deviations (plant specific and generic) are numbered sequentially.

All generic changes, accepted and pending, adopted by Plant Hatch are indicated on the applicable pages and annotated with a "GA" or "GP". The discussion of the origin of the generic change (e.g., NUREG change package BWR-18, Item C.1) is included in the Justification for Deviation section. A single "G" designator identifies each NUREG change package (e.g., BWR-18).

Plant Hatch's rationale for not adopting certain generic changes is based on plant-specific factors, which are annotated in the NUREG Comparison. These annotations follow one of the following formats:

a. A non-adopted generic change that is small (i.e., a few words) and does not make the markup too difficult to read is annotated and lined out. A "P" designator is placed beside the line-out. The Justification for Deviation identifies the generic change package and Plant Hatch's rationale for not adopting the change.

b. A large generic change is annotated by indicating the NUREG change package number (e.g., Note added by BWR-18, Item C.1). The change is lined out, and a "P" number is placed beside the line out. The Justification for Deviation identifies the generic change package and Plant Hatch's rationale for not adopting the change.

c. If the generic change is made to a section that has been deleted entirely (e.g., an LCO, ACTION, SR, or the applicable Bases), the deviation is identified in the markup near the lined-out section with words similar to "also modified by BWR-18 Item C.1." The "P" deviation states the reason the original NUREG requirement was not adopted, and thus, did not adopt the generic change. (The entire requirement is being deleted; thus, it appears unnecessary to state that a change to the NUREG is also not adopted.)

APPLICATION OF SELECTION CRITERIA TO THE

EDWIN I. HATCH NUCLEAR PLANT
UNIT 1
TECHNICAL SPECIFICATIONS

CONTENTS

1. INTRODUCTION

The purpose of this document is to confirm the results of the BWR Owners' Group application of the Technical Specifications selection criteria on a plant specific basis for Edwin I. Hatch Nuclear Plant Unit 1. Georgia Power Company has reviewed the application of the selection criteria to each of the Technical Specifications utilized in BWROG report NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment", including Supplement 1 (Reference 1), and NUREG 1433, "Standard Technical Specifications, General Electric Plants BWR/4," (Reference 2), as well as applying the criteria to each of the current Plant Hatch Unit 1 Technical Specifications. Additionally, in accordance with the NRC guidance, this confirmation of the application of selection criteria to Plant Hatch Unit 1 includes confirming the risk insights from PRA evaluations, provided in Reference 1, as applicable to Plant Hatch Unit 1.

2. SELECTION CRITERIA

Georgia Power Company (GPC) has utilized the selection criteria provided in the NRC Final Policy Statement on Technical Specification Improvements (52 FR 3788) of July 23, 1993 (Reference 3) to develop the results contained in the attached matrix. Probabilistic Risk Assessment (PRA) insights as used in the BWROG submittal were utilized, confirmed by GPC, and are discussed in the next section of this report. The selection criteria and discussion provided in the NRC Final Policy Statement are as follows:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary:

Discussion of Criterion 1: A basic concept in the adequate protection of the public health and safety is the prevention of accidents. Instrumentation is installed to detect significant abnormal degradation of the reactor coolant pressure boundary so as to allow operator actions to either correct the condition or to shut down the plant safely, thus reducing the likelihood of a loss-of-coolant accident.

This criterion is intended to ensure that Technical Specifications control those instruments specifically installed to detect excessive reactor coolant system leakage. This criterion should not, however, be interpreted to include instrumentation to detect precursors to reactor coolant pressure boundary leakage or instrumentation to identify the source of actual leakage (e.g., loose parts monitor, seismic instrumentation, valve position indicators).

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier:

Discussion of Criterion 2: Another basic concept in the adequate protection of the public health and safety is that the plant shall be operated within the bounds of the initial conditions assumed in the existing Design Basis Accident and Transient analyses and that the plant will be operated to preclude unanalyzed transients and accidents. These analyses consist of postulated events, analyzed in the FSAR, for which a structure, system, or component must meet specified functional goals. These analyses are contained in Chapters 6 and 15 of the FSAR (or equivalent chapters) and are identified as Condition II, III, or IV events (ANSI N18.2) (or equivalent) that either assume the failure of or present a challenge to the integrity of a fission product barrier.

As used in Criterion 2, process variables are only those parameters for which specific values or ranges of values have been chosen as reference bounds in the Design Basis Accident or Transient analyses and which are monitored and controlled during power operation such that process values remain within the analysis bounds. Process variables captured by Criterion 2 are not, however, limited to only those directly monitored and controlled from the control room. These could also include other features or characteristics that are specifically assumed in Design Basis Accident and Transient analyses even if they cannot be directly observed in the control room (e.g., moderator temperature coefficient and hot channel factors).

The purpose of this criterion is to capture those process variables that have initial values assumed in the Design Basis Accident and Transient analyses, and which are monitored and controlled during power operation. As long as these variables are maintained within the established values, risk to the public safety is presumed to be acceptably low. This criterion also includes active design features (e.g., high pressure/low pressure system valves and interlocks) and operating restrictions (pressure/temperature limits) needed to preclude unanalyzed accidents and transients.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier:

Discussion of Criterion 3: A third concept in the adequate protection of the public health and safety is that in the event that a postulated Design Basis Accident or Transient should occur, structures, systems, and components are available to function or to actuate in order to mitigate the consequences of the Design Basis Accident or Transient. Safety sequence analyses or their equivalent have been performed in recent years and provide a method of presenting the plant response to an accident These can be used to define the primary success paths.

A safety sequence analysis is a systematic examination of the actions required to mitigate the consequences of events considered in the plant's Design Basis Accident and Transient analyses, as presented in Chapters 6 and 15 of the plant's FSAR (or equivalent chapters). Such a safety sequence analysis considers all applicable events, whether explicitly or implicitly presented. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criteria), so that the plant response to Design Basis Accidents and Transients limits the consequences of these events to within the appropriate acceptance criteria.

It is the intent of this criterion to capture into Technical Specifications only those structures, systems, and components that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path for a particular mode of operation does not include backup and diverse equipment (e.g., rod withdrawal block which is a backup to the average power range monitor high flux trip in the startup mode, safety

Discussion of Criterion 3: (continued)
valves which are backup to low temperature overpressure relief valves during cold shutdown).

Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety:

Discussion of Criterion 4: It is the Commission policy that licensees retain in their Technical Specifications LCOs, actions statements, and Surveillance Requirements for the following systems (as applicable), which operating experience and [probabilistic safety assessment (PSA)] PSA have generally shown to be significant to public health and safety and any other structures, systems, or components that meet this criterion:

- Reactor Core Isolation Cooling/Isolation Condenser,
- Residual Heat Removal,
- Standby Liquid Control, and
- Recirculation Pump Trip.

The Commission recognizes that other structures, systems, or components may meet this criterion. Plant- and design-specific PSAs have yielded valuable insight to unique plant vulnerabilities not fully recognized in the safety analysis report Design Basis Accident or Transient analyses. It is the intent of this criterion that those requirements that PSA or operating experience exposes as significant to public health and safety, consistent with the Commission's Safety Goal and Severe Accident Policies, be retained or included in Technical Specifications.

The Commission expects that licensees, in preparing their Technical Specification related submittals, will utilize any plant-specific PSA or risk survey and any available literature on risk insights and PSAs. This material should be employed to strengthen the technical bases for those requirements that remain in Technical Specifications, when applicable, and

Discussion of Criterion 4: (continued)

to verify that none of the requirements to be relocated contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk. Similarly, the NRC staff will also employ risk insights and PSAs in evaluating Technical Specifications related submittals. Further, as a part of the Commission's ongoing program of improving Technical Specifications, it will continue to consider methods to make better use of the risk and reliability information for defining future generic Technical Specification requirements.

3. PROBABILISITIC RISK ASSESSMENT INSIGHTS

Introduction and Objectives

The Final Policy Statement includes a statement that NRC expects licensees to utilize the available literature on risk insights to verify that none of the requirements to be relocated contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk.

Those Technical Specifications proposed for relocation to other plant controlled documents will be maintained under the 10 CFR 50.59, safety evaluation review program. These Specifications have been compared to a variety of Probabilistic Risk Assessment (PRA) material with two purposes: 1) to identify if a component or variable is addressed by PRA, and 2) to judge if the component or variable is risk-important. In addition, in some cases risk was judged independent of any specific PRA material. The intent of the review was to provide a supplemental screen to the deterministic criteria. Those Technical Specifications proposed to remain a part of the Improved Technical Specifications were not reviewed. This review was accomplished in Reference 1, except where discussed in Appendix A, "Justification For Specification Relocation", and has been confirmed by GPC for those Specifications to be relocated. Where Reference 1 did not review a Technical Specification against the criteria of Reference 2, GPC performed a review similar (but not identical) to that described below for Reference 1. The results of these reviews are presented in Appendix B.

Assumptions and Approach

Briefly, the approach used in Reference 1 was the following:

> The risk assessment analysis evaluated the loss of function of the system or component whose LCO was being considered for relocation and qualitatively assessed the associated effect on core damage frequency and offsite releases. The assessment was based on available literature on plant risk

insights and PRAs. Table 3-1 lists the PRAs used for making the assessments. A detailed quantitative calculation of the core damage and offsite release effects was not performed. However, the analysis did provide an indication of the relative significance of those LCOs proposed for relocation on the likelihood or severity of the accident sequences that are commonly found to dominate plant safety risks. The following analysis steps were performed for each LCO proposed for relocation:

a. List the function(s) affected by removal of the LCO item.

b. Determine the effect of loss of the LCO item on the function(s).

c. Identify compensating provisions, redundancy, and backups related to the loss of the LCO item.

d. Determine the relative frequency (high, medium, and low) of the loss of the function(s) assuming the LCO item is removed from Technical Specifications and controlled by other procedures or programs. Use information from current PRAs and related analyses to establish the relative frequency.

e. Determine the relative significance (high, medium, and low) of the loss of the function(s). Use information from current PRAs and related analyses to establish the relative significance.

f. Apply risk category criteria to establish the potential risk significance or non-significance of the LCO item. Risk categories were defined as follows:

RISK CRITERIA

	Consequence		
Frequency	High	Medium	Low
High	S	S	NS
Medium	S	S	NS
Low	NS	NS	NS

S = Potential Significant Risk Contributor
NS = Risk Non-Significant

g. List any comments or caveats that apply to the above assessment. The output from the above evaluation was a list of LCOs proposed for relocation that could have potential plant safety risk significance if not properly controlled by other procedures or programs. As a result these Specifications will be relocated to other plant controlled documents outside the Technical Specifications.

TABLE 3-1

BWR PRAs USED IN NEDO 31466 (AND SUPPLEMENT 1)

RISK ASSESSMENT

- BWR/6 Standard Plant, GESSAR II, 238 Nuclear Island, BWR/6 Standard Plant Probabilistic Risk Assessment, Docket No. STN 50-447, March 1982.

- La Salle County Station, NEDO-31085, Probabilistic Safety Analysis, February 1988.

- Grand Gulf Nuclear Station, IDCOR, Technical Report 86.2GG, Verification of IPE for Grand Gulf, March 1987.

- Limerick, Docket Nos. 50-352, 50-353, 1981, "Probabilistic Risk Assessment, Limerick Generating Station", Philadelphia Electric Company.

- Shoreham, Probabilistic Risk Assessment Shoreham Nuclear Power Station, Long Island Lighting Company, SAI-372-83-PA-01, June 24, 1983.

- Peach Bottom 2, NUREG-75/0104, "Reactor Safety Study", WASH-1400, October 1975.

- Millstone Point 1, NUREG/CR-3085, "Interim Reliability Evaluation Program: Analysis of the Millstone Point Unit 1 Nuclear Power Plant", January 1983.

- Grand Gulf, NUREG/CR-1659, "Reactor Safety Study Methodology Applications Program: Grand Gulf #1 BWR Power Plant", October 1981.

- NEDC-30936P, "BWR Owners' Group Technical Specification Improvement Methodology (with Demonstration for BWR ECCS Activation Instrumentation) Part 2", June 1987.

4. RESULTS OF APPLICATION OF SELECTION CRITERIA

The selection criteria from Section 2 were applied to the Plant Hatch Unit 1 Technical Specifications. The attachment is a summary of that application indicating which Specifications are being retained or relocated. Discussions that document the rationale for the relocation of each Specification which failed to meet the selection criteria are provided in Appendix A. No Significant Hazards Determination (10 CFR 50.92) evaluations for those Specifications relocated are provided with the Discussion of Changes for the specific Technical Specifications. GPC will relocate those Specifications identified as not satisfying the criteria to plant specific controlled documents whose changes are governed by 10 CFR 50.59.

5. REFERENCES

1. NEDO-31466 (and Supplement 1), "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.

2. NUREG 1433, "Standard Technical Specifications, General Electric Plants BWR/4," September 1992.

3. NRC No. 93-102 "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

ATTACHMENT

SUMMARY DISPOSITION MATRIX
PLANT HATCH UNIT 1

SUMMARY DISPOSITION MATRIX
PLANT HATCH UNIT 1

Current Unit 1 Number	Title	STS Rev. 4 Number	New Unit 1 TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[a][c]
1.0	DEFINITIONS	1.0 3.10.1	1.1	Yes	See Note 1 and Note 4.
None	Reactor Mode Switch Interlock Testing	Table 1.2	3.10.2	Yes	See Note 4.
1.1/1.2	SAFETY LIMITS	2.0	2.0		
1.1	Fuel Cladding Integrity				
1.1.A	Reactor Pressure > 800 psia and Core Flow > 10% of Rated	2.1.2	2.1.1.1	Yes	See Note 2.
1.1.B	Core Thermal Power Limit, (Reactor Pressure \ 800 psia)	2.1.1	2.1.1.2	Yes	See Note 2.
1.1.C	Power Transient	None	Deleted	No	Deleted. See Safety Limit technical change discussion.
1.1.D	Reactor Water Level (Hot or Cold Shutdown Condition)	2.1.4	2.1.1.3	Yes	See Note 2.
1.2	Reactor Coolant System Integrity				
1.2.A.1	Reactor Vessel Steam Dome Pressure - Irradiated Fuel in Reactor	2.1.3	2.1.2	Yes	See Note 2.
1.2.A.2	Reactor Vessel Steam Dome Pressure - Operating RHR SDC Mode	None	Deleted	No	Deleted. See Safety Limit technical change discussion.
2.1/2.2	LIMITING SAFETY SYSTEM SETTINGS				
2.1	Fuel Cladding Integrity	2.2.1 3.3.3	3.3.1.1 3.3.5.1	Yes	The application of Technical Specification selection criteria is not appropriate. However, the fuel cladding integrity LSSS have been included as part of the RPS and ECCS instrumentation Specifications, which has been retained since the Functions either actuate to mitigate consequences of Design Basis Accidents (DBAs) and transients or are retained as directed by the NRC as the Functions are part of the RPS.
2.2	Reactor Coolant System Integrity	2.2.1 3.4.2.1	3.3.1.1 3.4.3	Yes	The application of Technical Specification selection criteria is not appropriate. However, the Reactor Coolant System integrity LSSS has been included as part of RPS and safety relief valve Specifications, which have been retained since the instrument Functions and the safety relief valves mitigate the consequences of DBAs and transients which would result in overpressurization of the RCS.

Current Unit 1 Number	Title	STS Rev. 4 Number	New Unit 1 TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[a][c]
None	LIMITING CONDITIONS FOR OPERATION APPLICABILITY	3.0	LCO 3.0.1 thru LCO 3.0.7	Yes	See Note 3.
None	SURVEILLANCE REQUIREMENTS APPLICABILITY	4.0	SR 3.0.1 thru SR 3.0.4	Yes	See Note 3.
3/4.1					
(b)	REACTOR PROTECTION SYSTEM	3/4 3.1	3.3.1.1		
3/4.1.A	Sources of a Trip Signal which Initiate a Reactor Scram	3/4.3.1	3.3.1.1	Yes-2,3	Retained as directed by the NRC as it is part of the RPS, or it actuates to mitigate consequences of a DBA and/or transients, or it is an initial assumption in a Transient analysis, or it provides an anticipatory scram to ensure the scram discharge volume and thus RPS remains operable.
3/4.2[b]	PROTECTIVE INSTRUMENTATION	3/4.3	3.3		
3/4.2.A	Initiates Reactor Vessel and Containment Isolation	3/4.3.2 3.3.6.2	3.3.6.1	Yes-3,4	Actuates to mitigate the consequences of a DBA LOCA or is retained due to risk significance.
3/4.2.A.4	Main Steam Line Radiation	3/4.3.2.3.b	Deleted	No	Deleted. See Primary Containment Isolation Instrumentation technical change discussion for MSLRM.
3/4.2.A.8	RWCU Differential Flow	3/4.3.2.4.a	Deleted	No	Deleted. See Primary Containment Isolation Instrumentation technical change discussion.
3/4.2.B	Initiates or Controls HPCI	3/4.3.2.6	3.3.5.1.3	Yes-4	Actuation instrumentation actuates to mitigate consequences of a LOCA.
3/4.2.B.3	HPCI Turbine Overspeed - Mechanical	None	Relocated	No	See Appendix A, Page 1.
3/4.2.B.4	HPCI Turbine Exhaust Pressure - High	None	Relocated	No	See Appendix A, Page 1.
3/4.2.B.5	HPCI Pump Suction Pressure - Low	None	Relocated	No	See Appendix A, Page 1.
3/4.2.B.16	HPCI Logic Power Failure Monitor	None	Deleted	No	Deleted. See ECCS Instrumentation technical change discussion.
3/4.2.C	Initiates or Controls RCIC	3/4.3.2.5	3.3.5.2	Yes-4	Retained in accordance with the NRC Final Policy Statement on Technical Specification improvements due to risk significance.
3/4.2.C.2	RCIC Turbine Overspeed - Electrical and Mechanical	None	Relocated	No	See Appendix A, Page 2.

