CP&L

Carolina Power & Light Company

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LYNN W. EURY
Senior Vice President
Operations Support

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 & 50-324/LICENSE NOS. DPR-71 & DPR-62
REQUEST FOR LICENSE AMENDMENT
PRIMARY CONTAINMENT ISOLATION SYSTEM

Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Parts 50.90 and 2.101, Carolina Power & Light Company (CP&L) hereby requests a revision to the Technical Specifications for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2.

The proposed changes revise TS Section 3/4.3.2 to clarify Instrument Tables 3.3.2-1, 3.3.2-2, 3.3.2-3 and 4.3.2-1, and revise TS Section 3/4.6.3 to more closely resemble the General Electric BWR/4 Standard Technical Specifications.

On September 29, 1987, CP&L requested an amendment to the BSEP-2 Technical Specifications relating to the main steam isolation valve setpoint change from reactor vessel water level - low, level 2 to low, level 3. This change is expected to be issued in the very near future and, thus, has been incorporated into this submittal as if it had already been approved. This action was taken to eliminate the need for an immediate update to this document when the September 29, 1987 request is issued.

Enclosure 1 provides a detailed description of the proposed changes, the basis for the changes, and the basis for the Company's determination that the proposed changes do not involve a significant hazards consideration.

Enclosure 2 provides instructions for incorporation of the proposed changes into the Technical Specifications for each unit.

Enclosures 3 and 4 provide the proposed Technical Specification pages for each unit.

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In accordance with the requirements of 10CFR170.12, a check for $150 is also enclosed.

In order to allow time for procedure revision and orderly incorporation into copies of the Technical Specifications, CP&L requests that the proposed amendments, once approved by the NRC, be issued with an effective date to be no later than 60 days from the issuance of the amendment.

Please refer any questions regarding this submittal to Mr. Leonard I. Loflin at (919) 836-6242.

Yours very truly,

L W Eury

L. W. Eury
Senior Vice President

bab/BAB(letter)

Enclosures:
1. Basis for Change Request
2. Instructions for Incorporation
3. Unit 1 Technical Specification Pages
4. Unit 2 Technical Specification Pages

cc: Mr. Dayne H. Brown
Dr. J. Nelson Grace
Mr. W. H. Ruland
Mr. E. D. Sylvester

L. W. Eury, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, contractors, and agents of Carolina Power & Light Company.

Ruby R. Morgan
Notary (Seal)

RUBY R. MORGAN
NOTARY
PUBLIC
WAKE COUNTY, N.C.

My commission expires: 11/27/89

ENCLOSURE 1

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2
NRC DOCKETS 50-325 & 50-324
OPERATING LICENSES DPR-71 & DPR-62
REQUEST FOR LICENSE AMENDMENT

BASIS FOR CHANGE REQUEST

Proposed Change Number 1

Delete reference to Table 3.6.3-1 on Page 1-5.

Basis

Technical Specification Section 3.6 addresses containment systems. Table 3.6.3-1 provides a partial listing of the automatic primary containment isolation valves (PCIVs). The Brunswick Plant Technical Specifications currently require the following surveillances:

Specification 4.6.3.1 requires that the primary containment isolation valves be demonstrated operable by cycling and isolation time testing following maintenance, repair, or replacement work.

Specification 4.6.3.2 requires verification of isolation valve operability through the application of an isolation test signal to verify the valve actuates properly. This testing is required every 18 months.

Specification 4.6.3.3 requires verification of the isolation time for power-operated and automatic valves.

Specification 4.6.4.4 requires that reactor instrumentation system isolation valves be demonstrated operable by cycling the valve every 18 months.

Specification 4.6.1.2.d requires Type B and C leak testing of primary containment isolation valves.

The Company has completed a review of the primary containment isolation system (PCIS) Technical Specifications and determined that certain changes to Table 3.6.3-1 are needed to accurately reflect the as-built design of the primary containment isolation system. However, rather than propose changes to the table, CP&L believes it is more appropriate to request removal of Table 3.6.3-1 from the Technical Specifications and replace references to the table with references to valve listings contained in plant procedures (the Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal). Our reasons for this determination are described below.

First, removal of the valve listings from the Technical Specifications is appropriate in the context of Technical Specification reform. The recent industry and NRC efforts to improve the Technical Specifications have identified component listings as items for removal from the Technical Specifications because they do not directly contribute to plant safety. In addition, the level of detail of such listings has resulted in frequent requests for license amendments. This process

has resulted in inherent inaccuracies in the Technical Specifications due to the delays in receiving license amendments brought about by the current regulatory environment. The NRC has already recommended and approved removal of the analogous snubber listing from the Technical Specifications. In addition, tables relating to fire protection devices and containment penetration protection circuit breakers have been removed from the Technical Specifications for other dockets. For example, listings relating to containment isolation valves, snubbers, containment penetration protection devices, motor operated valve thermal overload protection, the reactor vessel material surveillance program, engineered safety features actuation system instrumentation response times, and reactor trip system instrumentation response times have been removed from the Technical Specifications for the Shearon Harris Plant and references to plant procedures incorporated. The use of component listings through reference to plant procedures ensures that timely information is maintained that reflects recent plant modifications.