SUMMARY DISPOSITION MATRIX
PLANT HATCH UNIT 1

Current Unit 1 Number	Title	STS Rev. 4 Number	New Unit 1 TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[(a)(c)]
3/4.2.C.3	RCIC Turbine Exhaust Pressure - High	None	Relocated	No	See Appendix A, Page 2.
3/4.2.C.4	RCIC Pump Suction Pressure - Low	None	Relocated	No	See Appendix A, Page 2.
3/4.2.C.6	RCIC Pump Discharge Flow	None	Relocated	No	See Appendix A, Page 2.
3/4.2.C.13	RCIC Logic Power Failure Monitor	None	Deleted	No	Deleted. See RCIC Instrumentation technical change discussion.
3/4.2.D	Initiates or Controls ADS	3/4.3.3.4	3.3.5.1.4 3.3.5.1.5	Yes-3	Functions to mitigate the consequences of small break LOCAs.
3/4.2.D.7	Automatic Blowdown Control Power Failure Monitor	None	Deleted	No	Deleted. See ECCS Instrumentation technical change discussion.
3/4.2.E	Initiates or Controls the LPCI Mode of RHR	3/4.3.3.2	3.3.5.1.2	Yes-3	ECCS mitigate the consequences of a DBA LOCA.
3/4.2.E.5	LPCI Cross Connect Valve Open Annunciator	None	Relocated	No	See Appendix A, Page 4.
3/4.2.E.8	Valve Selection Times	None	Relocated	No	See Appendix A, Page 5.
3/4.2.E.9	RHR Relay Logic Power Failure Monitor	None	Deleted	No	Deleted. See ECCS Instrumentation technical change discussion.
3/4.2.F	Initiates or Controls Core Spray	3/4.3.3.1	3.3.5.1.1	Yes-3	ECCS mitigate the consequences of a DBA LOCA.
3/4.2.F.4	Core Spray Sparger Differential Pressure	3/4.5.1	Deleted	No	Deleted. See ECCS Instrumentation technical change discussion.
3/4.2.F.6	Core Spray Logic Power Failure Monitor	None	Deleted	No	Same as above.
3/4.2.G	Initiates Control Rod Blocks	3/4.3.6	3.3.2.1		
3/4.2.G.1	SRM	3/4.3.6.3	Relocated	No	See Appendix A, Page 6.
3/4.2.G.2	IRM	3/4.3.6.4	Relocated	No	See Appendix A, Page 7.
3/4.2.G.3	APRM	3/4.3.6.2	Relocated	No	See Appendix A, Page 8.
3/4.2.G.4	RBM	3/4.3.6.1	3.3.2.1.1	Yes-3	Prevents continuous withdrawal of a high worth control rod tnat could challenge the MCPR Safety Limit.
3/4.2.G.5	Scram Discharge Volume	3/4.3.6.5	Relocated	No	See Appendix A, Page 9.
3/4.2.H	Limit Radioactive Release	3/4.3.2 3/4.3.7.1	3.3.6.2 3.3.7.1		
3/4.2.H.1	Off-Gas Post Treatment Radiation Monitors	3/4.3.7.1.4	Relocated	No	See Appendix A, Page 10.
3/4.2.H.2	Refueling Floor Exhaust Vent Radiation Monitors	3/4.3.2.2.c	3.3.6.1.2.e 3.3.6.2.4	Yes-3	Actuates to mitigate consequences of a DBA LOCA and Fuel handling accident.
3/4.2.H.3	Reactor Building Exhaust Vent Radiation Monitors	None	3.3.6.1.2.d 3.3.6.2.3	Yes-3	Same as above.

SUMMARY DISPOSITION MATRIX
PLANT HATCH UNIT 1

Current Unit 1 Number	Title	STS Rev. 4 Number	New Unit 1 TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[(a)(c)]
3/4.2.H.4	Control Room Intake Radiation Monitors	3/4.3.7.1.5	3.3.7.1	Yes-3	Actuates to maintain control room habitability so that operation can continue from the control room following a DBA.
3/4.2.H.5	Main Steam Line Radiation Monitor	None	Deleted	No	Deleted. See radiation monitoring technical change discussion for MSLRM.
3/4.2.I	Initiates Recirculation Pump Trip	3/4.3.4.1 3/4.3.4.2	3.3.4.1 3.3.4.2		
3/4.2.I.1/2	ATWS-RPT	3/4.3.4.1	3.3.4.2	Yes-4	ATWS-RPT is being retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements due to risk significance.
3/4.2.I.3	EOC-RPT	3/4.3.4.2	3.3.4.1	Yes-3	EOC-RPT aids the reactor scram in protecting fuel cladding integrity by ensuring the fuel cladding integrity Safety Limit is not exceeded during a load rejection or turbine trip transient.
3/4.2.J	Monitors Leakage into the Drywell	3/4.4.3	3.4.5	Yes-1	Leak detection is used to indicate a significant abnormal condition of the reactor coolant pressure boundary.
3/4.2.K	Provides Surveillance Information	3/4.3.7.5	3.3.3.1	Yes-3	RG 1.97 Type A and Category 1 variables retained. See Appendix A, Page 11 for full discussion of all variables.
3/4.2.L	Degraded Station Voltage Protection Instrumentation	3/4.3.3.5			
3/4.2.L.1	4.16kv Emergency Bus Undervoltage Relay (Loss of Voltage Condition)	3/4.3.3.5.1	3.3.8.1	Yes-3	Actuates DGs to mitigate consequences of a loss of offsite power event.
3/4.2.L.2	4.16kv Emergency Bus Undervoltage Relay (Degraded Voltage Condition)	3/4.3.3.5.2	Deleted	No	Deleted. See LOP Instrumentation technical change discussion.
3/4.2.M	Deleted in Amendment No. 186				
3/4.2.N	Arms Low Low Set S/RV System	3/4.4.2.1 3/4.4.2.2	3.3.6.3	Yes-3	Actuates LLS S/RVs, which are assumed to function in the containment loading safety analysis.
None	Remote Shutdown System	3/4.3.7.4	3.3.3.2	Yes-4	Being added as directed by the NRC as it is a significant contributor to risk reduction.
None	Feedwater and Main Turbine Trip Instrumentation	3/4.3.9	3.3.2.2	Yes-3	Acts to limit feedwater addition to the reactor vessel on feedwater controller failure consistent with safety analysis assumptions. Limits neutron flux peak and thermal transient to avoid fuel damage.

SUMMARY DISPOSITION MATRIX
PLANT HATCH UNIT 1

Current Unit 1 Number	Title	STS Rev. 4 Number	New Unit 1 TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[a][c]
3/4.3	REACTIVITY CONTROL	3/4.1	3.1		
3/4.3.A	Core Reactivity Margin	3/4.1.1	3.1.1	Yes-2	Not a measured process variable, but is important parameter that is used to confirm the acceptability of the accident analysis. In addition, the LCO is retained as directed by the NRC.
3/4.3.B	Inoperable Control Rods				
3/4.3.B.1	No Movement by Control Rod Drive Pressure	3/4.1.3.1	3.1.3	Yes-3	Control rods are part of the primary success path in mitigating the consequences of DBAs and transients.
3/4.3.B.2	Excessive Scram Time	3/4.1.3.2	3.1.3	Yes-3	Same as above.
3/4.3.B.3	Inoperable Accumulators	3/4.1.3.5 3/4.1.3.7	3.1.3 3.1.5	Yes-3	Same as above.
3/4.3.B.4	Limiting Number of Inoperable Control Rods	3/4.1.3.1	3.1.3	Yes-3	Same as above.
3/4.3.C	Control Rod Drive System				
3/4.3.C.1	Control Rod Drive Coupling Integrity	3/4.1.3.6	3.1.3	Yes-3	Same as above.
3/4.3.C.2	Scram Insertion Times	3/4.1.3.3 3/4.1.3.4	3.1.4	Yes-3	Same as above.
3/4.3.C.3	Control Rod Drive Housing Support System	3/4.1.3.8	Deleted	No	See CRD Housing Support System technical change discussion.
3/4.3.D	Minimum Count Rate for Rod Withdrawal	3/4.3.7.6	3.3.1.2	Yes	Does not satisfy the selection criteria, however is being retained because the NRC considers it necessary for flux monitoring during shutdown, startup and refueling operations.
3/4.3.E	Rod Worth Inventory Determination	3/4.1.2	3.1.2	Yes-2	Confirms assumptions made in the reload safety analysis.
3/4.3.F	Operation with a Limiting Control Rod Pattern	3/4.3.6.1	3.3.2.1.1	Yes-3	Prevents continuous withdrawal of a high worth control rod that could challenge the MCPR Safety Limit.
3/4.3.G	Rod Worth Minimizer				
3/4.3.G.1	Operability	3/4.1.4.1	3.3.2.1.2	Yes-3	Prevents withdrawal of control rods outside BPWS constraints that might set up high rod worth conditions beyond CRDA assumptions.
3/4.3.G.2	Special Test Exceptions	3/4.10.2	3.10.7	Yes	See Note 4.
3/4.3.H	Shutdown Requirements	3/4.1 (all)	3.1 (all)	Yes-3	The LCOs this Specification is associated with provide the reactivity control requirements that mitigate the consequences of, or prevent a DBA or transient. Therefore, this Specification has been incorporated into Actions for the associated LCOs.

Current Unit 1 Number	Title	STS Rev. 4 Number	New Unit 1 TS Number	Criterion for Inclusion	Bases for Inclusion/Exclusion[(a)(c)]
3/4.3.I	Scram Discharge Volume Vent and Drain Valves	3/4.1.3.1	3.1.8	Yes-3	Contributes to the operability of the control rod scram function.
None	Rod Pattern Control	None	3.1.6	Yes-3	Assures initial conditions for the CRDA analysis are maintained.
None	SHUTDOWN MARGIN (SDM) Test - Refueling	3/4.10.3	3.10.8	Yes	See Note 4.
3/4.4	STANDBY LIQUID CONTROL SYSTEM	3/4.1.5	3.1.7	Yes-4	Being retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements due to risk significance.
3/4.5	CORE AND CONTAINMENT COOLING SYSTEMS	3/4.5 3/4.7 3/4.8	3.5 3.7 3.8		
3/4.5.A	Core Spray (CS) System	3/4.5.1	3.5.1	Yes-3	Functions to mitigate the consequences of a DBA LOCA.
3/4.5.B	Residual Heat Removal (RHR) System (LPCI and Containment Cooling Mode)	3/4.4.9.2 3/4.5.1 3/4.6.2.3 3/4.9.11.1 3/4.9.11.2	3.4.8 3.5.1 3.6.2.3 3.9.7 3.9.8	Yes-3,4	LPCI and containment cooling mode functions to mitigate the consequences of a DBA LOCA. The RHR Shutdown cooling mode is being retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements due to risk significance.
3/4.5.C	RHR Service Water System	3/4.7.1.1	3.7.1	Yes-3	Designed for heat removal from RHR heat exchangers following a DBA. As such, acts to mitigate the consequences of an accident.
3/4.5.D	High Pressure Coolant Injection (HPCI) System	3/4.5.1	3.5.1	Yes-4	While not assumed in a licensing basis accident analysis, HPCI is considered risk significant since it functions to mitigate the consequences of small break LOCAs.
3/4.5.E	Reactor Core Isolation Cooling (RCIC) System	3/4.7.4	3.5.3	Yes-4	Being retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements due to risk significance.
3/4.5.F	Automatic Depressurization System (ADS)	3/4.5.1	3.5.1	Yes-3	Functions to mitigate the consequences of small break LOCAs.
3/4.5.G	Minimum Core and Containment Cooling Systems Availability	3/4.5.2 3/4.8.1.1	3.5.2 3.8.1	Yes-3	Ensures inoperability of a diesel generator does not result in loss of containment cooling functions. Since ECCS is assumed to function to mitigate the consequences of a DBA LOCA, this requirement has been maintained and has been moved to the Actions Table of LCO 3.8.1. The low pressure ECCS requirement ensures systems are available to mitigate the consequences of a vessel draindown event.

Current Unit 1 Number	Title	STS Rev. 4 Number	New Unit 1 TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[a][c]
3/4.5.H	Maintenance of Filled Discharge Pipes	3/4.5.1	3.5.1 3.5.2 3.5.3	Yes-3	This Specification ensures the OPERABILITY of the ECCS and RCIC, which function to mitigate the consequences of a DBA LOCA (ECCS) or is required to be retained by the NRC Interim Policy Statement on Technical Specification Improvements (RCIC).
3/4.5.I	Minimum River Level	3/4.7.1.3	3.7.2	Yes-3	Heat sink for heat removal from safety related systems following a DBA. As such, acts to mitigate the consequences of an accident.
3/4.5.J	Plant Service Water System	3/4.7.1.2	3.7.2 3.7.3	Yes-3	Designed for heat removal from various safety related systems following a DBA. As such, acts to mitigate the consequences of an accident.
3/4.5.K	Equipment Area Coolers	None	None	No	Relocated to the Bases as they are part of ECCS and RCIC Operability. See ECCS and RCIC technical change discussion.
3/4.6	PRIMARY SYSTEM BOUNDARY	3/4.4	3.4		
3/4.6.A	Reactor Coolant Heat-Up and Cooldown	3/4.4.6.1	3.4.9	Yes-2	Establishes initial conditions to operation such that operation is prohibited in areas or at temperature rate changes that might cause undetected flaws to propagate in turn challenging the reactor coolant pressure boundary integrity.
3/4.6.B	Reactor Vessel Temperature and Pressure	3/4.4.6.1	3.4.9	Yes-2	Same as above.
3/4.6.C	Reactor Vessel Head Stud Tensioning	3/4.4.6.1	3.4.9	Yes-2	Same as above.
3/4.6.D	Idle Recirculation Loop Startup	3/4.4.1.4	3.4.9	Yes-2	Same as above.
3/4.6.E	Recirculation Pump Start	3/4.4.1.4	3.4.9	Yes-2	Same as above.
3/4.6.F	Reactor Coolant Chemistry				
3/4.6.F.1	Radioactivity	3/4.4.5	3.4.6	Yes-2	Specific activity provides an indication of the onset of significant fuel cladding failure and is an initial condition for evaluation (radiological calculations) of the consequences of an accident due to main steam line break outside containment.
3/4.6.F.2	Conductivity and Chloride	3/4.4.4	Relocated	No	See Appendix A, Page 13.

Current Unit 1 Number	Title	STS Rev. 4 Number	New Unit 1 TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion(a)(c)
3/4.6.G	Reactor Coolant Leakage				
3/4.6.G.1	Unidentified and Total	3/4.4.3.2	3.4.4	Yes-1	Leakage beyond limits would indicate an abnormal condition of the reactor coolant pressure boundary. Operation in this condition may result in reactor coolant pressure boundary failure.
3/4.6.G.2	Leakage Detection Systems	3/4.4.3.1	3.4.5	Yes-1	Leak detection is used to indicate an abnormal condition of the reactor coolant pressure boundary.
3/4.6.G.3	Shutdown Requirements	3/4.4.3.1 3/4.4.3.2	3.4.4 3.4.5	Yes	The LCOs this Specification is associated with provide the leakage requirements which meet criterion 1 (as listed in the two previous entries). Therefore, this Specification has been incorporated as Actions for the associated LCOs.
3/4.6.H.1	Relief/Safety Valves	3/4.4.2.1	3.3.6.3 3.4.3	Yes-3	A minimum number of S/RVs is assumed in the safety analysis to mitigate overpressure events.
3/4.6.H.2	Relief/Safety Valves Low Low Set Function	3/4.4.2.2	3.3.6.3 3.6.1.6	Yes-3	A minimum number of S/RVs is assumed in the containment loading safety analysis.
3/4.6.I	Jet Pumps	3/4.4.1.2	3.4.2	Yes-3	Jet pump operability is assumed in the LOCA analysis to assure adequate core reflood capability.
3/4.6.J	Recirculation System	3/4.4.1.1	3.4.1	Yes-2	Recirculation loop flow is an initial condition in the safety analysis.
None	RHR - Hot Shutdown	3/4.4.9.1	3.4.7	Yes-4	Being retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements due to risk significance.
None	Reactor Steam Dome Pressure	3/4.4.6.2	3.4.10	Yes-3	Reactor steam dome pressure is an initial condition in the reactor vessel overpressure safety analysis.
3/4.6.K	Structural Integrity	3/4.4.8	Relocated	No	See Appendix A, page 14.
3/4.6.L	Snubbers	3/4.7.5	Deleted	No	Deleted. See technical change discussion for Snubbers.
3/4.7	CONTAINMENT SYSTEMS	3/4.6	3.6		
3/4.7.A	Primary Containment				
3/4.7.A.1	Pressure Suppression Chamber	3/4.6.2.1	3.6.2.1 3.6.2.2	Yes-2 & 3	Suppression pool water level and temperature are initial conditions in the DBA LOCA analysis and mitigate the consequences of the DBA.
3/4.7.A.2	Primary Containment Integrity	3/4.6.1.1 3/4.6.1.2 3/4.6.1.3 3/4.6.1.5	3.6.1.1 3.6.1.2	Yes-3	Primary containment integrity functions to mitigate the consequences of a DBA. Primary Containment leakage is an assumption utilized in the LOCA safety analysis to ensure Primary Containment Operability.

SUMMARY DISPOSITION MATRIX
PLANT HATCH UNIT 1

Current Unit 1 Number	Title	STS Rev. 4 Number	New Unit 1 TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[(a)(c)]
3/4.7.A.3	Reactor Building to Pressure Suppression Chamber - Vacuum Relief System	3/4.6.4.2	3.6.1.7	Yes-3	Reactor building to pressure suppression chamber vacuum breaker operation is relied on to limit negative pressure differential, secondary to primary containment, that could challenge primary containment integrity.
3/4.7.A.4	Pressure Suppression Chamber to Drywell Vacuum Breakers	3/4.6.4.1	3.6.1.8	Yes-3	Pressure suppression chamber to drywell vacuum breaker operation is assumed in the LOCA analysis to limit drywell pressure thereby ensuring primary containment integrity.
3/4.7.A.5	Oxygen Concentration	3/4.6.6.4	3.6.3.2	Yes-2	Oxygen concentration is limited such that when combined with hydrogen that is postulated to evolve following a LOCA the total explosive gas concentration remains below explosive levels. Therefore, containment integrity is maintained.
3/4.7.A.6	Containment Atmosphere Dilution (CAD)	3/4.6.6.2	3.6.3.1	Yes-3	System ensures oxygen concentration is maintained below the explosive level following a LOCA by inerting the drywell with nitrogen. Therefore, containment integrity is maintained.
3/4.7.A.7	Primary Containment Purge System	3/4.6.1.8	3.6.1.3	Yes-3	Isolation valves function to limit DBA consequences.
3/4.7.A.8	Shutdown Requirements	3/4.6.1 3/4.6.2 3/4.6.4	3.6.1 3.6.2 3.6.3	Yes-3	The LCOs that this Specification are associated with, provide the primary containment and combustible gas control requirements that mitigate the consequences of a DBA. Therefore, this Specification has been incorporated into Actions for the associated LCOs.
None	RHR Suppression Pool Spray	3/4.6.2.2	3.6.2.4	Yes-3	Suppression pool spray functions to limit the effects of a DBA.
None	Primary Containment Pressure	3/4.6.1.6	3.6.1.4	Yes-2	Primary containment pressure is an initial condition in the LOCA safety analysis.
None	Primary Containment Air Temperature	3/4.6.1.7	3.6.1.5	Yes-2	Primary containment air temperature is an initial condition in the LOCA safety analysis.
3/4.7.B	Standby Gas Treatment System	3/4.6.5.3	3.6.4.3	Yes-3	SGT operation following a DBA acts to mitigate the consequences of offsite releases.
3/4.7.C	Secondary Containment	3/4.6.5.1 3/4.6.5.5	3.6.4.1 3.6.4.2	Yes-3	Secondary containment integrity is relied on to limit the offsite dose during an accident by ensuring a release to containment is delayed and treated prior to release to the environment. Damper operation within time limits establishes secondary containment and limits offsite releases to acceptable values.
3/4.7.D	Primary Containment Isolation Valves	3/4.6.3	3.6.1.3	Yes-3	Isolation Valves function to limit DBA consequences.

Current Unit 1 Number	Title	STS Rev. 4 Number	New Unit 1 TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[(a)(c)]
3/4.8	RADIOACTIVE MATERIALS	3/4.7.6			
3/4.8.A	Miscellaneous Radioactive Materials Sources	3/4.7.6	Relocated	No	See Appendix A, page 16.
3/4.9	AUXILIARY ELECTRICAL SYSTEMS	3/4.8	3.8		
3/4.9.A	Requirements for Reactor Startup				
3/4.9.A.1	Offsite Power Sources	3/4.8.1.1	3.8.1	Yes-3	Required to mitigate the consequences of a DBA.
3/4.9.A.2	Standby AC Power Supply (Diesel Generators 1A, 1B, and 1C)	3/4.8.1.1	3.8.1 3.8.3 3.8.4 3.8.6	Yes-3	Same as above.
3/4.9.A.3	125/250 Volt DC Emergency Power System (Plant Batteries 1A and 1B)	3/4.8.2.1	3.8.4 3.8.6	Yes-3	Same as above.
3/4.9.A.4	Emergency 4160 Volt Buses (1E, 1F, and 1G)	3/4.8.3.1	3.8.7	Yes-3	Same as above.
3/4.9.A.5	Emergency 600 Volt Buses (1C and 1D)	3/4.8.3.1	3.8.7	Yes-3	Same as above.
3/4.9.A.6	Emergency 250 Volt DC to 600 Volt AC Inverters	3/4.8.3.1	3.5.1	Yes-3	Same as above.
3/4.9.A.7	Logic Systems	3/4.8.1.1	3.8.1	Yes-3	Same as above.
3/4.9.B	Requirements for Continued Operation with Inoperable Components	3/4.8 (all)	3.8 (all)	Yes-3	The LCOs that this Specification are associated with, provide the electrical systems requirements that mitigate the consequences of a DBA. Therefore, this Specification has been incorporated into Actions for the associated LCOs.
3/4.9.C	Diesel Generator Requirements (Reactor in the Shutdown or Refuel Mode)	3/4.8.1.2	3.8.2 3.8.3 3.8.5 3.8.6	Yes-3	Functions to mitigate the consequences of a vessel drain-down event and is needed to support NRC Final Policy Statement requirement for decay heat removal.
3/4.9.D	Electric Power Monitoring for the Reactor Protection System	3/4.8.4.4	3.3.8.2	Yes-3	Provides protection for the RPS bus powered instrumentation against unacceptable voltage and frequency conditions that could degrade the instrumentation so that it would not perform the intended safety function.
None	AC Sources - Shutdown	3/4.8.1.2	3.8.2	Yes-3	Functions to mitigate the consequences of a vessel drain-down event and is needed to support NRC Final Policy Statement requirements for decay heat removal.
None	DC Sources - Shutdown	3/4.8.2.2	3.8.5	Yes-3	Same as above.
None	Distribution Systems - Shutdown	3/4.8.3.2	3.8.8	Yes-3	Same as above.

Current Unit 1 Number	Title	STS Rev. 4 Number	New Unit 1 TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[(a)(c)]
3/4.10	REFUELING	3/4.9	3.9		
3/4.10.A	Refueling Interlocks	3/4.9.1	3.9.1 3.9.2	Yes-3	Provides an interlock to preclude fuel loading with control rods withdrawn. Operation is assumed in the control rod removal error during refueling and fuel assembly insertion error during refueling accident analysis.
3/4.10.B	Fuel Loading	3/4.9.3	3.9.3	Yes-2	All control rods are required to be fully inserted when loading fuel. This requirement is assumed as an initial condition in the fuel assembly insertion error during refueling accident analysis.
3/4.10.C	Core Monitoring During Core Alterations	3/4.9.2	3.3.1.2	Yes	Does not satisfy criteria for inclusion but is retained because the NRC considers it necessary for flux monitoring during shutdown, startup, and refueling operations.
3/4.10.D	Spent Fuel Pool Water Level	3/4.9.9	3.7.8	Yes-2	A minimum amount of water is required to assure adequate scrubbing of fission products following a fuel handling accident.
3/4.10.E	Control Rod Drive Maintenance	3/4.9.10.1 3/4.9.10.2	3.10.4 3.10.5 3.10.6	Yes	See Note 4.
None	Single Control Rod Removal - Hot Shutdown	None	3.10.3	Yes	See Note 4.
3/4.10.F	Reactor Building Cranes	3/4.9.6	Relocated	No	See Appendix A, Page 17.
3/4.10.G	Spent Fuel Cask Lifting Trunnions and Yoke	3/4.9.6	Relocated	No	See Appendix A, Page 17.
3/4.10.H	Time Limitation	3/4.9.4	Relocated	No	Although this LCO satisfied criterion 2, the activities necessary prior to commencing movement of irradiated fuel ensure that there will always be 24 hours of subcriticality before movement of any irradiated fuel. Hence this Specification has been relocated.
3/4.10.I	Crane Travel - Spent Fuel Storage Pool	3/4.9.7	Relocated	No	See Appendix A, Page 19.
None	Control Rod Position Indication	3/4.1.3.7	3.9.4	Yes-3	Control Rods are part of the primary success path in mitigating the consequences of DBAs.
None	Control Rod OPERABILITY-Refueling	3/4.1.3.5	3.9.5	Yes-3	Same as above.
None	Reactor Pressure Vessel Water Level	3/4.9.8	3.9.6	Yes-2	A minimum amount of water is required to assure adequate scrubbing of fission products following a fuel handling accident.

Current Unit 1 Number	Title	STS Rev. 4 Number	New Unit 1 TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[(a)(c)]
3/4.11	FUEL RODS	3/4.2	3.2		
3/4.11.A	Average Planer Linear Heat Generation Rate (APLHGR)	3/4.2.1	3.2.1	Yes-2	Peak cladding temperature following a LOCA is primarily dependent on initial APLHGR. As such, it is an initial condition of a DBA analysis.
3/4.11.B	Linear Heat Generation Rate (LHGR)	3/4.2.4	Deleted	No	Deleted. See LHGR technical change discussion.
3/4.11.C	Minimum Critical Power Ratio (MCPR)	3/4.2.3	3.2.2	Yes-2	Utilized as an initial condition of the design basis transients. Transient analysis are performed to establish the largest reduction in Critical Power Ratio. This value is added to the fuel cladding integrity safety limit to determine the MCPR value.
3/4.12	Main Control Room Environmental Systems	3/4.3.7.1 3/4.7.2	3.3.7.1 3.7.4	Yes-3	Maintains habitability of the control room so that operators can remain in the control room following an accident. As such, it mitigates the consequences of an accident by allowing operators to continue accident mitigation activities from the control room.
None	Main Control Room Air Conditioning System	None	3.7.5	Yes-3	Ensures control room temperature is maintained such that control room safety related equipment remains OPERABLE following an accident. As such, functions to mitigate the consequences of an accident.
3/4.13	Removed in Amendment No. 133				
3/4.14	RADIOACTIVE EFFLUENT INSTRUMENTATION	None	None		
3/4.14.1	Removed in Amendment 190				
3/4.14.2	Explosive Gas Monitoring Instrumentation	None	Relocated	No	See Appendix A, Page 20.
3/4.15	RADIOACTIVE EFFLUENTS	None	None		
3/4.15.1	Liquid Effluents				
3/4.15.1.1- 3/4.15.1.3	Removed in Amendment No. 190				

Current Unit 1 Number	Title	STS Rev. 4 Number	New Unit 1 TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[(a)(c)]
3/4.15.1.4	Liquid Holdup Tanks	None	5.5.8	Yes	Although this specification does not meet any criteria of the NRC Final Policy statement, it has been retained in accordance with NRC letter from W. T. Russell to the Industry ITS chairpersons, dated October 25, 1993.
3/4.15.2	Gaseous Effluents				
3/4.15.2.1-3/4.15.2.5	Removed in Amendment No. 190				
3/4.15.2.6	Explosive Gas Mixture	None	5.5.8	Yes	Same as above.
3/4.15.2.7	Main Condenser	None	3.7.6	Yes-2	Main condenser offgas activity is an initial condition in the offgas system failure event.
3/4.15.3	Removed in Amendment No. 190				
3/4.16	Removed in Amendment No. 190				
5.0	MAJOR DESIGN FEATURES	5.0	4.0	Yes	See Note 5.
6.0	ADMINISTRATIVE CONTROLS	6.0	5.0	Yes	See Note 6.

NOTE 1: DEFINITIONS

This section provides definitions for several defined terms used throughout the remainder of Technical Specifications. They are provided to improve the meaning of certain terms. As such, direct application of the Technical Specification selection criteria is not appropriate. However, only those definitions for defined terms that remain as a result of application of the selection criteria, will remain as definitions in this section of Technical Specifications.

NOTE 2: SAFETY LIMITS/LSSS

Application of Technical Specification selection criteria is not appropriate. However, Safety Limits and Limiting Safety System Settings (as part of Reactor Protection System Instrumentation) will be included in Technical Specifications as required by 10 CFR 50.36.

NOTE 3: 3.0/4.0

These Specifications provide generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operation and Surveillance Requirements. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the general requirements of 3.0/4.0 will be added to the Plant Hatch Unit 1 Technical Specifications consistent with NUREG 1433.

NOTE 4: SPECIAL TEST EXCEPTIONS

These Specifications are provided to allow relaxation of certain Limiting Conditions for Operation under certain specific conditions to allow testing and maintenance. They are directly related to one or more Limiting Conditions for Operations. Direct application of the Technical Specification selection criteria is not appropriate. However, those special test exceptions, directly tied to Limiting Conditions for Operation that remain in Technical Specifications, will also remain as Technical Specifications. Those special test exceptions not applicable to Plant Hatch Unit 1 have been deleted.

NOTE 5: DESIGN FEATURES

Application of Technical Specification selection criteria is not appropriate. However, Design Features will be included in Technical Specifications as required by 10 CFR 50.36.

NOTE 6: ADMINISTRATIVE CONTROLS

Application of Technical Specification selection criteria is not appropriate. However, Administrative Controls will be included in Technical Specifications as required by 10 CFR 50.36.

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a. Where a current Technical Specification is referred to as being deleted, the technical change discussion is found the Discussion of Changes associated with the markup of the current Specification.

b. For current Technical Specifications 3/4.1, RPS, and 3/4.2, Protective Instrumentation, the current Technical Specification number consists of the Specification number and the instrumentation channel's number from the associated 3.1.x or 3.2.x Table. For example, the Main Steam Line Radiation channel for the RPS is numbered 3/4.1.A.9, where "3/4.1.A" is the Specification number and "9" is the scram number for the IRM channels in Table 3.1-1.

c. The applicable accident analyses are discussed in the Bases for the individual Technical Specification.

APPENDIX A

JUSTIFICATION FOR SPECIFICATION RELOCATION

APPENDIX A

3/4.2 PROTECTIVE INSTRUMENTATION

LCO Statement:

The Limiting Conditions for Operation of the protective instrumentation affecting each of the following protective actions shall be as indicated in the corresponding LCO table.

3/4.2.B Initiates or Controls HPCI

3/4.2.B.3 HPCI Turbine Overspeed - Mechanical
3/4.2.B.4 HPCI Turbine Exhaust Pressure - High
3/4.2.B.5 HPCI Pump Suction Pressure - Low

Discussion:

The function of these instruments is to provide a close signal to the HPCI turbine stop valve. In turn, the injection valve and minimum flow valve will close. All of which will prevent the system from operating. Signals from any of these three instruments will result in the valves listed above receiving a signal to close (directly or indirectly). These instruments actuate to provide turbine/pump protection only to preclude turbine/pump damage and possible breach of the system. The valves are not credited with providing primary containment isolation on these signals, nor are they credited with closing to isolate a primary coolant leak on these signals. No design basis analysis takes credit for these instruments.

Comparison to Deterministic Screening Criteria:

1. These instruments, in securing the HPCI System from operation, are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. These instruments do not sense a process variable that is an initial condition of a DBA or transient analysis.

3. These instruments are not part of a primary success path in the mitigation of a DBA or transient. They are not assumed to function during a DBA or transient. In addition, the valves closed by these instruments are not primary containment isolation valves.

As discussed in Appendix B (Page 1 of 8) of this document, the loss of these instruments was found to be a non-significant risk contributor to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, these HPCI Instrument LCOs and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.2 PROTECTIVE INSTRUMENTATION

LCO Statement:

The Limiting Conditions for Operation of the protective instrumentation affecting each of the following protective actions shall be as indicated in the corresponding LCO table.

3/4.2.C Initiates or Controls RCIC

3/4.2.C.2 RCIC Turbine Overspeed - Electrical and Mechanical
3/4.2.C.3 RCIC Turbine Exhaust Pressure - High
3/4.2.C.4 RCIC Pump Suction Pressure - Low
3/4.C.2.6 RCIC Pump Discharge Flow

Discussion:

The function of the first three above listed instruments is to provide a close signal to the RCIC turbine trip throttle valve. In turn, the injection valve and minimum flow valve will close. All of which will prevent the system from operating. Signals from any of these three instruments will result in the valves listed above receiving a signal to close (directly or indirectly). These instruments actuate to provide turbine/pump protection only to preclude turbine/pump damage and possible breach of the system. The valves are not credited with providing primary containment isolation on these signals nor are they credited with closing to isolate a primary coolant leak on these signals. No design basis analysis takes credit for these instruments.

The fourth instrument listed above controls the minimum flow valve. Failure of this valve could result in failure of the RCIC System. The instrument actuates to provide pump protection only to preclude pump damage. No design basis analysis takes credit for these instruments.

Comparison to Deterministic Screening Criteria:

1. These instruments, in securing the RCIC System from operation or operating the minimum flow valve, are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. These instruments do not sense a process variable that is an initial condition of a DBA or transient analysis.

3. These instruments are not part of a primary success path in the mitigation of a DBA or transient. They are not assumed to function during a DBA or transient. In addition, the valves closed by these instruments are not primary containment isolation valves.

As discussed in Appendix B (Page 4) of this document, the loss of these instruments was found to be a non-significant risk contributor to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, these RCIC Instrument LCOs and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.2 PROTECTIVE INSTRUMENTATION

LCO Statement:

The Limiting Conditions for Operation of the protective instrumentation affecting each of the following protective actions shall be as indicated in the corresponding LCO table.

3/4.2.E Initiates or Controls the LPCI Mode of RHR

3/4.2.E.5 LPCI Cross Connect Valve Open Annunciator

Discussion:

This instrument initiates annunciation in the control room when the LPCI cross-tie valve is not closed. During normal operation, the LPCI cross-tie valve is required, by current Specification 3.5.B.1.d, to be closed and the associated control circuit breaker (to the valve) locked in the off position. Thus, this instrument is not the primary method for ensuring the valve remains closed, nor does any accident analysis take credit for this instrument.

Comparison to Deterministic Screening Criteria:

1. The LPCI cross connect valve open annunciator is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The LPCI cross connect valve open annunciator is not a process variable that is an initial condition of a DBA or transient analysis.

3. The LPCI cross connect valve open annunciator is not part of the primary success path in the mitigation of a DBA or transient. It is not assumed to function during a DBA or transient.

As discussed in Section 6 and summarized in Table 6-1 (Item 332) of NEDO-31466, Supplement 1, the loss of the LPCI cross connect valve open annunciator was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 1.

Conclusion:

Since the screening criteria have not been satisfied, the LPCI Cross Connect Valve Open Annunciator LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.2 PROTECTIVE INSTRUMENTATION

LCO Statement:

The Limiting Conditions for Operation of the protective instrumentation affecting each of the following protective actions shall be as indicated in the corresponding LCO table.

3/4.2.E Initiates or Controls the LPCI Mode of RHR

3/4.2.E.8 Valve Selection Timers

Discussion:

The purpose of these instruments is to interlock open the RHR outboard isolation valves upon receipt of a LOCA signal, to ensure maximum LPCI flow to the reactor vessel. This ensures that a loss of LPCI flow will not occur due to an operator inadvertently closing these valves. However, while this feature may provide added assurance of LPCI flow under certain circumstances, it is not assumed in any design basis analysis. In addition, under certain conditions the operator must secure LPCI flow, and thus do so by other means that are not interlocked (e.g., secure the RHR pumps).

Comparison to Deterministic Screening Criteria:

1. The valve selection timers are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The valve selection timers are not a process variable that is an initial condition of a DBA or transient analysis.

3. The valve selection timers are not part of the primary success path in the mitigation of a DBA or transient. They are not assumed to function during a DBA or transient.

As discussed in Appendix B (Page 7) of this document, the loss of the valve selection timers was found to be a non-significant risk contributor to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the Valve Selection Timers LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.2 PROTECTIVE INSTRUMENTATION

LCO Statement:

The Limiting Conditions for Operation of the protective instrumentation affecting each of the following protective actions shall be as indicated in the corresponding LCO table.

3/4.2.G Initiates Control Rod Blocks

3/4.2.G.1 SRM

Discussion:

The Source Range Monitor (SRM) control rod block functions to prevent a control rod withdrawal error during reactor startup utilizing SRM signals to create the rod block signal. SRM signals are used to monitor neutron flux during refueling, shutdown and startup conditions. No design basis accident or transient analysis takes credit for rod block signals initiated by the SRMs.

Comparison to Screening Criteria:

1. The SRM control rod block is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The SRM control rod block instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient analyses.

3. The SRM control rod block signal is not a part of a primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 137) of NEDO-31466, the loss of the SRM control rod block function was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 1.

Conclusion:

Since the screening criteria have not been satisfied, the Control Rod Block LCO and Surveillances applicable to SRM instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.2 PROTECTIVE INSTRUMENTATION

LCO Statement:

The Limiting Conditions for Operation of the protective instrumentation affecting each of the following protective actions shall be as indicated in the corresponding LCO table.

3/4.2.G Initiates Control Rod Blocks

3/4.2.G.2 IRM

Discussion:

The Intermediate Range Monitor (IRM) control rod block functions to prevent a control rod withdrawal error during reactor startup utilizing IRM signals to create the rod block signal. IRMs are provided to monitor the neutron flux levels during refueling, shutdown and startup conditions. No design basis accidents or transient analysis takes credit for rod block signals initiated by IRMs.

Comparison to Screening Criteria:

1. The IRM control rod block is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The IRM control rod block instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient analyses.

3. The IRM control rod block signal is not a part of a primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 138) of NEDO-31466, the loss of the IRM control rod block function was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 1.

Conclusion:

Since the screening criteria have not been satisfied, the Control Rod Block LCO and Surveillances applicable to IRM instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.2 PROTECTIVE INSTRUMENTATION

LCO Statement:

The Limiting Conditions for Operation of the protective instrumentation affecting each of the following protective actions shall be as indicated in the corresponding LCO table.