Second, changes to these plant procedures are strictly controlled under 10CFR50.59. In addition, the Technical Specifications requirement to maintain primary containment integrity (Technical Specification 3.6.1.1), as well as ASME Code, Section XI testing requirements (Technical Specification 4.0.5) and 10CFR50, Appendix J requirements (Technical Specification 3.6.1.2) are maintained. Only the listing of valves would be subject to licensee control under 10CFR50.59. Plant procedures are auditable and subject to inspection by the NRC.

Third, appropriate mechanisms and controls exist to ensure correct testing for the valves through plant procedures. As discussed above, these referenced procedure(s) will incorporate valve list information developed to meet the requirements of Technical Specifications 4.6.3.1, 4.6.3.2, 4.6.3.3 and 4.6.3.4. In addition, Appendix J of 10CFR50 (implemented through Technical Specification 3/4.6.1.2) specifies leak rate test requirements for containment isolation valves, and thus maintenance of a list of such valves. Section XI of the ASME Code (implemented through Technical Specification 4.0.5) requires once per cycle testing of valve closure and timing for certain of the valves currently listed in Table 3.6.3-1. In addition, Technical Specification 4.3.2.2 requires once per cycle performance of a logic system functional test for most of the valves currently listed in Table 3.6.3-1. All the above requirements are appropriately integrated into plant procedure(s) which are subject to change only under requirements of 10CFR50.59. These plant procedures cover testing (to or beyond the present Technical Specification requirements) for the valves currently included in Table 3.6.3-1. A complete listing of containment isolation valves and required closure times is also provided as Table 6.2.4-2 of the Brunswick Updated Final Safety Analysis Report. References to plant procedures will replace references to Table 3.6.3-1 in the text of the Technical Specifications. Specific closure time testing requirements for the main steam isolation valves will be maintained in the Technical Specifications under the existing Specification 3/4.4.7.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. Removal of the primary containment isolation valve listing from the Technical Specifications is appropriate in the context of Technical Specification reform. Recent industry and NRC efforts to improve the Technical Specifications have identified component listings as an item for removal because they do not directly contribute to plant safety. The use of component listings through references to plant procedures will ensure maintenance of timely information that reflects recent plant modifications. Changes to these procedures will be controlled under the requirements of 10CFR50.59. Primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specifications 3.6.1.1 and 3.6.3 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3 (and thus the proper identification, control, and periodic testing of the appropriate containment isolation valves). Relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not change their design or operation, nor does it affect any of the accident analyses; therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. As discussed in Item 1 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3 (and thus the proper identification, control, and periodic testing of the appropriate containment isolation valves). The design and operation of these valves does not change. Thus, relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. As discussed in Items 1 and 2 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3. Furthermore, the relocation of the primary containment isolation valve listing to plant procedures will clarify the Technical Specifications and ensure a more up to date and accurate primary containment isolation valve listing, thereby avoiding possible operator confusion. There is no change in the containment design or its pressure retaining capability. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Proposed Change Number 2

Replace reference to Specification 3.6.3.1 with a reference to Specification 3.6.3 on Page 1-5.

Basis

There is no Technical Specification 3.6.3.1 in the BSEP Technical Specifications. The Specification referenced here should be Specification 3.6.3. The proposed change makes this correction.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications. It replaces a reference to Technical Specification 3.6.3.1 with a reference to Technical Specification 3.6.3, which reflects the current reference number of the appropriate Technical Specification. It does not reflect any change in the design, operation, reliability, or testing requirements of the plant. Thus, it does not increase the probability of an accident, nor does it change the consequences of an accident.

2. The proposed change revises a reference to Technical Specification 3.6.3.1 to reflect the correct reference number, 3.6.3, of the Technical Specification. It is purely an administrative change; it does not reflect any physical change to the plant systems. Thus, it does not create the possibility of any new or different type of accident.

3. The proposed change is administrative, and being made to make the Technical Specifications more accurate. It does not involve a design change, nor does it reflect any change in the capability of the system or the demands placed on any process barrier. Thus, there is no change in the margin of safety.

Proposed Change Number 3

Delete the word "or" from the end of Item a.2 under PRIMARY CONTAINMENT INTEGRITY on Page 1-5.

Basis

Section 1.0 of the Technical Specifications defines what equipment must be operable for primary containment integrity to exist. The current Technical Specification addresses five requirements for containment integrity:

a. All penetrations required to be closed during conditions are either:

   1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification 3.6.3.1, or

b. All equipment hatches are closed and sealed.

c. Each containment air lock is OPERABLE pursuant to Specification 3.6.1.3.

d. The containment leakage rates are within the limits of Specification 3.6.1.2. and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings is) is OPERABLE.

The wording of Item a.2 currently contains an administrative error. The conjunction "or" is attached to the end of the paragraph. It is unnecessary and grammatically out of place and should be deleted.

This error has apparently been in place since the original Technical Specifications were issued. There is no technical basis for the current wording. The conjunction "or" does not serve to represent any logic; there are clearly two choices under Item a, with an "or" conjunction between the two choices. All five main items, a,b,c,d, and e, must be true for containment integrity to exist; thus, the "or" after Item a.2 is inappropriate. Therefore, the proposed change is administrative and does not impact any accident analyses or plant equipment.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications. The conjunction "or" does not imply any sort of logic; all five of the main items, a, b, c, d, and e, must be true for primary containment integrity to exist. The proposed change does not change the design, operation, reliability, or testing requirements of plant systems. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It does not change the design, operation, reliability, or testing requirements of plant systems. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 4

Delete Valve Group 7 from Item 1.a.1, "Reactor Vessel Water Level - Low, Level 1," in Table 3.3.2-1 under Primary Containment Isolation on Page 3/4 3-11.