3/4.2.G Initiates Control Rod Blocks

3/4.2.G.3 APRM

Discussion:

The Average Power Range Monitor (APRM) control rod block functions to prevent a control rod withdrawal error during power range operations utilizing LPRM signals to create the APRM rod block signal. APRMs provide information about the average core power and APRM rod blocks are not used to mitigate a DBA or transient.

Comparison to Screening Criteria:

1. The APRM control rod block is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The APRM control rod block instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient analyses.

3. The APRM control rod block signal is not a part of a primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 135) of NEDO-31466, the loss of the APRM control rod block function was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 1.

Conclusion:

Since the screening criteria have not been satisfied, the Control Rod Block LCO and Surveillances applicable to IRM instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.2 PROTECTIVE INSTRUMENTATION

LCO Statement:

The Limiting Conditions for Operation of the protective instrumentation affecting each of the following protective actions shall be as indicated in the corresponding LCO table.

3/4.2.G Initiates Control Rod Block

3/4.2.G.5 Scram Discharge Volume

Discussion:

The SDV control rod block functions to prevent control rod withdrawals during power range operations, utilizing scram discharge volume (SDV) signals to create the rod block signal if water is accumulating in the SDV. The purpose of measuring the SDV water level is to ensure that there is sufficient volume to contain the water discharged by the control rod drives during a scram, thus ensuring that the control rods will be able to insert fully. This rod block signal provides an indication to the operator that water is accumulating in the SDV and prevents further rod withdrawals. With continued water accumulation, a reactor protection system initiated scram signal will occur. Thus, the SDV water level rod block signal provides an opportunity for the operator to take action to avoid a subsequent scram. No design basis accident or transient takes credit for rod block signals initiated by the SDV instrumentation.

Comparison to Screening Criteria:

1. The SDV control rod block is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The SDV control rod block instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient analyses.

3. The SDV control rod block signal is not a part of a primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 139) of NEDO-31466, the loss of the SDV control rod block function was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 1.

Conclusion:

Since the screening criteria have not been satisfied, the Control Rod Block LCO and Surveillances applicable to SDV instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.2 PROTECTIVE INSTRUMENTATION

LCO Statement:

The Limiting Conditions for Operation of the protective instrumentation affecting each of the following protective actions shall be as indicated in the corresponding LCO table.

3/4.2.H Limit Radioactive Release

3/4.2.H.1 Off-Gas Post Treatment Radiation Monitors

Discussion:

The radioactive gas processing system is not a safety system and is not connected to the primary coolant piping. The off-gas post treatment monitors are used to show conformance with the discharge limits of 10 CFR 20. There is another Specification (which is being retained-proposed LCO 3.7.6) that ensures 10 CFR 100 limits are not exceeded. Information provided by these instruments on the radiation levels would have limited or no use in identifying/assessing core damage and they are not installed to detect excessive reactor coolant leakage.

Comparison to Deterministic Screening Criteria:

1. These monitors are not used for, nor are capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The monitored parameters are not assumed as initial conditions of a DBA or transient analyses that assumes the failure of, or presents a challenge to the integrity of a fission product barrier.

3. These monitors do not act as part of a primary success path in the mitigation of a DBA or transient that assumes the failure of, or presents a challenge to the integrity of a fission product barrier.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 145) of NEDO-31466, the loss of these monitors was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 1.

Conclusion:

Since the screening criteria have not been satisfied, the Off-Gas Post Treatment Radiation Monitors LCO and Surveillance may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.2 PROTECTIVE INSTRUMENTATION

LCO Statement:

The Limiting Conditions for Operation of the protective instrumentation affecting each of the following protective actions shall be as indicated in the corresponding LCO table.

3/4.2.K Provides Surveillance Information

Discussion:

Each individual accident monitoring parameter has a specific purpose, however, the general purpose for all accident monitoring instrumentation is to provide sufficient information to confirm an accident is proceeding per prediction, i.e. automatic safety systems are performing properly, and deviations from expected accident course are minimal.

Comparison to Deterministic Screening Criteria:

The NRC position on application of the deterministic screening criteria to post-accident monitoring instrumentation is documented in letter dated May 7, 1988 from T.E. Murley (NRC) to R.F. Janecek (BWROG). The position was that the post-accident monitoring instrumentation table list should contain, on a plant specific basis, all Regulatory Guide 1.97 Type A instruments specified in the plant's Safety Evaluation Report (SER) on Regulatory Guide 1.97, and all Regulatory Guide 1.97 Category 1 instruments. Accordingly, this position has been applied to the Plant Hatch Unit 1 Regulatory Guide 1.97 instruments. Those instruments meeting this criteria have remained in Technical Specifications. The instruments not meeting this criteria may be relocated from the Technical Specifications to plant controlled documents.

The following summarizes the Plant Hatch Unit 1 position for those instruments currently in Technical Specifications.

From NRC SER dated 7/30/85, Subject: Conformance to R.G. 1.97.

Type A Variables

1. Reactor Pressure
2. Drywell Temperature
3. Suppression Chamber Water Temperature
4. Hydrogen and Oxygen Analyzer

Other Type, Category 1 Variables

1. Reactor Vessel Water Level
2. Shroud Water Level
3. Drywell Pressure
4. Suppression Chamber Water Level
5. Drywell High Range Pressure
6. Drywell High Range Radiation

For other post-accident monitoring instrumentation currently in Technical Specifications, their loss is not considered risk-significant since the variable they monitor does not qualify as a Type A or Category 1 variable (one that is important to safety, and needed by the operator so that the operator can perform necessary manual actions).

Conclusion

Since the screening criteria have not been satisfied for non-Regulatory Guide 1.97 Type A or Category 1 variable instruments, their associated LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications. The instruments to be relocated are as follows:

1. Suppression Chamber Air Temperature
2. Suppression Chamber Pressure
3. Rod Position Information System
4. Post LOCA Radiation Monitoring System
5. Safety/Relief Valve Position Indication
6. Main Stack Post-Accident Effluent Monitor
7. Reactor Building Vent Plenum Post-Accident Effluent Monitor

3/4.6.F.2 CONDUCTIVITY AND CHLORIDE

LCO Statement: (paraphrased)

The chemistry of the reactor coolant system shall be maintained within the limits specified in 3/4.6.F.2.a, b, and d.

Discussion:

Poor reactor coolant water chemistry may contribute to the long term degradation of system materials and thus is not of immediate importance to the plant operator. Reactor coolant water chemistry is monitored for a variety of reasons. One reason is to reduce the possibility of failures in the reactor coolant system pressure boundary caused by corrosion. Severe chemistry transients have resulted in failure of thin walled LPRM instrument dry tubes in a relatively short period of time. However, these LPRM dry tube failures result in loss of the LPRM function and are readily detectable. In summary, the chemistry monitoring activity serves a long term preventative rather than mitigative purpose.

Comparison to Screening Criteria:

1. Reactor coolant water chemistry is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Reactor coolant water chemistry is not used to monitor a process variable that is an initial condition of a DBA or transient.

3. Reactor coolant water chemistry is not supportive of any primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 211) of NEDO-31466, the reactor coolant water chemistry was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 1.

Conclusion:

Since the screening criteria have not been satisfied, the Conductivity and Chloride LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.6.K STRUCTURAL INTEGRITY

LCO Statement:

The structural integrity of ASME Code Class 1, 2, and 3 (equivalent) components shall be maintained in accordance with the Surveillance Requirements of Specification 4.6.K.

Discussion:

The inservice testing requirements on pumps and valves required by 4.6.K.1 have been moved to Specification 5.5.6 and are not part of this discussion.

The inspection programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity of these components will be maintained throughout the components life. Other Technical Specifications require important systems to be operable (for example, ECCS 3/4.5.1) and in a ready state for mitigative action. This Technical Specification is more directed toward prevention of component degradation and continued long term maintenance of acceptable structural conditions. Hence it is not necessary to retain this Specification to ensure immediate operability of safety systems.

Further, this Technical Specification prescribes inspection requirements which are performed during plant shutdown. It is, therefore, not directly important for responding to design basis accidents.

Comparison to Screening Criteria:

1. The inspections stipulated by this Specification are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The inspections stipulated by this Specification do not monitor process variables that are initial assumptions in a DBA or transient analyses.

3. The ASME Code Class 1, 2, and 3 components inspected per this Specification are assumed to function to mitigate a DBA. Their capability to perform this function is addressed by other Technical Specifications. This Technical Specification, however, only specifies inspection requirements for these components; and these inspections can only be performed when the plant is shutdown. Therefore, Criterion 3 is not satisfied.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 216) of NEDO-31466, the assurance of operability of the entire system as verified in the system operability Specification dominates the risk contribution of the system. As such, the lack of a long term assurance of structural integrity Specification was found to be a non-significant risk contributor to core damage frequency and offsite releases. Furthermore, the requirement is currently covered by 10 CFR 50.55a and the plant's Inservice Inspection Program. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 1.

Conclusion:

Since the screening criteria have not been satisfied, the Structural Integrity LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.8.A MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

LCO Statement:

1. The leakage test shall be capable of detecting the presence of 0.005 microcurie of radioactive material on the test sample. If the test reveals the presence of 0.005 microcurie or more of removable contamination, it shall immediately be withdrawn from use, decontaminated and repaired, or disposed of in accordance with Commission regulations. Sealed sources are exempt from such leak tests when the source contains 100 microcuries or less of beta and/or gamma emitting material or 10 microcuries or less of alpha emitting material.

2. A complete inventory of radioactive materials in possession shall be maintained current at all times.

Discussion:

The limitations on sealed source contamination are intended to ensure that the total body or individual organ irradiation doses does not exceed allowable limits in the event of ingestion or inhalation. This is done by imposing a maximum limitation of $\leq$ 0.005 microcuries of removable contamination on each sealed source. This requirement and the associated Surveillance Requirements bear no relation to the conditions or limitations which are necessary to ensure safe reactor operation.

Comparison to Screening Criteria:

1. Miscellaneous radioactive materials sources are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Miscellaneous radioactive materials sources are not a process variable that is an initial condition of a DBA or transient.

3. Miscellaneous radioactive materials sources are not used in any part of a primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 267) of NEDO-31466, the miscellaneous radioactive materials sources being not within limits were found to be non-significant risk contributors to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 1.

Conclusion:

Since the screening criteria have not been satisfied, the Miscellaneous Radioactive Materials Sources LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.10.F REACTOR BUILDING CRANES
3/4.10.G SPENT FUEL CASK LIFTING TRUNNIONS AND YOKE

LCO Statement:

3/4.10.F

1. Main Reactor Building Crane

The initial lifting of a spent fuel cask shall not be undertaken without prior AEC approval. The reactor building crane will be modified to increase its ability to withstand a single failure prior to lifting a spent fuel cask. Technical Specifications governing spent fuel cask handling will be developed prior to lifting the first spent fuel cask.

2. Limiting Height Above Refueling Floor

See note for Specification 3.10.F.1 above.

3. Monorail Hoist

The monorail hoist shall be operating properly whenever new fuel or the fuel pool gates are handled.

3/4.10.G

Spent Fuel Cask Lifting Trunnions and Yoke

See note for specification 3.10.F.1 above.

Discussion:

Operability of the above mentioned equipment (cranes, hoists and lifting equipment) ensures that only the proper equipment will be used to handle fuel or casks within the storage pool, hoists have sufficient load capacity for handling fuel assemblies or other loads and the possibility of dropping a load on the core internals and pressure vessel are minimized during lifting operations. The design of the reactor building and crane is such that casks of current design cannot be lifted more than two feet above the refueling floor. An analysis has been made which shows that the floor over which the spent fuel cask is handled can satisfactorily sustain a dropped cask from a height of 2 feet. Administratively limiting the height that the spent fuel cask is raised over the refueling floor serves as a backup to minimize the damage that could result from an accident. Although this Technical Specification supports a refueling accident analysis, the crane limits are not monitored and controlled during operation; they are checked on a periodic basis to ensure operability. The deterministic criteria for Technical Specification retention are, therefore, not satisfied.

Comparison to Screening Criteria:

1. The reactor building crane and lifting equipment are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The reactor building crane and lifting equipment are not used to monitor a process variable that is an initial condition of a DBA or transient.

3. The reactor building crane and lifting equipment are not part of a primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 287) of NEDO-31466, the reactor building crane and lifting equipment was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 1.

Conclusion:

Since the screening criteria have not been satisfied, the Reactor Building Crane and Spent Fuel Cask Lifting Trunnions and Yoke LCOs and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.10.I CRANE TRAVEL - SPENT FUEL STORAGE POOL

LCO Statement:

A maximum weight of 1600 pounds may be permitted to be transported over stored spent fuel in order to minimize the consequences of a load handling accident.

Discussion:

The Technical Specification limit of 1600 pounds for loads over the spent fuel contained in the storage pool is further reduced by the heavy loads analysis to 725 pounds or the weight of a single fuel bundle. This 725 pound imposed limit ensures that in the event the load is dropped, the activity release will be bounded by the analysis of the refueling accident and any possible distortion of the fuel in the storage racks will not result in a critical array. Administrative monitoring of loads moving over the fuel storage racks serves as a backup to the Unit 1 crane interlocks. While the Unit 2 crane does not have interlocks, its use is strictly governed by administrative controls.

Although this Technical Specification supports the maximum refueling accident assumption in the DBA, the applicable fuel handling crane travel limits are not monitored and controlled during operation; they are checked on a periodic basis to ensure operability. The deterministic criteria for Technical Specification retention are, therefore, not satisfied.

Comparison to Screening Criteria:

1. The fuel handling crane travel limits are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The maximum severity assumed for the fuel handling DBA is limited by the limits placed on the crane travel. These crane travel limits are not, however, process variables monitored and controlled by the operator. They are interlocks and/or physical stops. Therefore, Criterion 2 is not satisfied.

3. The fuel handling crane travel limits are not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA.

Traditional PRAs do not review risks associated with the spent fuel storage pool. Design basis analyses indicate that the release associated with fuel assembly damage in the spent fuel storage pool due to crane accidents is significantly lower than releases of concern evaluated by PRAs.

Conclusion:

Since the screening criteria have not been satisfied, the Crane Travel - Spent Fuel Storage Pool LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.14.2 EXPLOSIVE GAS MONITORING INSTRUMENTATION

LCO Statement:

The explosive gas monitoring instrumentation channels shown in table 3.14.2-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.15.2.6 are not exceeded.

Discussion:

The explosive gas monitor Specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the gaseous radwaste treatment system are adequately monitored. This will help ensure that the concentration is maintained below the flammability limit of hydrogen. However, the offgas system is designed to contain detonation and its loss of function will not affect the function of any safety related equipment. The concentration of hydrogen in the offgas stream is not an initial assumption of any design basis accident or transient analysis.

Comparison to Screening Criteria:

1. The explosive gas monitoring instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The explosive gas monitoring instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient. Excessive system effluent is not an indication of a DBA or transient.

3. The explosive gas monitoring instrumentation is not part of a primary success path in the mitigation of a DBA or transient. Excessive discharge is not considered to initiate a primary success path in mitigating a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Items 189 and 306) of NEDO-31466, the loss of the explosive gas monitoring instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has review this evaluation and considers it applicable to Plant Hatch Unit 1.

Conclusion:

Since the screening criteria have not been satisfied, the Explosive Gas Monitoring Instrumentation LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

APPENDIX B

PLANT SPECIFIC

RISK SIGNIFICANT EVALUATIONS

APPENDIX B

TECHNICAL SPECIFICATION:

3/4.2.B.3 HPCI Turbine Overspeed - Mechanical
3/4.2.B.4 HPCI Turbine Exhaust Pressure - High
3/4.2.B.5 HPCI Pump Suction Pressure - Low

DESCRIPTION OF REQUIREMENT:

These protective trips function to reduce the probability of pump/turbine damage under certain conditions. The above listed Technical Specifications specify the minimum number of operable channels per trip system; the limiting trip settings; and the minimum calibration, functional test and channel check frequencies (channel check and functional test are not applicable to the mechanical overspeed trip system).

REFERENCES:

Hatch IPE
Monticello IPE
NUREG 1150 (Peach Bottom)

DISCUSSION:

PRAs do not model conditions in which these trips are required to prevent pump/turbine damage. If a condition requiring one of the above trips exists, it is typically assumed that HPCI is unavailable for the remainder of the event. In a risk analysis, it is irrelevant whether HPCI is unavailable because it tripped on overspeed due to a control system failure or failed catastrophically. The following insights for the above trip signals were extracted from a review of PRAs:

1. Spurious operation of any protective trip could result in failure of HPCI to inject. The Specifications provide only a maximum setting for the overspeed and exhaust pressure trips and a minimum setting for the low suction pressure trip. Thus, the limiting settings do not provide protection against spurious trips. The functional testing required by these Specifications would provide some protection against spurious trips from setpoint drift, but setpoint drift of any magnitude away from the limiting setting would not be a violation of these specifications. Relocation of the Specifications will not result in any measurable impact on HPCI reliability for the following reasons:

   a. Spurious trips are historically a minor contributor to HPCI unavailability, by comparison with active hardware failures and maintenance unavailability.

   b. Other surveillance tests, some of which remain in the Technical Specifications, provide protection against spurious trips. For example, Technical Specifications require HPCI to be tested quarterly.

c. The issue of protection against spurious trips is purely a reliability issue. There are many other activities that affect HPCI reliability which are not controlled by Technical Specifications. As an example, the total unavailability due to maintenance is a key factor in overall HPCI unavailability: yet the Technical Specifications only limit the duration of individual maintenance events. Other aspects, such as the frequency of maintenance and the minimization of downtime, are left within the control of plant management. Because plant management already controls many of the factors affecting HPCI reliability via mechanisms not controlled by the Technical Specifications, the relocation of the above testing requirements is not expected to have an appreciable impact on HPCI reliability.

2. The high exhaust pressure trip is used to trip the HPCI turbine on indication that an exhaust line blockage exists. An additional signal, exhaust diaphragm high pressure, is used to isolate the steam supply to the turbine. This signal will also provide a trip signal to the turbine, regardless of the availability of the exhaust pressure signal. The frequency of exhaust line blockage combined with both signals failing to trip the HPCI turbine is assessed as low, and the risk significance is consequently small.

3. The low suction pressure trip instrumentation provides protection against mispositioned or failed closed valves in the HPCI pump suction lines. A mispositioned or failed closed valve could result in damage to the HPCI pump. From a PRA perspective, if the suction paths for HPCI are failed, it is irrelevant whether HPCI fails due to cavitation problems or successfully trips, as it will be unavailable for the remainder of the transient. Human errors leading to mispositioned valves that are normally locked open and hardware failures of locked open valves are historically small contributors to HPCI unavailability.

4. The mechanical overspeed could be called upon to function following a control system failure or during an event in which HPCI was operating when DC power was lost. These two cases are discussed below:

   a. With a control system failure, HPCI will be unavailable for the remainder of the event. From a PRA perspective, regardless of whether the overspeed trip functions or not, HPCI is unavailable for continued injection.

b. The mechanical overspeed could be called upon to actuate if HPCI is operating, and DC power is depleted. Station blackout would be an example of this type of scenario. The response of HPCI following dc power depletion is not completely predictable, but it is possible that HPCI would overspeed. Battery depletion would result in the electrical overspeed control being unavailable, and the mechanical overspeed would be the only operable trip for HPCI. With no AC or DC power, the containment isolation valves in the HPCI steam line would remain open. The HPCI stop and control valves would function to isolate the HPCI steam line when hydraulic power was lost, regardless of whether the turbine mechanical overspeed functioned or not.

CONCLUSION:

Based on the above insights it can be concluded that Specifications 3/4.2.B.3, 3/4.2.B.4, and 3/4.2.B.5 may be relocated from the Plant Hatch Unit 1 Technical Specifications to a Plant Hatch controlled document with no significant impact on the potential for core damage and the risk of offsite consequences.

TECHNICAL SPECIFICATION:

3/4.2.C.3 RCIC Turbine Overspeed - Electrical and Mechanical
3/4.2.C.4 RCIC Turbine Exhaust Pressure - High
3/4.2.C.5 RCIC Pump Suction Pressure - Low
3/4.2.C.6 RCIC Pump Discharge Flow

DESCRIPTION OF REQUIREMENT:

These protective trips function to reduce the probability of pump/turbine damage under certain conditions. The above listed Technical Specifications specify the minimum number of operable channels per trip system; the limiting trip settings; and the minimum calibration, functional, test and channel check frequencies. (Channel check and functional test are not applicable to the mechanical overspeed trip system.)

REFERENCES:

Hatch IPE
Monticello IPE
NUREG 1150 (Peach Bottom)

DISCUSSION:

PRAs do not model conditions in which the first three instruments are required to prevent pump/turbine damage. If a condition requiring one of the above trips exists, it is typically assumed that RCIC is unavailable for the remainder of the event. In a risk analysis, it is irrelevant whether RCIC is unavailable because it tripped on overspeed due to a control system failure or failed catastrophically. The following insights for the above trip signals were extracted from a review of PRAs:

1. Spurious operation of any protective trip could result in failure of RCIC to inject. The Specifications provide only a maximum setting for the overspeed and exhaust pressure trips and a minimum setting for the low suction pressure trip. Thus, the limiting settings do not provide protection against spurious trips. The functional testing required by these Specifications would provide some protection against spurious trips from setpoint drift, but setpoint drift of any magnitude away from the limiting setting would not be a violation of these Specifications. Relocation of the Specifications will not result in any measurable impact on RCIC reliability for the following reasons:

    a. Spurious trips are historically a minor contributor to RCIC unavailability, by comparison with active hardware failures and maintenance unavailability.

    b. Other surveillance tests, some of which remain in the Technical Specifications, provide protection against spurious trips. For example, Technical Specifications require RCIC to be tested quarterly.

c. The issue of protection against spurious trips is purely a reliability issue. There are many other activities that affect RCIC reliability which are not controlled by Technical Specifications. As an example, the total unavailability due to maintenance is a key factor in overall RCIC unavailability: yet the Technical Specifications only limit the duration of individual maintenance events. Other aspects, such as the frequency of maintenance and the minimization of downtime, are left within the control of plant management. Because plant management already controls many of the factors affecting RCIC reliability via mechanisms not controlled by the Technical Specifications, the relocation of the above testing requirements is not expected to have any appreciable impact on RCIC reliability.

2. The high exhaust pressure trip is used to trip the RCIC turbine on indication that an exhaust line blockage exists. An additional signal, exhaust diaphragm high pressure, is used to isolate the steam supply to the turbine. This signal will also provide a trip signal to the turbine, regardless of the availability of the exhaust pressure signal. The frequency of exhaust line blockage combined with both signals failing to trip the RCIC turbine is assessed as low, and the risk significance is consequently small.

3. The low suction pressure trip instrumentation provides protection against mispositioned or failed closed valves in the RCIC pump suction lines. A mispositioned or failed closed valve could result in damage to the RCIC pump. From a PRA perspective, if the suction paths for RCIC are failed, it is irrelevant whether RCIC fails due to cavitation problems or successfully trips, as it will be unavailable for the remainder of the transient. Human errors leading to mispositioned valves that are normally locked open and hardware failures of locked open valves are historically small contributors to RCIC unavailability.

4. The electrical and mechanical overspeeds could be called upon to function following a control system failure or during an event in which RCIC was operating when DC power was lost. These two cases are discussed below (case a applies to both the electrical and mechanical overspeed trips while case b applies only to the mechanical overspeed trip):

   a. With a control system failure, RCIC will be unavailable for the remainder of the event. From a PRA perspective, regardless of whether the electrical and mechanical overspeed trips function or not, RCIC is unavailable for continued injection.

b. The mechanical overspeed could be called upon to actuate if RCIC is operating, and DC power is depleted. Station blackout would be an example of this type of scenario. The response of RCIC following DC power depletion is not completely predictable, but it is possible that RCIC would overspeed. Battery depletion would result in the electrical overspeed control being unavailable, and the mechanical overspeed would be the only operable trip for RCIC. With no AC or DC power, the containment isolation valves in the RCIC steam line would remain open. The RCIC trip throttle valve would function to isolate the RCIC steam line when hydraulic power was lost, regardless of whether the turbine mechanical overspeed functioned or not.

The fourth instrument listed above controls the minimum flow valve. The minimum flow valve for RCIC is normally closed, and must open only when no flow path is available to the vessel or the CST. The pump low flow signal provides the open signal to the minimum flow valve. Because vessel pressure is never expected to exceed the capacity of RCIC to inject, the only causes for loss of a RCIC flow path are control failures or valve failures in the discharge lines, and operator alignment errors. Control failures and valve failures would be modeled in PRAs as loss of RCIC, regardless of the availability of the minimum flow path. Operator errors that result in no flow path for RCIC are typically included in an operator error for failure to control flow, or in a human error involving system misalignment. In either case, RCIC would typically be considered failed for the remainder of the event. Failure of the minimum flow path followed by pump failure is thus a superfluous failure with regards to the immediate availability of RCIC.

CONCLUSION:

Based on the above insights, it can be concluded that Specifications 3/4.2.C.3, 3/4.2.C.4, 3/4.2.C.5, and 3/4.2.C.6 may be relocated from the Plant Hatch Unit 1 Technical Specifications to a Plant Hatch controlled document, with no significant impact on the potential for core damage and risk of offsite consequences.

TECHNICAL SPECIFICATION:

3/4.2.E.8 LPCI Injection Valve Selection Timers

DESCRIPTION OF REQUIREMENT:

These instruments function to electrically lock open the RHR outboard isolation valves for ten minutes on receipt of a LOCA signal. The outboard isolation valves are normally open, and are used to throttle flow once adequate core cooling is established. The inboard isolation valves are normally closed and are the only LPCI valves required to open for LPCI injection. The timers do not affect the LOCA signal to the inboard isolation valves, which remain electrically locked open as long as the LOCA signal exists.

REFERENCES:

Hatch IPE
Monticello IPE
NUREG 1150 (Peach Bottom)

DISCUSSION:

The following insights were extracted from a review of PRAs:

1. The selection timers have normally closed contacts that transmit the LOCA signal to the outboard isolation valves. These contacts open after 10 minutes to remove the LOCA open signal, and allow the operators to throttle flow. If the LOCA timers fail completely, such that the normally closed contacts fail to transmit the LOCA open signal to the normally open outboard injection valve, LPCI injection will still occur.

2. Core cooling is affected only when a failure of the timers is combined with additional failures and human errors. For example, a timer must fail such that the LOCA signal is cleared significantly before 10 minutes, and the timer in the other loop must fail simultaneously, or the other LPCI loop must fail to provide flow. These failures must be combined with one of the following combinations of failures: a) vessel level instrumentation fails such that the operators believe that LPCI flow should be throttled, or b) operators completely misdiagnose the situation and believe that vessel level should be controlled, or c) operators inadvertently close the LPCI valves by mistake. The combinations of errors and hardware failures that would have to occur, in combination with an event such as a large break LOCA that requires low pressure injection in the early stages of the accident, would be insignificant compared to some of the active hardware failures that could result in the same effect. For example, failure of the normally closed inboard injection valves to open would have a much higher probability than any of the combinations of human errors and hardware failures postulated above.

3. During an ATWS event with low pressure injection, the operators must throttle LPCI injection to limit reactor power. Procedures direct the operators to override the LOCA signal to these valves before the 10 minute timers expire. These procedures direct the operators to lift leads to bypass the LOCA timer, such that failure of the timer cannot impact the ability to control LPCI flow.

CONCLUSION:

Based on the above insights, relocation of Specification 3/4.2.E.8 to a Plant Hatch controlled document should have no impact on plant safety.

APPLICATION OF SELECTION CRITERIA TO THE

EDWIN I. HATCH NUCLEAR PLANT
UNIT 2
TECHNICAL SPECIFICATIONS

CONTENTS

1. INTRODUCTION

The purpose of this document is to confirm the results of the BWR Owners' Group application of the Technical Specification selection criteria on a plant specific basis for Edwin I. Hatch Nuclear Plant Unit 2. Georgia Power Company has reviewed the application of the selection criteria to each of the Technical Specifications utilized in BWROG report NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment" including Supplement 1 (Reference 1), and NUREG 1433, "Standard Technical Specification, General Electric Plants BWR/4," (Reference 2), as well as applying the criteria to each of the current Plant Hatch Unit 2 Technical Specifications. Additionally, in accordance with the NRC guidance, this confirmation of the application of selection criteria to Plant Hatch Unit 2 includes confirming the risk insights from Probabilistic Risk Assessment (PRA) evaluations, provided in Reference 1, as applicable to Plant Hatch Unit 2.

2. SELECTION CRITERIA

Georgia Power Company (GPC) has utilized the selection criteria provided in the NRC Final Policy Statement on Technical SpecificationImprovements (52 FR 3788) of July 23, 1993 (Reference 3) to develop the results contained in the attached matrix. Probabilistic Risk Assessment (PRA) insights as used in the BWROG submittal were utilized, confirmed by GPC, and are discussed in the next section of this report. The selection criteria and discussion provided in the NRC Final Policy Statement are as follows:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary:

Discussion of Criterion 1: A basic concept in the adequate protection of the public health and safety is the prevention of accidents. Instrumentation is installed to detect significant abnormal degradation of the reactor coolant pressure boundary so as to allow operator actions to either correct the condition or to shut down the plant safely, thus reducing the likelihood of a loss-of-coolant accident.

This criterion is intended to ensure that Technical Specifications control those instruments specifically installed to detect excessive reactor coolant system leakage. This criterion should not, however, be interpreted to include instrumentation to detect precursors to reactor coolant pressure boundary leakage or instrumentation to identify the source of actual leakage (e.g., loose parts monitor, seismic instrumentation, valve position indicators).

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier:

Discussion of Criterion 2: Another basic concept in the adequate protection of the public health and safety is that the plant shall be operated within the bounds of the initial conditions assumed in the existing Design Basis Accident and Transient analyses and that the plant will be operated to preclude unanalyzed transients and accidents. These analyses consist of postulated events, analyzed in the FSAR, for which a structure, system, or component must meet specified functional goals. These analyses are contained in Chapters 6 and 15 of the FSAR (or equivalent chapters) and are identified as Condition II, III, or IV events (ANSI N18.2) (or equivalent) that either assume the failure of or present a challenge to the integrity of a fission product barrier.

As used in Criterion 2, process variables are only those parameters for which specific values or ranges of values have been chosen as reference bounds in the Design Basis Accident or Transient analyses and which are monitored and controlled during power operation such that process values remain within the analysis bounds. Process variables captured by Criterion 2 are not, however, limited to only those directly monitored and controlled from the control room. These could also include other features or characteristics that are specifically assumed in Design Basis Accident and Transient analyses even if they cannot be directly observed in the control room (e.g., moderator temperature coefficient and hot channel factors).

The purpose of this criterion is to capture those process variables that have initial values assumed in the Design Basis Accident and Transient analyses, and which are monitored and controlled during power operation. As long as these variables are maintained within the established values, risk to the public safety is presumed to be acceptably low. This criterion also includes active design features (e.g., high pressure/low pressure system valves and interlocks) and operating restrictions (pressure/temperature limits) needed to preclude unanalyzed accidents and transients.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier:

Discussion of Criterion 3: A third concept in the adequate protection of the public health and safety is that in the event that a postulated Design Basis Accident or Transient should occur, structures, systems, and components are available to function or to actuate in order to mitigate the consequences of the Design Basis Accident or Transient. Safety sequence analyses or their equivalent have been performed in recent years and provide a method of presenting the plant response to an accident. These can be used to define the primary success paths.

A safety sequence analysis is a systematic examination of the actions required to mitigate the consequences of events considered in the plant's Design Basis Accident and Transient analyses, as presented in Chapters 6 and 15 of the plant's FSAR (or equivalent chapters). Such a safety sequence analysis considers all applicable events, whether explicitly or implicitly presented. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criteria), so that the plant response to Design Basis Accidents and Transients limits the consequences of these events to within the appropriate acceptance criteria.

It is the intent of this criterion to capture into Technical Specifications only those structures, systems, and components that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path for a particular mode of operation does not include backup and diverse equipment (e.g., rod withdrawal block which is a backup to the average power range monitor high Flux Trip in the startup mode, safety

Discussion of Criterion 3: (continued)
valves which are backup to low temperature overpressure relief valves during cold shutdown).

Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety:

Discussion of Criterion 4: It is the Commission policy that licensees retain in their Technical Specifications LCOs, actions statements, and Surveillance Requirements for the following systems (as applicable), which operating experience and [probabilistic safety assessment (PSA)] PSA have generally shown to be significant to public health and safety and any other structures, systems, or components that meet this criterion:

- Reactor Core Isolation Cooling/Isolation Condenser,
- Residual Heat Removal,
- Standby Liquid Control, and
- Recirculation Pump Trip.

The Commission recognizes that other structures, systems, or components may meet this criterion. Plant- and design-specific PSAs have yielded valuable insight to unique plant vulnerabilities not fully recognized in the safety analysis report Design Basis Accident or Transient analyses. It is the intent of this criterion that those requirements that PSA or operating experience exposes as significant to public health and safety, consistent with the Commission's Safety Goal and Severe Accident Policies, be retained or included in Technical Specifications.

The Commission expects that licensees, in preparing their Technical Specification related submittals, will utilize any plant-specific PSA or risk survey and any available literature on risk insights and PSAs. This material should be employed to strengthen the technical bases for those requirements that remain in Technical Specifications, when applicable, and

Discussion of Criterion 4: (continued)

to verify that none of the requirements to be relocated contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk. Similarly, the NRC staff will also employ risk insights and PSAs in evaluating Technical Specifications related submittals. Further, as a part of the Commission's ongoing program of improving Technical Specifications, it will continue to consider methods to make better use of risk and reliability information for defining future generic Technical Specification requirements.

3. PROBABILISITIC RISK ASSESSMENT INSIGHTS

Introduction and Objectives

The Final Policy Statement includes a statement that NRC expects licensees to utilize the available literature on risk insights to verify that none of the requirements to be relocated contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk.

Those Technical Specifications proposed for relocation to other plant controlled documents will be maintained under the 10 CFR 50.59, safety evaluation review program. These Specifications have been compared to a variety of Probabilistic Risk Assessment (PRA) material with two purposes: 1) to identify if a component or variable is addressed by PRA, and 2) to judge if the component or variable is risk-important. In addition, in some cases risk was judged independent of any specific PRA material. The intent of the review was to provide a supplemental screen to the deterministic criteria. Those Technical Specifications proposed to remain a part of the Improved Technical Specifications were not reviewed. This review was accomplished in Reference 1, except where discussed in Appendix A, "Justification For Specification Relocation", and has been confirmed by GPC for those Specifications to be relocated. Where Reference 1 did not review a Technical Specification against the criteria of Reference 2, GPC performed a review similar (but not identical) to that described below for Reference 1. The results of these reviews are presented in Appendix B.

Assumptions and Approach

Briefly, the approach used in Reference 1 was the following:

> The risk assessment analysis evaluated the loss of function of the system or component whose LCO was being considered for relocation and qualitatively assessed the associated effect on core damage frequency and offsite releases. The assessment was based on available literature on plant risk insights and PRAs. Table 3-1 lists the PRAs used for making the assess-

ments. A detailed quantitative calculation of the core damage and offsite release effects was not performed. However, the analysis did provide an indication of the relative significance of those LCOs proposed for relocation on the likelihood or severity of the accident sequences that are commonly found to dominate plant safety risks. The following analysis steps were performed for each LCO proposed for relocation:

a. List the function(s) affected by removal of the LCO item.

b. Determine the effect of loss of the LCO item on the function(s).

c. Identify compensating provisions, redundancy, and backups related to the loss of the LCO item.

d. Determine the relative frequency (high, medium, and low) of the loss of the function(s) assuming the LCO item is removed from Technical Specifications and controlled by other procedures or programs. Use information from current PRAs and related analyses to establish the relative frequency.

e. Determine the relative significance (high, medium, and low) of the loss of the function(s). Use information from current PRAs and related analyses to establish the relative significance.

f. Apply risk category criteria to establish the potential risk significance or non-significance of the LCO item. Risk categories were defined as follows:

RISK CRITERIA

	Consequence		
Frequency	High	Medium	Low
High	S	S	NS
Medium	S	S	NS
Low	NS	NS	NS

S = Potential Significant Risk Contributor
NS = Risk Non-Significant

g. List any comments or caveats that apply to the above assessment. The output from the above evaluation was a list of LCOs proposed for relocation that could have potential plant safety risk significance if not properly controlled by other procedures or programs. As a result these Specifications will be relocated to other plant controlled documents outside the Technical Specifications.

TABLE 3-1

BWR PRAs USED IN NEDO 31466 (AND SUPPLEMENT 1)

RISK ASSESSMENT

- BWR/6 Standard Plant, GESSAR II, 238 Nuclear Island, BWR/6 Standard Plant Probabilistic Risk Assessment, Docket No. STN 50-447, March 1982.

- La Salle County Station, NEDO-31085, Probabilistic Safety Analysis, February 1988.

- Grand Gulf Nuclear Station, IDCOR, Technical Report 86.2GG, Verification of IPE for Grand Gulf, March 1987.

- Limerick, Docket Nos. 50-352, 50-353, 1981, "Probabilistic Risk Assessment, Limerick Generating Station", Philadelphia Electric Company.

- Shoreham, Probabilistic Risk Assessment Shoreham Nuclear Power Station, Long Island Lighting Company, SAI-372-83-PA-01, June 24, 1983.

- Peach Bottom 2, NUREG-75/0104, "Reactor Safety Study", WASH-1400, October 1975.

- Millstone Point 1, NUREG/CR-3085, "Interim Reliability Evaluation Program: Analysis of the Millstone Point Unit 2 Nuclear Power Plant", January 1983.