Basis

Currently, Technical Specification Table 3.6.3-1 identifies the reactor vessel head spray, residual heat removal (RHR) shutdown cooling supply, and RHR injection isolation valves as Group 8 valves. Table 3.3.2-1 indicates Group 8 valves isolate on reactor vessel water level - low, level 1 (LL1) and reactor steam dome pressure - high.

Table 3.3.2-1 also lists Group 7 valves as having isolation signals from reactor vessel water level - low, level 1, drywell pressure - high, and reactor steam dome pressure. However, Table 3.6.3-1 does not identify any valves as Group 7 valves.

The proposed change deletes the reference to Valve Group 7 under "Valve Groups Operated By Signal" in Item 1.a.1 of Table 3.3.2-1. The following factors contributed to the confusion associated with creation of a Valve Group 7.

CUSTOM TO STANDARD TECHNICAL SPECIFICATION CONVERSION

In 1977, BSEP converted from the custom Technical Specification format to the standard format. The Standard Technical Specifications (STS) were requested for both units on August 22, 1977, and issued on November 23, 1977.

As part of this conversion, several changes were made to the valve groups to more completely and consistently describe the isolation signals. Valve Group 2 originally contained those valves which were assumed to isolate during a loss of coolant accident but were not located on high energy lines with their own leak detection. The isolation signals associated with these valves were LL1 and drywell pressure - high. There were also several subgroups within Group 2 that isolated on additional signals. In the conversion from custom to STS, new valve groups were created to recognize these distinctions. Group 6 was created for those valves originally in Group 2 that also isolated on reactor building vent high radiation. Group 8 was created for those valves originally in Group 2 associated with the RHR system that isolate on LL1 and reactor vessel steam dome pressure - high.

LOGIC DESIGN CHANGES

In 1975, General Electric issued Service Information Letter 131, which recommended several changes to the containment isolation logic. Experience had shown that the normal water level drops that occurred during scrams due to collapse of voids exceeded the LL1 setpoint. This caused unnecessary valve group isolations.

Three changes were recommended for BSEP to reduce the complexity of scram recoveries and to reduce demands on the operators caused by unnecessary system isolations and transients so they could concentrate on the scram itself and be more observant for unusual occurrences. These changes were:

1. Lower reactor water cleanup system isolation from LL1 to reactor vessel water level - low, level 2 (LL2),

2. Lower secondary containment isolation and standby gas treatment system initiation from LL1 to LL2,

3. Delete the drywell pressure - high isolation signals for RHR suction and discharge valves.

Plant modification and Technical Specification change packages were prepared to support these logic changes. The original amendment request for BSEP-2 was made on February 2, 1976. The BSEP-2 amendment request was submitted in custom format. On June 22, 1977, a submittal was made that indicated that the same changes were applicable to BSEP-1 as well, and requested that the original change be amended to include BSEP-1. The amendment for BSEP-2 was issued on October 12, 1977. The amendment for BSEP-1 was issued in STS format on April 4, 1979, after conversion to STS format.

Both requests associated with the logic change requested that the drywell pressure - high signal be deleted as an RHR isolation signal. However, the Technical Specification pages originally submitted with the BSEP-2 submittal failed to indicate that the drywell pressure - high signal was also to be deleted for the RHR head spray valves (E11-F022 and E11-F023). This change was encompassed in the logic change; however, the Technical Specification pages were not changed to reflect this.

Apparently, this failure to specifically identify the RHR reactor head spray valves as no longer having the drywell pressure - high signal associated with them, led those preparing the BSEP-2 custom to STS conversion package to assume that they should be treated differently (i.e., no longer as Group 2 valves, but not as Group 8 valves either, as they should have been). The isolation signals associated with the Group 7 valves were assumed to be LL1, reactor steam dome pressure - high and drywell pressure - high. Thus, when Valve Groups 6 and 8 were created in the BSEP-2 STS, Valve Group 7 was created specifically for the RHR reactor head spray valves solely because they were thought to still isolate on drywell pressure - high.

Valve Group 7 was created in Table 3.6.3-1 only for BSEP-2 in the original issuance of the STS. The BSEP-2 Technical Specifications now listed a Valve Group 7 for the E11-F022 and E11-F023 valves in Table 3.6.3-1, and Group 7 as a valve group operated by the LL1, reactor steam dome pressure - high and drywell pressure - high isolation signals in Table 3.2.3-1. When the BSEP-1 STS were issued, there was no reference to Group 7 listed anywhere because the logic modification had not yet been performed.

The BSEP-1 Technical Specification change associated with the logic change was developed from the initial logic change request for BSEP-2. As a result, the BSEP-1 amendment, which was an amendment to the STS, included the Group 7 references in Table 3.2.3-1 as a valve group operated by the LL1, reactor steam

dome pressure - high and drywell pressure - high signals, but did not identify Valve Group 7 and its associated valves in Table 3.6.3-1.