- Grand Gulf, NUREG/CR-1659, "Reactor Safety Study Methodology Applications Program: Grand Gulf #1 BWR Power Plant", October 1981.

- NEDC-30936P, "BWR Owners' Group Technical Specification Improvement Methodology (with Demonstration for BWR ECCS Activation Instrumentation) Part 2", June 1987.

4. RESULTS OF APPLICATION OF SELECTION CRITERIA

The selection criteria from Section 2 were applied to the Plant Hatch Unit 2 Technical Specifications. The attachment is a summary of that application indicating which Specifications are being retained or relocated. Discussions that document the rationale for the relocation of each Specification which failed to meet the selection criteria are provided in Appendix A. No Significant Hazards Determination (10 CFR 50.92) evaluations for those Specifications relocated are provided with the Discussion of Changes for the specific Technical Specifications. GPC will relocate those Specifications identified as not satisfying the criteria to plant specific controlled documents whose changes are governed by 10 CFR 50.59.

5. REFERENCES

1. NEDO-31466 (and Supplement 1), "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.

2. NUREG 1433, "Standard Technical Specifications, General Electric Plants BWR/4," September 1992.

3. NRC No. 93-102 "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

ATTACHMENT

SUMMARY DISPOSITION MATRIX
PLANT HATCH UNIT 2

SUMMARY DISPOSITION MATRIX
PLANT HATCH UNIT 2

Current Unit 2 TS Number	Title	New Unit 2 TS Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion[a][c]
1.0	DEFINITIONS	1.1 3.10.1	Yes	See Note 1 and Note 4.
2.1	SAFETY LIMITS	2.0		
2.1.1	THERMAL POWER (Low Pressure or Low Flow)	2.1.1.1	Yes	See Note 2.
2.1.2	THERMAL POWER (High Pressure and High Flow)	2.1.1.2	Yes	See Note 2.
2.1.3	Reactor Coolant System Pressure	2.1.2	Yes	See Note 2.
2.1.4	Reactor Vessel Water level	2.1.1.3	Yes	See Note 2.
2.2	LIMITING SAFETY SYSTEM SETTINGS			
2.2.1	Reactor Protection System Instrumentation Setpoints	3.3.1.1	Yes	The application of Technical Specification selection criteria is not appropriate. However, the RPS LSSS have been included as part of the RPS instrumentation Specification which has been retained since the Functions either actuate to mitigate consequences of design basis accidents and transients or are retained as directed by the NRC as the Functions are part of the RPS.
3.0	LIMITING CONDITIONS FOR OPERATION - APPLICABILITY			
3.0.1	Operational Conditions	LCO 3.0.1	Yes	See Note 3.
3.0.2	Noncompliance	LCO 3.0.2	Yes	See Note 3.
3.0.3	Generic Actions	LCO 3.0.3	Yes	See Note 3.
3.0.4	Entry into Operational Conditions	LCO 3.0.4	Yes	See Note 3.
3.0.5	Operability Exception	3.8.1	Yes	The application of Technical Specification selection criteria is not appropriate. However, this exception to the definition of OPERABILITY has been included as a part of Required Actions in LCO 3.8.1.

Current Unit 2 TS Number	Title	New Unit 2 TS Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion[a][c]
4.0	SURVEILLANCE REQUIREMENTS - APPLICABILITY			
4.0.1	Operational Conditions	SR 3.0.1	Yes	See Note 3.
4.0.2	Time of Performance	SR 3.0.2	Yes	See Note 3.
4.0.3	Noncompliance	SR 3.0.3	Yes	See Note 3.
4.0.4	Entry into Operational Conditions	SR 3.0.4	Yes	See Note 3.
4.0.5	ASME Code Class 1, 2, 3 Components	5.5.6	Yes	See Note 3.
3/4.1	REACTIVITY CONTROL SYSTEMS	3.1		
3/4.1.1	Shutdown Margin	3.1.1	Yes-2	Not a measured process viable, but is important an parameter used to confirm the acceptability of the accident analysis. In addition, the LCO is retained as directed by the NRC.
3/4.1.2	Reactivity Anomalies	3.1.2	Yes-2	Confirms assumptions made in the reload safety analysis.
3/4.1.3	Control Rods			
3/4.1.3.1	Control Rod Operability	3.1.3	Yes-3	Control rods are part of the primary success path in mitigating the consequences of design basis accidents (DBAs) and transients.
3/4.1.3.2	Control Rod Maximum Scram Insertion Times	3.1.3	Yes-3	Same as above.
3/4.1.3.3	Control Rod Average Scram Insertion Times	3.1.4	Yes-3	Same as above.
3/4.1.3.4	Four Control Rod Group Scram Insertion Times	3.1.4	Yes-3	Same as above.
3/4.1.3.5	Control Rod Scram Accumulators	3.1.5 3.9.5	Yes-3	Same as above.
3/4.1.3.6	Control Rod Drive Coupling	3.1.3	Yes-3	Same as above.
3/4.1.3.7	Control Rod Position Indication	3.1.3 3.9.4	Yes-3	Same as above.
3/4.1.3.8	Control Rod Drive Housing Support	Deleted	No	See CRD Housing Support System technical change discussion.
3/4.1.4	Control Rod Program Controls			
3/4.1.4.1	Rod Worth Minimizer	3.3.2.1.2	Yes-3	Prevents withdrawal of control rods outside BPWS constraints that might set-up high rod worth conditions beyond CRDA assumptions.
3/4.1.4.2	Deleted in Amendment No. 121			

Current Unit 2 TS Number	Title	New Unit 2 TS Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion[a][c]
3/4.1.4.3	Rod Block Monitor	3.3.2.1.1	Yes-3	Prevents continuous withdrawal of a high worth control rod that could challenge the MCPR Safety Limit.
3/4.1.5	Standby Liquid Control System	3.1.7	Yes-4	Being retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements due to risk significance.
3/4.1.6	Scram Discharge Volume Vent and Drain Valves	3.1.8	Yes-3	Contributes to the operability of the control rod scram function.
None	Rod Pattern Control	3.1.6	Yes-3	Assures initial conditions for the CRDA analysis are maintained.
3/4.2	POWER DISTRIBUTION LIMITS	3.2		
3/4.2.1	Average Planar Linear Heat Generation Rate	3.2.1	Yes-2	Peak cladding temperature following a LOCA is primarily dependent on initial APLHGR. As such it is an initial condition of a DBA analysis.
3/4.2.2	Deleted in Amendment No. 39			
3/4.2.3	Minimum Critical Power Ratio	3.2.2	Yes-2	Utilized as an initial condition of the design basis transients. Transient analysis are performed to establish the largest reduction in Critical Power Ratio. This value is added to the fuel cladding integrity safety limit to determine the MCPR value.
3/4.2.4	Linear Heat Generation Rate	Deleted	No	Deleted. See technical change discussion for LHGR.
3/4.3[b]	INSTRUMENTATION	3.3		
3/4.3.1	Reactor Protection System Instrumentation	3.3.1.1	Yes-3	Retained as directed by the NRC as it is part of the RPS, or it actuates to mitigate consequences of a DBA and/or transients, or it provides an anticipatory scram to ensure the scram discharge volume and thus RPS remains operable.

Current Unit 2 TS Number	Title	New Unit 2 TS Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion[a][c]
3/4.3.2	Isolation Actuation Instrumentation	3.3.6.1 3.3.6.2	Yes-3,4	Actuates to mitigate the consequences of a DBA LOCA, or actuates to mitigate the consequences of a DBA LOCA release to the environment and a fuel handling accident, or actuates to isolate potential leakage paths to secondary containment consistent with safety analysis assumptions, or is retained due to risk significance, or is retained due to importance of RHR system and risk significance.
3/4.3.2.1.c.1	Main Steam Line Radiation - High	Deleted	No	Deleted. See Primary Containment Isolation Instrumentation technical change discussion for MSLRM.
3/4.3.2.4.j	Logic Power Monitor	Deleted	No	Deleted. See Primary Containment Isolation Instrumentation technical change discussion.
3/4.3.2.5.i	Logic Power Monitor	Deleted	No	Deleted. See Primary Containment Isolation Instrumentation technical change discussion.
3/4.3.3	Emergency Core Cooling System Actuation Instrumentation	3.3.5.1	Yes-3,4	ECCS mitigate the consequences of a DBA LOCA, or is being retained due to risk significance, or functions to mitigate the consequences of a small break LOCA, or a minimum number of S/RVs is assumed to function in the containment loading safety analysis.
3/4.3.3.1.d	Logic Power Monitor	Deleted	No	Deleted. See ECCS Instrumentation technical change discussion.
3/4.3.3.2.g	Logic Power Monitor	Deleted	No	Deleted. See ECCS Instrumentation technical change discussion.
3/4.3.3.3.e	Logic Power Monitor	Deleted	No	Deleted. See ECCS Instrumentation technical change discussion.
3/4.3.3.4.h	Control Power Monitor	Deleted	No	Deleted. See ECCS Instrumentation technical change discussion.
3/4.3.4	Reactor Core Isolation Cooling System Actuation Instrumentation	3.3.5.2	Yes-4	Retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements due to risk significance.
3/4.3.5	Control Rod Withdrawal Block Instrumentation	3.3.2.1		
3/4.3.5.1	APRM	Relocated	No	See Appendix A, Page 1.
3/4.3.5.2	Rod Block Monitor	3.3.2.1.1	Yes-3	Prevents continuous withdrawal of a high worth control rod that could challenge the MCPR Safety Limit.

Current Unit 2 TS Number	Title	New Unit 2 TS Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion[a][c]
3/4.3.5.3	Source Range Monitors	Relocated	No	See Appendix A, Page 2.
3/4.3.5.4	Intermediate Range Monitors	Relocated	No	See Appendix A, Page 3.
3/4.3.5.5	Scram Discharge Volume	Relocated	No	See Appendix A, Page 4.
3/4.3.6	Monitoring Instrumentation			
3/4.3.6.1	Radiation Monitoring Instrumentation			
3/4.3.6.1.1	Off-Gas Post Treatment Monitors	Relocated	No	See Appendix A, Page 5.
3/4.3.6.1.2	Control Room Intake Monitors	3.3.7.1	Yes-3	Actuates to maintain control room habitability so that operation can continue from the control room following DBAs.
3/4.3.6.2	Seismic Monitoring Instrumentation	Relocated	No	See Appendix A, Page 6.
3/4.3.6.3	Remote Shutdown Monitoring Instrumentation	3.3.3.2	Yes-4	Retained as directed by the NRC as it is a significant contributor to risk reduction.
3/4.3.6.4	Post-Accident Monitoring Instrumentation	3.3.3.1	Yes-3	RG 1.97 Type A and Category 1 varisbles retained. See Appendix A, Page 7 for full discussion of all variables.
3/4.3.6.5	Source Range Monitors	3.3.1.2	Yes	Does not satisfy the selection criteria, however is being retained because the NRC considers it necessary for flux monitoring during shutdown, startup and refueling operations.
3/4.3.6.6	Traversing Incore Probe System	Relocated	No	See Appendix A, Page 9.
3/4.3.6.7	MCRECS Actuation Instrumentation	3.3.7.1		
3/4.3.6.7.1	Reactor Vessel Water Level - Low Low Low (Level 1)	Relocated	No	See Appendix A, Page 10.
3/4.3.6.7.2	Drywell Pressure - High	Relocated	No	Same as above.
3/4.3.6.7.4	Main Steam Line Flow - High	Relocated	No	Same as above.
3/4.3.6.7.5	Refueling Floor Area Radiation - High	Relocated	No	Same as above.
3/4.3.6.7.6	Control Room Air Inlet Radiation - High	3.3.7.1	Yes-3	Actuates to maintain control room habitability so that operation can continue from the control room following DBAs.
3/4.3.6.8	Deleted in Amendment No. 70			
3/4.3.6.9	Removed in Admentment No. 129			

Current Unit 2 TS Number	Title	New Unit 2 TS Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion[a][c]
3/4.3.6.10	Explosive Gas Monitoring Instrumentation	Relocated	No	See Appendix A, Page 11.
3/4.3.7	Turbine Overspeed Protection System	Relocated	No	See Appendix A, Page 12.
3/4.3.8	Degraded Station Voltage Protection Instrumentation	3.3.8.1		
3/4.3.8.1	4.16kv Emergency Bus Undervoltage Relay (Loss of Voltage Condition)	3.3.8.1	Yes-3	Actuates DGs to mitigate consequences of a loss of offsite power event.
3/4.3.8.2	4.16kv Emergency Bus Undervoltage Relay (Degraded Voltage Condition)	Deleted	No	Deleted. See LOP Instrumentation technical change discussion.
3/4.3.9	Recirculation Pump Trip Actuation Instrumentation			
3/4.3.9.1	ATWS Recirculation Pump Trip System Instrumentation	3.3.4.2	Yes-4	ATWS-RPT is being retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements due to risk significance.
3/4.3.9.2	End-of-Cycle Recirculation Pump Trip System Instrumentation	3.3.4.1	Yes-3	EOC-RPT aids the reactor scram in protecting fuel cladding integrity by ensuring the fuel cladding integrity safety limit is not exceeded during a load rejection or turbine trip transient.
None	Feedwater and Main Turbine Trip Instrumentation	3.3.2.2	Yes-3	Acts to limit feedwater addition to the reactor vessel on feedwater controller failure consistent with safety analysis assumptions. Limits neutron flux peak and thermal transient to avoid fuel damage.
3/4.4	REACTOR COOLANT SYSTEM	3.4		
3/4.4.1	Recirculation System			
3/4.4.1.1	Recirculation Loops	3.4.1	Yes-2	Recirculation loop flow is an initial condition in the safety analysis.
3/4.4.1.2	Jet Pumps	3.4.2	Yes-3	Jet pump operability is assumed in the LOCA analyses to assure adequate core reflood capability.
3/4.4.1.3	Idle Recirculation Loop Startup	3.4.9	Yes-2	Establishes initial conditions to operation such that operation is prohibited in areas or at temperature rate changes that might cause undetected flaws to propagate, in turn challenging the reactor coolant pressure boundary integrity.
3/4.4.2	Safety/Relief Valves			
3/4.4.2.1	Safety/Relief Valves	3.3.6.3 3.4.3	Yes-3	A minimum number of S/RVs is assumed in the safety analyses to mitigate overpressure events.
3/4.4.2.2	S/RV Low-Low Set Function	3.3.6.3	Yes-3	A minimum number of S/RVs is assumed in the containment loading safety analysis.

Current Unit 2 TS Number	Title	New Unit 2 TS Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion[a][c]
3/4.4.3	Reactor Coolant System Leakage			
3/4.4.3.1	Leakage Detection System	3.4.5	Yes-1	Leak detection is used to indicate an abnormal condition of the reactor coolant pressure boundary.
3/4.4.3.2	Operational Leakage	3.4.4	Yes-1	Leakage beyond limits would indicate an abnormal condition of the reactor coolant pressure boundary. Operation in this condition may result in reactor coolant pressure boundary failure.
3/4.4.4	Chemistry	Relocated	No	See Appendix A, Page 13.
3/4.4.5	Specific Activity	3.4.6	Yes-2	Specific activity provides an indication of the onset of significant fuel cladding failure and is an initial condition for evaluation (radiological calculations) of the consequences of an accident due to main steam line break outside containment.
3/4.4.6	Pressure/Temperature Limits			
3/4.4.6.1	Reactor Coolant System	3.4.9	Yes-2	Establishes initial conditions to operation such that operation is prohibited in areas or at temperature rate changes that might cause undetected flaws to propagate in turn challenging the reactor coolant pressure boundary integrity.
3/4.4.6.2	Reactor Steam Dome	3.4.10	Yes-3	Reactor steam dome pressure is an initial condition in the reactor vessel overpressure safety analysis.
3/4.4.7	Main Steam Line Isolation Valves	3.6.1.3	Yes-3	Main steam line isolation within specified time limits ensures the release to the environment is consistent with the assumptions in the LOCA analysis.
3/4.4.8	Structural Integrity	Relocated	No	See Appendix A, Page 14.
None	Residual Heat Removal - Hot Shutdown	3.4.7	Yes-4	Added in accordance with the NRC Interim Policy Statement on Technical Specification Improvements due to risk significance.
None	Residual Heat Removal - Cold Shutdown	3.4.8	Yes-4	Same as above.
3/4.5	EMERGENCY CORE COOLING SYSTEMS	3.5		
3/4.5.1	High Pressure Coolant Injection System	3.5.1	Yes-4	While not assumed in a licensing basis accident analysis, HPCI is considered risk significant since it functions to mitigate the consequences of small break LOCAs.
3/4.5.2	Automatic Depressurization System	3.5.1	Yes-3	Functions to mitigate the consequences of small break LOCAs.

Current Unit 2 TS Number	Title	New Unit 2 TS Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion[a][c]
3/4.5.3	Low Pressure Core Cooling Systems			
3/4.5.3.1	Core Spray System	3.5.1 3.5.2	Yes-3	Functions to mitigate the consequences of a DBA and a vessel draindown event.
3/4.5.3.2	Low Pressure Coolant Injection System	3.5.1 3.5.2	Yes-3	Same as above.
3/4.5.4	Suppression Chamber	3.5.1 3.5.2 3.6.2.2	Yes-3	Same as above.
3/4.6	CONTAINMENT SYSTEMS	3.6		
3/4.6.1	Primary Containment			
3/4.6.1.1	Primary Containment Integrity	3.6.1.1	Yes-3	Primary containment integrity functions to mitigate the consequences of a DBA.
3/4.6.1.2	Primary Containment Leakage	3.6.1.1 3.6.1.2 3.6.1.3	Yes-3	Containment leakage is an assumption utilized in the LOCA safety analysis (but it is not a process variable). Therefore, it is being retained to ensure Primary Containment Operability.
3/4.6.1.3	Primary Containment Air Lock	3.6.1.2	Yes-3	Credit for air tightness is considered in safety analysis to limit offsite dose rates during a DBA.
3/4.6.1.4	MSIV Leakage Control System	None	No	Deleted. See MSIV-LCS technical change discussion.
3/4.6.1.5	Primary Containment Structural Integrity	3.6.1.1	Yes-3	Primary containment functions to mitigate the consequences of a DBA.
3/4.6.1.6	Primary Containment Internal Pressure	3.6.1.4	Yes-2	Primary containment pressure is an initial condition in the LOCA safety analysis.
3/4.6.1.7	Drywell Average Air Temperature	3.6.1.5	Yes-2	Drywell air temperature is an initial condition in the LOCA safety analysis.
3/4.6.2	Depressurization Systems			
3/4.6.2.1	Suppression Chamber	3.6.2.1 3.6.2.2	Yes-2 & 3	Suppression pool water level and temperature are initial conditions in the DBA LOCA analysis and mitigate the consequences of the DBA.
3/4.6.2.2	Suppression Pool Cooling	3.6.2.3	Yes-3	Suppression pool cooling functions to limit the effects of a DBA.