On September 7, 1982, an additional amendment request was submitted which stated that a typographical error had been made in the classification of the RHR head spray valves. The valves were not actually Group 7; they were to be Group 8 valves. This amendment was issued on March 6, 1984 and changed the classification of the valves in Table 3.6.3-1 from Group 7 to Group 8. This actually applied only to BSEP-2, since BSEP-1 never identified a Group 7 in Table 3.6.3-1; however, the amendment was issued for both units. This amendment request did not delete the superfluous references to Valve Group 7 in Table 3.2.3-1. Thus, these references need to be deleted.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change deletes references to a valve group that currently does not exist, and was never intended to exist. The valves once considered Group 7 valves, E11-F022 and E11-F023, are actually Group 8 valves, and isolate on the signals which isolate the Group 8 valves. There are no valves associated with Valve Group 7, and it has been determined that no valves should be associated with the Valve Group 7 currently listed in this item. Since Group 7 does not currently provide any means of protection, this change does not affect the probability or consequences of an accident.

2. Valve Group 7 currently does not include any valves. Therefore, it can perform no safety function, nor hamper any existing safety function. Thus, its deletion does not create the possibility of a new or different kind of accident.

3. Valve Group 7 cannot provide any means of protection, because it does not currently include any valves. Its deletion does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, the margin of safety remains unchanged.

Proposed Change Number 5

Place the operability requirements for Valve Group 8 on a separate line in Item 1.a.1, "Reactor Vessel Water Level - Low, Level 1," in Table 3.3.2-1 under Primary Containment Isolation on Page 3/4 3-11 and specify the correct requirements for the valve group.

Basis

Currently, Item 1.a.1 specifies the same operability requirements for Valve Groups 2, 6, 7, and 8. Deletion of Valve Group 7 is being addressed in Proposed Change Number 4. The existing operability requirements apply only to Groups 2 and 6, which isolate the drywell equipment and floor drains and drywell/suppression chamber purge and vent valves. Action Statement 20, "Be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours," applies to Valve Groups 2 and 6, while Action 27, "Deactivate the shutdown cooling supply and reactor vessel head spray isolation valves in the closed position until the reactor steam dome pressure is within the specified limits," applies to the Group 8 valves. The Group 8 valves isolate the reactor vessel head spray, residual heat removal (RHR) shutdown cooling supply, and RHR injection isolation valves on a signal from the reactor vessel water level - low, level 1 instrumentation.

The proposed change places the Valve Group 8 information on a separate line with its own action statement (Action 27) under Item 1.a.1. This makes the information provided more complete and accurate. No change has been made to the design or operation of the instrumentation or equipment; only the way it is referenced in Item 1.a.1 will be changed.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change provides additional information in Item 1.a.1 to make it a more complete and accurate representation of the existing equipment and instrumentation. The logic associated with the instrumentation has not changed, nor has it design or operation. The proposed change references the appropriate operational conditions for Valve Group 8 under Item 1.a.1. The same operability requirements are listed for Group 8 in Item 5.a under shutdown cooling. Because this change provides additional information which will present a more accurate description of the existing instrumentation, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of any accident previously evaluated.

2. The proposed change will provide a more complete and accurate description of existing equipment and instrumentation. It does not represent a change in logic, nor does it reflect a change in the instrument design or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change has no effect on the design or operation of the equipment represented. It provides a more appropriate representation of existing equipment. Therefore, there is no impact on the margin of safety.

Proposed Change Number 6

Place the operability requirements for Valve Group 3 on a separate line in Item 1.a.2, "Reactor Vessel Water Level - Low, Level 2," in Table 3.3.2-1 under Primary Containment Isolation on Page 3.4 3-11 and specify the correct requirements for the valve group (BSEP-1 only).

Basis

Currently, Item 1.a.2 specifies the same operability requirements for Valve Groups 1 and 3. The existing operability requirements apply only to Group 1, which isolates the main steam line isolation, main steam line drain isolation, and reactor water sample line isolation valves. Action Statement 20, "Be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours," applies to Valve Group 1, while Action Statement 24, "Isolate the reactor water cleanup system," applies to the Group 3 valves. The Group 3 valves isolate the reactor water cleanup system on a signal from the reactor vessel water level - low, level 2 instrumentation.

The proposed change places the Valve Group 3 information on a separate line with its own action statement (Action 24) under Item 1.a.2. This makes the information provided more complete and accurate. No change has been made to the design or operation of the instrumentation or equipment; only the way it is referenced in Item 1.a.2 will be changed.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change provides additional information in Item 1.a.2 to make it a more complete and accurate representation of the existing equipment and instrumentation. The logic associated with the instrumentation has not changed, nor has its design or operation. The proposed change references the appropriate operational conditions for Valve Group 3 under Item 1.a.2. The same operability requirements are listed for Group 3 in Item 3.e under reactor water cleanup system isolation. Because this change provides additional information which will present a more accurate description of the existing instrumentation, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of any accident previously evaluated.

2. The proposed change will provide a more complete and accurate description of existing equipment and instrumentation. It does not represent a change in logic, nor does it reflect a change in the instrument design or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change has no effect on the design or operation of the equipment represented. It provides a more appropriate representation of existing equipment. Therefore, there is no impact on the margin of safety.

Proposed Change Number 7

Replace "and" with ";" in the instrument number listings in Item 1.a.2, "Reactor Vessel Water Level - Low, Level 2," in Table 3.3.2-1 under Primary Containment Isolation on Page 3/4 3-11 (BSEP-1 only).

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of any accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table, and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process function. Therefore, there is no impact on the margin of safety.

Proposed Change Number 8

Delete Item 1.a.2, "Reactor Vessel Water Level - Low, Level 2," from Table 3.3.2-1 on Page 3/4 3-11 and re-label Item 1.a.3 as Item 1.a.2 (BSEP-2 only).

Basis

The Technical Specification change request submitted on September 29, 1987, as supplemented on October 14, 1987 and November 24, 1987, revised the reactor vessel water level trip function for the Valve Group 1 isolation valves from reactor vessel water level - low, level 2 to reactor vessel water level - low, level 3 This resulted in only Valve Group 3 being actuated by the reactor vessel water level - low, level 2 trip function. Valve Group 3 isolates the reactor water cleanup system and is addressed specifically in Item 3.e for the reactor vessel water level - low level 2 instrumentation. The proposed change deletes Item 1.a.2 because the instrumentation no longer actuates any valve groups that need to be addressed under Item 1. This change does not represent any physical change to the design or operation of any systems. It only more accurately describes the trip function associated with the Group 3 valves.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change deletes information which no longer belongs under Item 1. Via a previous submittal, the trip function for the Group 1 isolation valves was revised from reactor vessel water level - low, level 2 to reactor vessel water level - low, level 3. The information associated with the remaining valve group actuated by that trip function, Valve Group 3, is more appropriately referenced in Item 3.e, which describes the instrumentation that actuates reactor water cleanup system isolation. The proposed change does not reflect a change to the design or operation of the instrumentation and valve groups; it only clarifies existing information in the table. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change clarifies the table by placing the information associated with the reactor vessel water level - low, level 2 instrumentation in its appropriate place. It does not reflect a change in the design or operation of the system. Therefore, it does not create the possibility of a new or different accident.

3. The proposed change does not reflect a change to the design or operation of any equipment. It merely provides the applicable references to the reactor vessel water level - low, level 2 instrumentation in the appropriate place in the table. Therefore, there is no impact on the margin of safety.

Proposed Change Number 9

Delete Valve Group 7 from Item 1.b, "Drywell Pressure - High," in Table 3.3.2-1 under Primary Containment Isolation on Page 3/4 3-11.

Basis

Currently, Technical Specification Table 3.6.3-1 identifies the reactor vessel head spray, residual heat removal (RHR) shutdown cooling supply, and RHR injection isolation valves as Group 8 valves. Table 3.3.2-1 indicates Group 8 valves isolate on reactor vessel water level - low, level 1 (LL1) and reactor steam dome pressure - high.

Table 3.3.2-1 also lists Group 7 valves as having isolation signals from reactor vessel water level - low, level 1, drywell pressure - high, and reactor steam dome pressure. However, Table 3.6.3-1 does not identify any valves as Group 7 valves.

The proposed change deletes the reference to Valve Group 7 under "Valve Groups Operated By Signal" in Item 1.b of Table 3.3.2-1. The following factors contributed to the confusion associated with creation of a Valve Group 7.

CUSTOM TO STANDARD TECHNICAL SPECIFICATION CONVERSION

In 1977, BSEP converted from the custom Technical Specification format to the standard format. The Standard Technical Specifications (STS) were requested for both units on August 22, 1977, and issued on November 23, 1977.

As part of this conversion, several changes were made to the valve groups to more completely and consistently describe the isolation signals. Valve Group 2 originally contained those valves which were assumed to isolate during a loss of coolant accident but were not located on high energy lines with their own leak detection. The isolation signals associated with these valves were LL1 and drywell pressure - high. There were also several subgroups within Group 2 that isolated on additional signals. In the conversion from custom to STS, new valve groups were created to recognize these distinctions. Group 6 was created for those valves originally in Group 2 that also isolated on reactor building vent high radiation. Group 8 was created for those valves originally in Group 2 associated with the RHR system that isolate on LL1 and reactor vessel steam dome pressure - high.

LOGIC DESIGN CHANGES

In 1975, General Electric issued Service Information Letter 131, which recommended several changes to the containment isolation logic. Experience had shown that the normal water level drops that occurred during scrams due to collapse of voids exceeded the LL1 setpoint. This caused unnecessary valve group isolations.

Three changes were recommended for BSEP to reduce the complexity of scram recoveries and to reduce demands on the operators caused by unnecessary system isolations and transients so they could concentrate on the scram itself and be more observant for unusual occurrences. These changes were:

1. Lower reactor water cleanup system isolation from LL1 to reactor vessel water level - low, level 2 (LL2),

2. Lower secondary containment isolation and standby gas treatment system initiation from LL1 to LL2,

3. Delete the drywell pressure - high isolation signals for RHR suction and discharge valves.

Plant modification and Technical Specification change packages were prepared to support these logic changes. The original amendment request for BSEP-2 was made on February 2, 1976. The BSEP-2 amendment request was submitted in custom format. On June 22, 1977, a submittal was made that indicated that the same changes were applicable to BSEP-1 as well, and requested that the original change be amended to include BSEP-1. The amendment for BSEP-2 was issued on October 12, 1977. The amendment for BSEP-1 was issued in STS format on April 4, 1979, after conversion to STS format.

Both requests associated with the logic change requested that the drywell pressure - high signal be deleted as an RHR isolation signal. However, the Technical Specification pages originally submitted with the BSEP-2 submittal failed to indicate that the drywell pressure - high signal was also to be deleted for the RHR head spray valves (E11-F022 and E11-F023). This change was encompassed in the logic change; however, the Technical Specification pages were not changed to reflect this.