Current Unit 2 TS Number	Title	New Unit 2 TS Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion[a][c]
None	Suppression Pool Spray	3.6.2.4	Yes-3	Suppression pool spray functions to limit the effects of a DBA.
3/4.6.3	Primary Containment Isolation Valves	3.6.1.3	Yes-3	Isolation valves function to limit DBA consequences.
3/4.6.4	Vacuum Relief			
3/4.6.4.1	Suppression Chamber-Drywell Vacuum Breakers	3.6.1.8	Yes-3	Suppression chamber - drywell vacuum breaker operation is assumed in the LOCA analysis to limit drywell pressure thereby ensuring primary containment integrity.
3/4.6.4.2	Reactor Building - Suppression Chamber Vacuum Breakers	3.6.1.7	Yes-3	Reactor building - suppression chamber vacuum breaker operation is relied on to limit negative pressure differential, secondary to primary containment, that could challenge primary containment integrity.
3/4.6.5	Secondary Containment			
3/4.6.5.1	Secondary Containment Integrity	3.6.4.1 3.6.4.2	Yes-3	Secondary containment integrity is relied on to limit the offsite dose during an accident by ensuring a release to containment is delayed and treated prior to release to the environment.
3/4.6.5.2	Secondary Containment Automatic Isolation Dampers	3.6.4.4 3.6.4.5	Yes-3	Valve operation within time limits establishes secondary containment and limits offsite dose releases to acceptable values.
3/4.6.6	Containment Atmosphere Control			
3/4.6.6.1	Standby Gas Treatment System	3.3.6.2 3.6.4.7 3.6.4.8	Yes-3	SGT operation following a DBA acts to mitigate the consequences of offsite releases.
3/4.6.6.2	Primary Containment Hydrogen Recombiner Systems	3.6.3.1	Yes-3	Operates, post LOCA, to limit hydrogen and oxygen concentrations to below explosive concentrations that might otherwise challenge containment integrity.
3/4.6.6.3	Primary Containment Hydrogen Mixing System	3.6.3.3	Yes-3	Same as above.
3/4.6.6.4	Primary Containment Oxygen Concentration	3.6.3.2	Yes-4	Oxygen concentration is limited such that when combined with hydrogen that is postulated to evolve following a LOCA the total explosive gas concentration remains below explosive levels. Therefore, containment integrity is maintained.
3/4.6.6.5	Primary Containment Purge System			
3/4.6.6.5.1	Primary Containment Purge Valves	3.6.1.3	Yes-3	Isolation valves function to limit DBA consequences.
3/4.6.6.5.2	Primary Containment Fast Acting Dampers	3.6.1.3	Yes-3	Same as above.

Current Unit 2 TS Number	Title	New Unit 2 TS Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion[a][c]
3/4.7	PLANT SYSTEMS	3.7		
3/4.7.1	Service Water Systems			
3/4.7.1.1	RHRSW System	3.7.1	Yes-3	Designed for heat removal from RHR heat exchangers following a DBA. As such, acts to mitigate the consequences of an accident.
3/4.7.1.2	Plant Service Water System	3.7.2 3.7.3	Yes-3	Designed for heat removal from various safety related systems following a DBA. As such, acts to mitigate the consequences of an accident.
3/4.7.2	Main Control Room Environmental Control System	3.7.4	Yes-3	Maintains habitability of the control room so that operators can remain in the control room following an accident. As such, it mitigates the consequences of an accident by allowing operators to continue accident mitigation activities from the control room.
None	Main Control Room Air Conditioning System	3.7.5	Yes-3	Ensures control room temperature is maintained such that control room safety related equipment remains operable following an accident. As such, functions to mitigate the consequences of an accident.
3/4.7.3	Reactor Core Isolation Cooling System	3.5.3	Yes-4	Retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements due to risk significance.
3/4.7.4	Snubbers	Deleted	No	Deleted. See technical change discussion for Snubbers.
3/4.7.5	Sealed Source Contamination	Relocated	No	See Appendix A, Page 15.
None	Main Turbine Bypass System	3.7.7	Yes-3	Acts to mitigate the consequences of a feedwater controller failure - maximum demand transient and a turbine trip with bypass event.
3/4.8	ELECTRICAL POWER SYSTEMS	3.8		
3/4.8.1	A.C. Sources			
3/4.8.1.1	A.C. Sources - Operating	3.8.1 3.8.3 3.8.4 3.8.6	Yes-3	Required to mitigate the consequences of a DBA.

Current Unit 2 TS Number	Title	New Unit 2 TS Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion[a][c]
3/4.8.1.2	A.C. Sources - Shutdown	3.8.2 3.8.3 3.8.5 3.8.6	Yes-3	Functions to mitigate the consequences of a vessel drain-down event and is needed to support NRC Final Policy Statement requirement for decay heat removal.
3/4.8.2	Onsite Power Distribution Systems			
3/4.8.2.1	A.C. Distribution - Operating	3.8.7	Yes-3	Required to mitigate the consequences of a DBA.
3/4.8.2.2	A.C. Distribution - Shutdown	3.8.8	Yes-3	Functions to mitigate the consequences of a vessel draind-own event and is being retained to support the NRC Final Policy Statement requirement for decay heat removal.
3/4.8.2.3	D.C. Distribution - Operating	3.8.4 3.8.6 3.8.7	Yes-3	Required to mitigate the consequences of a DBA.
3/4.8.2.4	D.C. Distribution - Shutdown	3.8.5 3.8.6 3.8.8	Yes-3	Functions to mitigate the consequences of a vessel drain down event and is being retained to support the NRC Final Policy Statement requirement for decay heat removal.
3/4.8.2.5	A.C. Circuits Inside Primary Containment	Relocated	No	See Appendix A, Page 16.
3/4.8.2.6	Primary Containment Penetration Conductor Overcurrent Protective Devices	Relocated	No	See Appendix A, Page 17.
3/4.8.2.7	Electric Power Monitoring for Reactor Protection System	3.3.8.2	Yes-3	Provides protection for the RPS bus powered instrumentation against unacceptable voltage and frequency conditions that could degrade the instrumentation so that it would not perform the intended safety function.
3/4.9	REFUELING OPERATIONS	3.9		
3/4.9.1	Reactor Mode Switch	3.9.1 3.9.2	Yes-3	Provides an interlock to preclude fuel loading with control rods withdrawn. Operation is assumed in the control rod removal error during refueling and fuel assembly insertion error during refueling accident analysis.
3/4.9.2	Instrumentation	3.3.1.2	Yes	Does not satisfy selection criteria but is retained because the NRC considers it necessary for flux monitoring during shutdown, startup and refueling operations.
3/4.9.3	Control Rod Position	3.9.3	Yes-2	All control rods are required to be fully inserted when loading fuel. This requirement is assumed as an initial condition in the fuel assembly insertion error during refueling accident analysis.

Current Unit 2 TS Number	Title	New Unit 2 TS Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion[a][c]
3/4.9.4	Decay Time	Relocated	No	Although this LCO satisfied criterion 2, the activities necessary prior to commencing movement of irradiated fuel ensure that there will always be 24 hours of subcriticality before movement of any irradiated fuel. Hence this Specification has been relocated.
3/4.9.5	Secondary Containment			
3/4.9.5.1	Refueling Floor	3.6.4.3	Yes-3	Secondary containment integrity is relied on to limit the offsite dose during a fuel handling accident by ensuring the release to containment is delayed and treated prior to release to the environment.
3/4.9.5.2	Secondary Containment Automatic Isolation Dampers	3.6.4.6	Yes-3	Valve operation within time limits establishes secondary containment and limits offsite dose releases to acceptable values.
3/4.9.5.3	Standby Gas Treatment System	3.6.4.9	Yes-3	Operation following a fuel handling accident acts to mitigate the consequences of offsite releases.
3/4.9.6	Communications	Relocated	No	See Appendix A, Page 19.
3/4.9.7	Crane and Hoist Operability	Relocated	No	See Appendix A, Page 20.
3/4.9.8	Crane Travel - Spent Fuel Storage Pool	Relocated	No	See Appendix A, Page 21.
3/4.9.9	Water Level - Reactor Vessel	3.9.6	Yes-2	A minimum amount of water is required to assure adequate scrubbing of fission products following a fuel handling accident.
3/4.9.10	Water Level - Spent Fuel Storage Pool	3.7.8	Yes-2	Same as above.
3/4.9.11	Control Rod Removal			
3/4.9.11.1	Single Control Rod Removal	3.10.5	Yes	See Note 4.
3/4.9.11.2	Multiple Control Rod Removal	3.10.6	Yes	See Note 4.
3/4.9.12	Reactor Coolant Circulation	3.9.7 3.9.8	Yes	Does not satisfy the selection criteria, however is being retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements.
3/4.10	SPECIAL TEST EXCEPTIONS	3.10		
3/4.10.1	Primary Containment Integrity	Deleted	No	The latitude of this Special Test Exception is not required at Hatch Unit 2.
3/4.10.2	Rod Worth Minimizer	3.10.7	Yes	See Note 4.
3/4.10.3	Shutdown Margin Demonstrations	3.10.8	Yes	See Note 4.

Current Unit 2 TS Number	Title	New Unit 2 TS Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion[a][c]
3/4.10.4	Recirculation Loops	Deleted	No	The latitude of this Special Test Exception is not required at Hatch Unit 2.
3/4.10.5	Single Control Rod Withdrawal - Cold Shutdown	3.10.4	Yes	See Note 4.
None	Reactor Mode Switch Interlock Testing	3.10.2	Yes	See Note 4.
None	Single Control Rod Withdrawal - Hot Shutdown	3.10.3	Yes	See Note 4.
3/4.11	RADIOACTIVE EFFLUENTS	None		
3/4.11.1	Liquid Effluents			
3/4.11.1.1-3/4.11.1.3	Removed in Admentment No. 129			
3/4.11.1.4	Liquid Holdup Tanks	5.5.8	Yes	Although this Specification does not meet any Criteria of the NRC Final Policy Statement, it has been retained in accordance with the NRC letter from W.T. Russell to the Industry ITS chairpersons, dated October 25, 1993.
3/4.11.2	Gaseous Effluents			
3/4.11.2.1-3/4.11.2.5	Removed in Admentment No. 129			
3/4.11.2.6	Explosive Gas Mixture	5.5.8	Yes	Same as above.
3/4.11.2.7	Main Condenser	3.7.6	Yes-2	Main condenser offgas activity is an initial condition in the offgas system failure event analysis.
3/4.11.3	Removed in Admentment No. 129			
3/4.12	Removed in Admentment No. 129			
5.0	DESIGN FEATURES	4.0	Yes	See Note 5.
6.0	ADMINISTRATIVE CONTROLS	5.0	Yes	See Note 6.

NOTES:

NOTE 1: DEFINITIONS

This section provides definitions for several defined terms used throughout the remainder of Technical Specifications. They are provided to improve the meaning of certain terms. As such, direct application of the Technical Specification selection criteria is not appropriate. However, only those definitions for defined terms that remain as a result of application of the selection criteria, will remain as definitions in this section of Technical Specifications.

NOTE 2: SAFETY LIMITS/LSSS

Application of Technical Specifications selection criteria is not appropriate. However, Safety Limits and Limiting Safety System Settings (as part of Reactor Protection System Instrumentation) will be included in Technical Specifications as required by 10 CFR 50.36.

NOTE 3: 3.0/4.0

These Specifications provide generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operation and Surveillance Requirements. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the general requirements of Plant Hatch Unit 2 Technical Specifications 3.0/4.0 will be modified consistent with NUREG 1433.

NOTE 4: SPECIAL TEST EXCEPTIONS

These Specifications are provided to allow relaxation of certain Limiting Conditions for Operation under certain specific conditions to allow testing and maintenance. They are directly related to one or more Limiting Conditions for Operations. Direct application of the Technical Specification selection criteria is not appropriate. However, those special test exceptions, directly tied to Limiting Conditions for Operation that remain in Technical Specifications, will also remain as Technical Specifications. Those special test exceptions not applicable to Plant Hatch Unit 2 have been deleted.

NOTE 5: DESIGN FEATURES

Application of Technical Specification selection criteria is not appropriate. However, Design Features will be included in Technical Specifications as required by 10 CFR 50.36.

NOTE 6: ADMINISTRATIVE CONTROLS

Application of Technical Specification selection criteria is not appropriate. However, Administrative Controls will be included in Technical Specifications as required by 10 CFR 50.36.

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a. Where a current Technical Specification is referred to as being deleted, the technical change discussion is found in the Discussion of Changes associated with the markup of the current Specification.

b. For current Technical Specification 3/4.3, Instrumentation, the Current Technical Specification number consists of the Specification number followed by the instrumentation channel number from the associated 3.3.x Table. For example, the Main Steam Line Radiation-High channel for the RPS is numbered 3/4.3.1.6, where "3/4.3.1" is the Specification number and the last "6" is the functional unit number for the IRM channels in Table 3.3.1-1.

c. The applicable accident analyses are discussed in the Bases for the individual Technical Specification.

APPENDIX A

JUSTIFICATION FOR SPECIFICATION RELOCATION

APPENDIX A

3/4.3.5 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LCO Statement:

The control rod withdrawal block instrumentation shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2.

3/4.3.5.1 APRM

Discussion:

The Average Power Range Monitor (APRM) control rod block functions to prevent a control rod withdrawal error during power range operations utilizing LPRM signals to create the APRM rod block signal. APRMs provide information about the average core power and APRM rod blocks are not used to mitigate a DBA or transient.

Comparison to Screening Criteria:

1. The APRM control rod block is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The APRM control rod block instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient analyses.

3. The APRM control rod block signal is not a part of a primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 135) of NEDO-31466, the loss of the APRM control rod block function was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 2.

Conclusion:

Since the screening criteria have not been satisfied, the Control Rod Block LCO and Surveillances applicable to APRM instrumentation may be relocated to other plant controlled documents outside the Technical Specification.

3/4.3.5 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LCO Statement:

The control rod withdrawal block instrumentation shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2.

3/4.3.5.3 Source Range Monitors

Discussion:

The Source Range Monitor (SRM) control rod block functions to prevent a control rod withdrawal error during reactor startup utilizing SRM signals to create the rod block signal. SRM signals are used to monitor neutron flux during refueling, shutdown and startup conditions. No design basis accident or transient analysis takes credit for rod block signals initiated by the SRMs.

Comparison to Screening Criteria:

1. The SRM control rod block is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The SRM control rod block instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient analyses.

3. The SRM control rod block signal is not a part of a primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 137) of NEDO-31466, the loss of the SRM control rod block function was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 2.

Conclusion:

Since the screening criteria have not been satisfied, the Control Rod Block LCO and Surveillances applicable to SRM instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.3.5 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LCO Statement:

The control rod withdrawal block instrumentation shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2.

3/4.3.5.4 Intermediate Range Monitors

Discussion:

The IRM control rod block functionS to prevent a control rod withdrawal error during reactor startup utilizing IRM signals to create the rod block signal. IRMs are provided to monitor the neutron flux levels during refueling, shutdown and startup conditions. No design basis accidents or transient analysis takes credit for rod block signals initiated by IRMs.

Comparison to Screening Criteria:

1. The IRM control rod block is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The IRM control rod block instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient analyses.

3. The IRM control rod block signal is not a part of a primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 138) of NEDO-31466, the loss of the IRM control rod block function was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 2.

Conclusion:

Since the screening criteria have not been satisfied, the Control Rod Block LCO and Surveillances applicable to IRM instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.3.5 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LCO Statement:

The control rod withdrawal block instrumentation shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2.

3/4.3.5.5 Scram Discharge Volume

Discussion:

The SDV control rod block functions to prevent control rod withdrawals during power range operations, utilizing scram discharge volume (SDV) signals to create the rod block signal if water is accumulating in the SDV. The purpose of measuring the SDV water level is to ensure that there is sufficient volume to contain the water discharged by the control rod drives during a scram, thus ensuring that the control rods will be able to insert fully. This rod block signal provides an indication to the operator that water is accumulating in the SDV and prevents further rod withdrawals. With continued water accumulation, a reactor protection system initiated scram signal will occur. Thus, the SDV water level rod block signal provides an opportunity for the operator to take action to avoid a subsequent scram. No design basis accident or transient takes credit for rod block signals initiated by the SDV instrumentation.

Comparison to Screening Criteria:

1. The SDV control rod block is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The SDV control rod block instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient analyses.

3. The SDV control rod block signal is not a part of a primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 139) of NEDO-31466, the loss of the SDV control rod block function was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 2.

Conclusion:

Since the screening criteria have not been satisfied, the Control Rod Block LCO and Surveillances applicable to SDV instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.3.6.1 RADIATION MONITORING INSTRUMENTATION

LCO Statement:

The radiation monitoring instrumentation channels shown in Table 3.3.6.1-1 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

3/4.3.6.1.1 Off-Gas Post-Treatment Monitors

Discussion:

The radioactive gas processing system is neither a safety system nor is it connected to the primary coolant piping. The off-gas post-treatment monitors are used to show conformance with the discharge limits of 10 CFR 20. There is another Specification (which is being retained - proposed LCO [3.7.6]) that ensures 10 CFR 100 limits are not exceeded. Information provided by these instruments on the radiation levels would have limited or no use in identifying/assessing core damage and they are not installed to detect excessive reactor coolant leakage.

Comparison to Deterministic Screening Criteria:

1. These monitors are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The monitored parameters are not assumed as initial conditions of a DBA or transient analyses that assumes the failure of, or presents a challenge to the integrity of a fission product barrier.

3. These monitors do not act as part of a primary success path in the mitigation of a DBA or transient that assumes the failure of, or presents a challenge to the integrity of a fission product barrier.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 145) of NEDO-31466, the loss of these monitors was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 2.

Conclusion:

Since the screening criteria have not been satisfied, the Off-Gas Post-Treatment Monitors LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.3.6.2 SEISMIC MONITORING INSTRUMENTATION

LCO Statement:

The seismic monitoring instrumentation shown in Table 3.3.6.2-1 shall be OPERABLE.

Discussion:

In the event of an earthquake, seismic instrumentation is required to permit comparison of the measured response to that used in the design basis of the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. Since this is determined after the event has occurred, it has no bearing on the mitigation of any DBA.

Comparison to Deterministic Screening Criteria:

1. These instruments are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. These instruments do not monitor a process variable that is an initial condition to a DBA or transient analyses.

3. These instruments do not act as part of a primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 151) of NEDO-31466, the loss of seismic monitoring instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 2.

Conclusion:

Since the screening criteria have not been satisfied, the Seismic Monitoring Instrumentation LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.3.6.4 POST-ACCIDENT MONITORING INSTRUMENTATION

LCO Statement:

The post-accident monitoring instrumentation channels shown in Table 3.3.6.4-1 shall be OPERABLE.

Discussion:

Each individual accident monitoring parameter has a specific purpose, however, the general purpose for all accident monitoring instrumentation is to provide sufficient information to confirm an accident is proceeding per prediction, i.e. automatic safety systems are performing properly, and deviations from expected accident course are minimal.