Apparently, this failure to specifically identify the RHR reactor head spray valves as no longer having the drywell pressure - high signal associated with them, led those preparing the BSEP-2 custom to STS conversion package to assume that they should be treated differently (i.e., no longer as Group 2 valves, but not as Group 8 valves either, as they should have been). The isolation signals associated with the Group 7 valves were assumed to be LL1, reactor steam dome pressure - high and drywell pressure - high. Thus, when Valve Groups 6 and 8 were created in the BSEP-2 STS, Valve Group 7 was created specifically for the RHR reactor head spray valves solely because they were thought to still isolate on drywell pressure - high.

Valve Group 7 was created in Table 3.6.3-1 only for BSEP-2 in the original issuance of the STS. The BSEP-2 Technical Specifications now listed a Valve Group 7 for the E11-F022 and E11-F023 valves in Table 3.6.3-1, and Group 7 as a valve group operated by the LL1, reactor steam dome pressure - high and drywell pressure - high isolation signals in Table 3.2.3-1. When the BSEP-1 STS were issued, there was no reference to Group 7 listed anywhere because the logic modification had not yet been performed.

The BSEP-1 Technical Specification change associated with the logic change was developed from the initial logic change request for BSEP-2. As a result, the BSEP-1 amendment, which was an amendment to the STS, included the Group 7 references in Table 3.2.3-1 as a valve group operated by the LL1, reactor steam

dome pressure - high and drywell pressure - high signals, but did not identify Valve Group 7 and its associated valves in Table 3.6.3-1.

On September 7, 1982, an additional amendment request was submitted which stated that a typographical error had been made in the classification of the RHR head spray valves. The valves were not actually Group 7; they were to be Group 8 valves. This amendment was issued on March 6, 1984 and changed the classification of the valves in Table 3.6.3-1 from Group 7 to Group 8. This actually applied only to BSEP-2, since BSEP-1 never identified a Group 7 in Table 3.6.3-1; however, the amendment was issued for both units. This amendment request <u>did not</u> delete the superfluous references to Valve Group 7 in Table 3.2.3-1. Thus, these references need to be deleted.

<u>10CFR50.92 Evaluation</u>

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change deletes references to a valve group that currently does not exist, and was never intended to exist. The valves once considered Group 7 valves, E11-F022 and E11-F023, are actually Group 8 valves, and isolate on the signals which isolate the Group 8 Valves. There are no valves associated with Valve Group 7, and it has been determined that no valves should be associated with the current Valve Group 7. Since Group 7 does not currently provide any means of protection, this change does not affect the probability or consequences of an accident.

2. Valve Group 7 currently does not include any valves. Therefore, it can perform no safety function, nor hamper any existing safety function. Thus, its deletion does not create the possibility of a new or different kind of accident.

3. Valve Group 7 cannot provide any means of protection, because it does not currently include any valves. Its deletion does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, the margin of safety remains unchanged.

Proposed Change Number 10

Add related Instrument Number D12-RE-N006A,B,C,D to Item 1.c.1, "Main Steam Line Radiation - High," in Table 3.3.2-1 under Primary Containment Isolation on Page 3/4 3 11.

Basis

The main steam line radiation - high instrumentation detects radioactivity that may be released as a result of a control rod drop accident and provides isolation signals to the main steam isolation and drain isolation valves. This limits the amount of radioactivity released from containment.

The current Technical Specifications reference only the instrument number for the radiation monitor drawer located in the control room. It does not list the associated radiation detectors which are located in the main steam isolation valve (MSIV) pit. The proposed change adds Instrument Number D12-RE-N006A,B,C,D, which represents the radiation detectors, to Item 1.c.1 in Table 3.3.2-1. These instruments are not being physically changed or added to the instrument loop; they currently exist, but are not referenced in the appropriate place in the table. By adding these instruments to the list, the instrument loop is more clearly and completely identified in the Technical Specifications. This may reduce the possibility of confusion concerning which instrument channel is subject to the requirements of Item 1.c.1.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change adds a reference to Instrument Number D12-RE-N006A,B,C,D, which represents the radiation detectors located in the MSIV pit, in Item 1.c.1 so the appropriate devices are clearly identified in the Technical Specifications in a format consistent with other plant documents and programs. It does not change the devices which perform the described function, nor does it change their ability to mitigate accidents. It simply references them more completely and correctly. This change will reduce the possibility of misunderstanding about which devices are associated with the isolation function, thereby potentially reducing errors. Therefore, it does not impact the probability or consequences of any accident previously evaluated.

2. The proposed change completes the instrument number references in Item 1.c.1 so the appropriate devices are clearly identified in the Technical Specifications in a format consistent with other plant documents and programs. The instrumentation performs the same function as before for mitigating and detecting accident conditions. It is merely being referenced more completely and correctly. This change will reduce the potential for misunderstanding about which devices are associated with the isolation function, thereby reducing the potential for errors. Therefore, no new accident possibilities are created.

3. The proposed change adds an instrument number reference so the appropriate devices are completely and clearly identified in the Technical Specifications in a format consistent with other plant documents and programs. It does not change the response, capability, reliability, or testing requirements of the instrumentation, nor does it change the ability of the instrumentation to mitigate or detect accident conditions. This change will reduce the possibility of misunderstanding about which devices are associated with the isolation function, thereby reducing potential errors. It does not change the design or capability of the primary containment system, nor does is change the demands placed upon this process barrier. Therefore, there is no effect on the margin of safety.

Proposed Change Number 11

Delete Footnote (h) from under the Applicable Operational Condition column in Item 1.c.1, "Main Steam Line Radiation - High," in Table 3.3.2-1 under Primary Containment Isolation on Page 3/4 3-11 (BSEP-2 only).

Basis

Footnote (h) was added to the Technical Specifications via Amendment 131 on December 10, 1986 to support the hydrogen injection test which took place in January, 1987. This was a one-time test; therefore, the footnote is no longer applicable or necessary for normal operation.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change deletes a footnote which no longer applies. The footnote was added to support a one-time, hydrogen injection test, which was completed on January 5, 1987. No additional testing is planned. Thus, the proposed change has no effect on the probability of an accident, nor does it affect the consequences of any accidents.

2. The referenced footnote no longer applies to BSEP-2. The hydrogen injection test was successfully completed on January 5, 1987. Thus, this footnote is no longer necessary, and deletion of it will not create the possibility of a new or different type of accident.

3. Footnote (h) was added to support a one-time hydrogen injection test, which was completed on January 5, 1987. No additional testing is planned; therefore, the footnote no longer applies and should be deleted. This deletion has no impact on the margin of safety.

Proposed Change Number 12

Add Footnote (i) under Valve Groups in Items 1.c.2, 1.c.3, 1.c.4 (BSEP-2 only), 1.d, 1.e, and 1.f in Table 3.3.2-1 under Primary Containment Isolation on Pages 3/4 3-11 and 3/4 3-12.

Basis

The reactor water sample line valves, B32-F019 and B32-F020, isolate on reactor vessel water level - low and main steam line radiation - high. The remaining Group 1 isolation signals have no bearing on the reactor water sample line; they would not detect a reactor water sample line rupture, nor would isolation of the reactor water sample line serve to mitigate or prevent a main steam line break outside containment. This is because of the small diameter of the sample line, and the fact that it is routed separately from the main steam lines.

The current Technical Specifications incorrectly indicate that the B32-F019 and B32-F020 reactor water sample isolation valves isolate automatically on all of the Group 1 isolation signals. It should show that these valves isolate only on reactor water level - low or main steam line radiation - high.

This change adds a reference to a new Footnote (i) to Items 1.c.2, 1.c.3, 1.c.4 (BSEP-2 only), 1.d, 1.e, and 1.f. The footnote states "Does not isolate B32-F019 or B32-F020." This would correct the current information which implies that the B32-F019 and B32-F020 valves isolate on main steam line pressure - low, main steam line flow - high, main steam line tunnel temperature - high, condenser vacuum - low, and turbine building area temperature - high.

This change does not represent any physical changes to the instruments themselves; it only provides a more adequate description of the existing instrumentation and its function.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The Group 1 isolation signals function to prevent, detect and/or mitigate a main steam line break outside containment. The only signals which serve to isolate the reactor water sample line are the reactor water level - low or main steam line radiation - high. The remaining Group 1 signals would not detect a need to isolate the reactor water sample line, nor would isolation of this line mitigate or prevent a main steam line break outside containment.

   The proposed change does not represent a change in instrumentation or logic; it merely provides a more adequate description of existing instrumentation and its function. The existing design of the system is more clearly defined by referencing the Footnote (i), thus, reducing confusion about the operation of the system. It is sufficient for accomplishing the required safety function. Therefore, there is no change in the probability of occurrence or the consequences of a pipe break outside containment, which is the only accident to which this change would apply.

2. Failure of the reactor water sample line is bounded by the main steam line break accident. The Group 1 signals, with the exception of the reactor vessel water level - low and main steam line radiation - high signals, would not detect and are not needed to isolate the reactor water sample line when necessary. No instrumentation is being changed; it is merely being referenced in a more accurate manner. Therefore, the possibility of a new accident is not created by this change.

3. The Group 1 isolation signals, except for reactor water level - low and main steam line radiation - high, would not detect the need to isolate the reactor water sample line and isolation of the line would not mitigate the consequences of a main steam line break. These signals do not cause isolation of the reactor water sample line, and never have. Therefore, these isolation signals, when associated with Valves B32-F019 and B32-F020, do not accomplish any safety function. Thus, they do not contribute to the margin of safety and their elimination from the Technical Specifications would not change the margin of safety. The automatic isolation of the reactor water sample line on the existing two trips are the ones which are necessary and sufficient to ensure isolation of Valves B32-F019 and B32-F020 when required.

Proposed Change Number 13

Add new Item 1.g, "Reactor Building Exhaust Radiation - High," and associated Instrument Numbers D12-RE-N010A,B and D12-RM-K609A,B to Table 3.3.2-1 under Primary Containment Isolation on Page 3/4 3-12.

Basis

The reactor building exhaust radiation - high isolation signal causes the Group 6 isolation valves, which include the containment atmospheric control valves and certain containment atmospheric monitoring and post-accident sampling system valves, to close during a loss of coolant accident (LOCA). This minimizes the amount of radiation released from primary containment under LOCA conditions. This signal is not of primary importance for Group 6 isolation; the reactor vessel water level - low, level 2 and drywell pressure - high signals provide Group 6 isolation signals much earlier in a LOCA scenario. These two signals directly detect a LOCA, while the reactor building exhaust radiation - high instrumentation detects radiation released from primary containment to the reactor building. It does not directly detect a LOCA and will isolate Group 6 much later than the other two signals will.