Comparison to Deterministic Screening Criteria:

The NRC position on application of the deterministic screening criteria to post-accident monitoring instrumentation is documented in letter dated May 7, 1988 from T.E. Murley (NRC) to R.F. Janecek (BWROG). The position was that the post-accident monitoring instrumentation table list should contain, on a plant specific basis, all Regulatory Guide 1.97 Type A instruments specified in the plant's Safety Evaluation Report (SER) on Regulatory Guide 1.97, and all Regulatory Guide 1.97 Category 1 instruments. Accordingly, this position has been applied to the Plant Hatch Unit 2 Regulatory Guide 1.97 instruments. Those instruments meeting this criteria have remained in Technical Specifications. The instruments not meeting this criteria may be relocated from the Technical Specifications to plant controlled documents.

The following summarizes the Plant Hatch Unit 2 position for those instruments currently in Technical Specifications.

From NRC SER dated 7/30/85, Subject: Conformance to R.G. 1.97.

Type A Variables

1. Reactor Vessel Pressure
2. Suppression Chamber Water Temperature
3. Drywell Temperature
4. Drywell H_2 - O_2 Analyzer

Other Type, Category 1 Variables

1. Reactor Vessel Shroud Water Level
2. Suppression Chamber Water Level
3. Drywell Pressure
4. Drywell High Range Pressure
5. Drywell High Range Radiation

For other post-accident monitoring instrumentation currently in Technical Specifications, their loss is not risk-significant since the variable they monitored did not qualify as a Type A or Category 1 variable (one that is important to safety and needed by the operator, so that the operator can perform necessary manual actions).

Conclusion

Since the screening criteria have not been satisfied for non-Regulatory Guide 1.97 Type A or Category 1 variable instruments, their associated LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications. The instruments to be relocated are as follows:

1. Suppression Chamber Pressure
2. Post-LOCA Gamma Radiation
3. Safety/Relief Valve Position
4. Main Stack Post-Accident Effluent Monitor
5. Reactor Building Vent Plenum Post-Accident Effluent Monitor

3/4.3.6.6 TRAVERSING INCORE PROBE SYSTEM

LCO Statement:

The traversing incore probe system shall be OPERABLE with:

a. Four movable detectors, drives and readout equipment to map the core, and

b. Indexing equipment to allow all four detectors to be normalized in a common location.

Discussion:

The TIP system is used for calibration of the LPRM detectors. The TIP system is positioned axially and radially throughout the core to calibrate the local power range monitors (LPRMs). When not in use the TIP instruments are retracted into a storage position outside the drywell. The TIP system supports the operability of the LPRMs. With LPRM operability addressed there is no need to address the TIP system in the Technical Specifications.

Comparison to Screening Criteria:

1. The TIP system is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The TIP system alone is not used to monitor a process variable, nor is the system a process variable that is an initial condition of a DBA or transient analyses.

3. The TIP system is not a part of a primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 183) of NEDO-31466, the loss of the TIP system was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 2.

Conclusion:

Since the screening criteria have not been satisfied, the TIP System LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.3.6.7 MAIN CONTROL ROOM ENVIRONMENTAL CONTROL SYSTEM (MCRECS) ACTUATION INSTRUMENTATION

LCO Statement:

The MCRECS actuation instrumentation channels shown in table 3.3.6.7-1 shall be OPERABLE, with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6.7-1.

3/4.3.6.7.1 Reactor Vessel Water Level - Low Low Low (Level 1)
3/4.3.6.7.2 Drywell Pressure - High
3/4.3.6.7.4 Main Steam Line Flow - High
3/4.3.6.7.5 Refueling Floor Area Radiation - High

Discussion:

These instruments provide signals to automatically place the MCREC system in the radiation protection mode. However, these instruments are anticipatory only, and are not assumed in any DBA or transient. There is another instrument (which is being retained) that provides the actuation signal assumed in the accident analysis.

Comparison to Screening Criteria:

1. These instruments are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. These instruments are not used to monitor a process variable that is an initial condition of a DBA or transient analyses.

3. These instruments are not part of a primary success path in the mitigation of a DBA or transient.

As discussed in Appendix B (Page 1 of 1) of this document, the loss of these instruments was found to be a non-significant risk contributor to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, these MCRECS Actuation Instrumentation LCOs and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.3.6.10 EXPLOSIVE GAS MONITORING INSTRUMENTATION

LCO Statement:

The explosive gas monitoring instrumentation channels shown in table 3.3.6.10-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.6 are not exceeded.

Discussion:

The explosive gas monitor Specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the gaseous radwaste treatment system is adequately monitored, which will help ensure that the concentration is maintained below the flammability limit of hydrogen. However, the offgas system is designed to contain detonations and will not affect the function of any safety related equipment. The concentration of hydrogen in the offgas stream is not an initial assumption of any design basis accident or transient analysis.

Comparison to Screening Criteria:

1. The explosive gas monitoring instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The explosive gas monitoring instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient. Excessive system effluent is not an indication of a DBA or transient.

3. The explosive gas monitoring instrumentation is not part of a primary success path in the mitigation of a DBA or transient. Excessive discharge is not considered to initiate a primary success path in mitigating a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Items 189 and 306) of NEDO-31466, the loss of the explosive gas monitoring instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 2.

Conclusion:

Since the screening criteria have not been satisfied, the Explosive Gas Monitoring Instrumentation LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.3.7 TURBINE OVERSPEED PROTECTION SYSTEM

LCO Statement:

At least one turbine overspeed protection system shall be OPERABLE.

Discussion:

This Specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are operable and will protect the turbine from excessive overspeed. Excessive overspeed could potentially result in the generation of missiles which could impact and damage safety related components, equipment or structures, depending on the size and trajectory of the missiles. Given that the probability of turbine missile damage is acceptably low, the transient due to the actuation of the turbine stop or control valves in response to a turbine overspeed event should be considered i.e. turbine trip or load rejection. For this event the closure of the turbine stop or control valves initiates the design basis transient (turbine trip or load rejection) and not the turbine overspeed itself. The overspeed instruments do not perform a subsequent function to mitigate the effects of the transient.

Comparison to Screening Criteria:

1. The turbine overspeed protection system is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The turbine overspeed protection system is not used to monitor a process variable that is an initial condition of a DBA or transient.

3. The turbine overspeed protection system is not part of a primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 201) of NEDO-31466, the loss of the turbine overspeed protection system was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 2.

Conclusion:

Since the screening criteria have not been satisfied, the Turbine Overspeed Protection System LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.4.4 CHEMISTRY

LCO Statement:

The chemistry of the reactor coolant system shall be maintained within the limits specified in Table 3.4.4-1.

Discussion:

Poor reactor coolant water chemistry may contribute to the long term degradation of system materials and thus is not of immediate importance to the plant operator. Reactor coolant water chemistry is monitored for a variety of reasons. One reason is to reduce the possibility of failures in the reactor coolant system pressure boundary caused by corrosion. Severe chemistry transients have resulted in failure of thin walled LPRM instrument dry tubes in a relatively short period of time. However, these LPRM dry tube failures result in loss of the LPRM function and are readily detectable. In summary, the chemistry monitoring activity serves of a long term preventative rather than mitigative purpose.

Comparison to Screening Criteria:

1. Reactor coolant water chemistry is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Reactor coolant water chemistry is not used to monitor a process variable that is an initial condition of a DBA or transient.

3. Reactor coolant water chemistry is not supportive of any primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 211) of NEDO-31466, the reactor coolant water chemistry was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 2.

Conclusion:

Since the screening criteria have not been satisfied, the Reactor Coolant System Chemistry LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.4.8 STRUCTURAL INTEGRITY

LCO Statement:

The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.8.

Discussion:

The inspection programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity of these components will be maintained throughout the components life. Other Technical Specifications require important systems to be operable (for example, ECCS 3/4.5.1) and in a ready state for mitigative action. This Technical Specificationsis more directed toward prevention of component degradation and continued long term maintenance of acceptable structural conditions. Hence it is not necessary to retain this Specification to ensure immediate operability of safety systems.

Further, this Technical Specificationsprescribes inspection requirements which are performed during plant shutdown. It is, therefore, not directly important for responding to design basis accidents.

Comparison to Screening Criteria:

1. The inspections stipulated by this Specification are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The inspections stipulated by this Specification do not monitor process variables that are initial assumptions in a DBA or transient analyses.

3. The ASME Code Class 1, 2, and 3 components inspected per this Specification are assumed to function to mitigate a DBA. Their capability to perform this function is addressed by other Technical Specifications. This Technical Specification, however, only specifies inspection requirements for these components; and these inspections can only be performed when the plant is shutdown. Therefore, Criterion 3 is not satisfied.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 216) of NEDO-31466, the assurance of operability of the entire system as verified in the system operability Specification dominates the risk contribution of the system. As such, the lack of a long term assurance of structural integrity Specification was found to be a non-significant risk contributor to core damage frequency and offsite releases. Furthermore, the requirement is currently covered by 10 CFR 50.55a and the plant's Inservice Inspection Program. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 2.

Conclusion:

Since the screening criteria have not been satisfied, the Structural Integrity LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.7.5 SEALED SOURCE CONTAMINATION

LCO Statement:

Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of ≥ 0.005 microcuries of removable contamination.

Discussion:

The limitations on sealed source contamination are intended to ensure that the total body or individual organ irradiation doses does not exceed allowable limits in the event of ingestion or inhalation. This is done by imposing a maximum limitation of ≤ 0.005 microcuries of removable contamination on each sealed source. This requirement and the associated Surveillance Requirements bear no relation to the conditions or limitations which are necessary to ensure safe reactor operation.

Comparison to Screening Criteria:

1. Sealed source contamination is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Sealed source contamination is not a process variable that is an initial condition of a DBA or transient.

3. Sealed source contamination is not used in any part of a primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 267) of NEDO-31466, the sealed source contamination being not within limits was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 2.

Conclusion:

Since the screening criteria have not been satisfied, the Sealed Source Contamination LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.8.2.5 A.C. CIRCUITS INSIDE PRIMARY CONTAINMENT

LCO Statement:

The following A.C. circuits inside primary containment shall be de-energized.

Discussion:

The circuits involved in this LCO are kept normally de-energized and do not participate in plant safety actions. These circuits are primarily for lighting, utility outlets and convenient power plugs, to be used in the event of plant walkdowns, maintenance and in-situ test and/or observations. Therefore, they are of non-Class 1E nature.

They are properly separated from all other Class 1E circuits, and operation or failure of these non-Class 1E circuits do not impose any degradation on Class 1E circuits. Thus, in any event (energized or de-energized state), these circuits have no impact on plant safety systems.

Comparison to Screening Criteria:

1. The AC circuits listed in this Specification are de-energized during operation and are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The AC circuits listed in this Specification are not used to monitor a process variable that is an initial condition of a DBA or transient.

3. The AC circuits listed in this Specification are not part of a primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 275) of NEDO-31466, the AC circuits inside primary containment governed by this Specification were found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 2.

Conclusion:

Since the screening criteria have not been satisfied, the A.C. Circuits Inside Primary Containment LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.8.2.6 PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LCO Statement:

All primary containment penetration conductor overcurrent protective devices shown in Table 3.8.2.6-1 shall be OPERABLE.

Discussion:

The primary feature of these protective devices is to open the control and/or power circuit whenever the load conditions exceed the preset current demands. This is to protect the circuit conductors against damage or failure due to overcurrent heating effects. Primary and backup electrical protection for short circuits is provided for all penetrations.

The continuous monitoring of the operating status of the overcurrent protection devices is impracticable and not covered as part of the control room monitoring, except after trip condition indication.

In the event of failure of the primary protective device to trip the circuit, the backup protective device is expected to operate and isolate the faulty circuit. Thus, the upper level (back-up) protection will prevent loss of redundant power source. In the worst case fault condition, a single division of protective functions can be lost. However, this scenario is covered under single failure criterion.

The overcurrent protection devices ensure the pressure integrity of the containment penetration. With failure of the device it is postulated that the wire insulation will degrade resulting in a containment leak path during a LOCA. However, containment leakage is not a process variable and is not considered as part of the primary success path. Containment penetration degradation will be identified during the normal containment leak rate tests required by 10 CFR Part 50, Appendix J.

Comparison to Screening Criteria:

1. The primary containment penetration conductor overcurrent protection devices are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The primary containment penetration conductor overcurrent protection devices specific circuits are not used to monitor a process variable that is an initial condition of a DBA or transient.

3. The specific circuits of the primary containment penetration conductor overcurrent protection devices are not part of a primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 276) of NEDO-31466, the loss of the overcurrent protection function of the circuits associated with the primary containment penetration conductor overcurrent protection devices was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 2.

Conclusion:

Since the screening criteria have not been satisfied, the Primary Containment Penetration Conductor Overcurrent Protective Devices LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.9.6 COMMUNICATIONS

LCO Statement:

Direct communications shall be maintained between the control room and refueling platform personnel.

Discussion:

Communication between the control room and refueling floor personnel is maintained to ensure that refueling personnel can be promptly informed of significant changes in the plant status or core reactivity condition during refueling. The communications allow for coordination of activities that require interaction between the control room and refueling floor personnel (such as the insertion of a control rod prior to loading fuel). However, the refueling system design accident or transient response does not take credit for communications and is designed to ensure safe refueling operations.

Comparison to Screening Criteria:

1. Communications during any mode of plant operation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Communications during any mode of plant operation is not used to indicate status of, or monitor a process variable that is an initial condition of a DBA or transient.

3. Communication during any mode of plant operation does not contribute to a primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 286) of NEDO-31466, the loss of direct communication was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 2.

Conclusion:

Since the screening criteria have not been satisfied, the Communications LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.9.7 CRANE AND HOIST OPERABILITY

LCO Statement:

All cranes and hoists used for handling fuel assemblies or control rods within the reactor pressure vessel shall be OPERABLE.

Discussion:

Operability of the refueling platform equipment (crane, main hoist, fuel grapple, and auxiliary hoist) ensures that only the proper refueling platform equipment will be used to handle fuel within the reactor pressure vessel, hoists have sufficient load capacity for handling fuel assemblies and/or control rods, and the core internals and pressure vessel are protected from excessive lifting force if they are inadvertently engaged during lifting operations. Although the interlocks designed to provide the above capabilities can prevent damage to the refueling platform equipment and core internals, they are not assumed to function to mitigate the consequences of a design basis accident. Further, in analyzing the control rod withdrawal error during refueling, if any one of the operations involved in initial failure or error is followed by any other single equipment failure or single operator error, the necessary safety actions are taken (e.g., rod block or scram) automatically prior to violation of any limits. Hence the refueling platform interlocks are not part of the primary success path in mitigating the control rod withdrawal error during refueling.

Comparison to Screening Criteria:

1. The refueling platform and associated instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The refueling platform and associated instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient.

3. The refueling platform and associated instrumentation is not part of a primary success path in the mitigation of a DBA or transient.

As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 287) of NEDO-31466, the refueling platform and associated instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. GPC has reviewed this evaluation and considers it applicable to Plant Hatch Unit 2.

Conclusion:

Since the screening criteria have not been satisfied, the Crane and Hoist Operability LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.9.8 CRANE TRAVEL - SPENT FUEL STORAGE POOL

LCO Statement:

Loads in excess of 1600 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks.

Discussion:

The Technical Specificationslimit of 1600 pounds for loads over the spent fuel contained in the storage pool is further reduced by the heavy loads analysis to 725 pounds or the weight of a single fuel bundle. This 725 pound imposed limit ensures that in the event the load is dropped, the activity release will be bounded by the analysis of the refueling accident and any possible distortion of the fuel in the storage racks will not result in a critical array. Administrative monitoring of loads moving over the fuel storage racks serves as a backup to the Unit 1 crane interlocks. While the Unit 2 crane does not have interlocks, its use is strictly governed by administrative controls.

Although this Technical Specificationssupports the maximum refueling accident assumption in the DBA, the applicable fuel handling crane travel limits are not monitored and controlled during operation; they are checked on a periodic basis to ensure operability. The deterministic criteria for Technical Specificationsretention are, therefore, not satisfied.

Comparison to Screening Criteria:

1. The fuel handling crane travel limits are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The maximum severity assumed for the fuel handling DBA is limited by the limits placed on the crane travel. These crane travel limits are not, however, process variables monitored and controlled by the operator. They are interlocks and/or physical stops. Therefore, Criterion 2 is not satisfied.

3. The fuel handling crane travel limits are not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA.

Traditional PRAs do not review risks associated with the spent fuel storage pool. Design basis analyses indicate that the release associated with fuel assembly damage in the spent fuel storage pool due to crane accidents is significantly lower than releases of concern evaluated by PRAs.

Conclusion:

Since the screening criteria have not been satisfied, the Crane Travel - Spent Fuel Storage Pool LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

APPENDIX B

PLANT SPECIFIC

RISK SIGNIFICANT EVALUATIONS

APPENDIX B

TECHNICAL SPECIFICATION:

3/4.3.6.7.1 Reactor Vessel Water Level - Low Low Low (Level 1)
3/4.3.6.7.2 Drywell Pressure - High
3/4.3.6.7.4 Main Steam Line Flow - High
3/4.3.6.7.5 Refueling Floor Area Radiation - High

DESCRIPTION OF REQUIREMENT:

The instrumentation covered by the above specifications provides actuation signals to the Main Control Room Environmental Control System (MCRECS) to initiate the pressurization mode of control room ventilation in anticipation of a release.

REFERENCES:

Plant Hatch IPE
NUREG 1150 (Peach Bottom)
Monticello IPE

DISCUSSION:

All of the above signals are anticipatory, in that they initiate the pressurization mode of MCRECS prior to the actual presence of airborne radiation at the Control Room air inlet.

The above actuation signals for the MCRECS have no significance to the risk of core damage and negligible risk to the potential for significant offsite releases. Failure to initiate the pressurization mode of the MCRECS has no bearing on the ability of the operators to bring the plant to a safe shutdown condition. With regards to preventing offsite releases, failure of the above signals is unlikely to have any impact on the progression of events following a severe accident for the following reasons:

1. The MCR air inlet high radiation signal provides reliable automatic actuation for the system when it is realistically required.

2. Failure to pressurize the control room has a very indirect effect on the magnitude of a release. For pressurization to be realistically required, a release must already have occurred. PRAs do not typically model the very subtle impact that increased dose rates to operators might have on the progression of an accident. It is likely that the operators would continue to attempt to control the accident using available airborne radiation protection equipment, rather than abandoning the control room. The subtle impacts on human error rates are a negligible consideration when the operator actions at that time are primarily recovery actions anyway.

CONCLUSION:

Based on the above insights, it can be concluded that the above specifications may be relocated from the Plant Hatch Unit 2 Technical Specifications to Plant Hatch controlled documents with no significant impact on the potential for core damage and the risk of offsite consequences.

TABLE OF CONTENTS

1

(continued)