Item 1 of Table 3.3.2-1, "Primary Containment Isolation," provides information concerning isolation of the main steam lines, drywell drains, and the containment atmospheric control and monitoring systems. The main steam lines are isolated by the Group 1 valves, the drywell drains and associated systems by the Group 2 valves, and the containment atmospheric control and monitoring systems by the Group 6 valves.

The new Item 1.g provides information relating to the reactor building exhaust radiation - high instrumentation with respect to the Group 6 isolation valves. This instrumentation consists of two channels; one per trip system, either of which can initiate a Group 6 isolation. The radiation monitors listed are located in the reactor building exhaust plenum and monitor normal HVAC effluent from secondary containment in the reactor building.

Item 1.g provides the appropriate operability and surveillance requirements for isolation of the Group 6 valves on a signal from the reactor building exhaust radiation - high instrumentation. These requirements are the same as those listed for the other Group 6 signals specified under Item 1 since they perform the same safety function.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes specific operability and surveillance requirements for instrumentation which is designed to detect accident conditions, not prevent an accident. These requirements are consistent with the accident analysis assumptions and are similar to those for other instrumentation in this group. There is no change in how the equipment is operated. This change establishes operability and surveillance requirements for existing equipment and does not change the design or affect the operation of the instrumentation. Thus, there is no increase in the probability of an accident.

   The proposed change adds operability and surveillance requirements which are appropriate for the Group 6 isolation valves and their related safety function. The existing requirements in the Technical Specifications are listed under the Secondary Containment section, and are not directly applicable to the Group 6 valves. The reactor building exhaust radiation - high signal is not depended upon for Group 6 isolation for any design basis accident; thus, there is no impact on the consequences of any accident.

2. Specifying the correct operability and surveillance requirements provides additional assurance that the instrumentation will perform its intended safety function. It does not change the design or operation of any equipment or cause it to operate in a manner not previously evaluated. It does not affect any equipment which could cause an accident. It only affects existing instrumentation designed to detect accident conditions. Therefore, it does not create the possibility of a new accident.

3. Specifying the correct operability and surveillance requirements does not affect any equipment which could cause an accident. It only affects instrumentation designed to detect radioactivity in secondary containment and is not depended upon for mitigating any design basis accident. Therefore, it is not a factor in the margin of safety for any design basis accident. The instrumentation is designed to detect and mitigate the design basis accidents is not affected by this change and will have the same reliability and response characteristics.

Proposed Change Number 14

Add reference to Footnote (k), which states, "Secondary containment isolation dampers as listed in plant procedure _______," with its own operability requirements under the valve group column in Item 2.a, "Reactor Building Exhaust Radiation - High," in Table 3.3.2-1 under Secondary Containment Isolation on Page 3/4 3-13.

Basis

A signal from the reactor building exhaust radiation - high instrumentation closes the secondary containment isolation dampers, thereby establishing secondary containment isolation and minimizing the release of radiation from the reactor building during an accident. The dampers work in conjunction with the standby gas treatment system to establish a negative pressure in the reactor building and a controlled vent path through the standby gas treatment system filters and the stack. Secondary containment isolation would occur during a LOCA due to low reactor vessel water level or drywell pressure - high. This would minimize the release of radioactivity from the reactor building. Secondary containment isolation is also assumed to occur during a refueling accident due to a signal from the reactor building exhaust radiation - high instrumentation to contain radiation released in a fuel handling accident.

The current Technical Specifications reference Valve Group 6 under Item 2.a but do not reference the dampers which in fact establish secondary containment isolation. The proposed change adds a reference to Footnote (k) to the current references provided in Item 2.a to provide the needed reference to the secondary containment isolation dampers, which isolate secondary containment. The appropriate action statement associated with the secondary containment dampers is Action Statement 23, which states, "Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour," while the appropriate action statement for Valve Group 6 is Action Statement 20, "Be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours," as is also shown in Item 1.g. Thus, the two items are listed on separate lines under Item 2.a.

Footnote (k) states, "Secondary containment isolation dampers as listed in plant procedure ________." The secondary containment isolation dampers are currently listed in Table 3.6.5.2-1 of the Technical Specifications. This table is being deleted, and the information relocated to a plant procedure, as described elsewhere in this submittal.

Addition of the footnote reference is necessary because there is no reference to the secondary containment dampers in Item 2.a. This change will simplify the Technical Specifications, eliminate discrepancies, and reduce the potential for misinterpretation. The procedure which will list the secondary containment isolation dampers once they are removed from the Technical Specifications is currently being developed. The Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal.

10CFR50.92 Evaluation

This change does not involve a significant hazards consideration for the following reasons:

1. The proposed change provides a more accurate reference to the appropriate valves covered by Item 2.a under Secondary Containment Isolation. The current Technical Specifications list a valve group which is isolated by a signal from the reactor building exhaust radiation - high instrumentation, but does not establish secondary containment isolation. The dampers which establish secondary containment isolation upon receipt of a signal from the reactor building exhaust radiation - high instrumentation will be listed in plant procedure _____.

   There is no change in the design, operation, reliability, or testing requirements of existing plant systems. This change provides a more complete reference to existing plant equipment in its appropriate place in the table. Thus, there is no change in the probability of occurrence of an accident, nor is there any change in the consequences of any accident previously evaluated.

2. The proposed change provides a more accurate reference to the appropriate valves covered by Item 2.a. The current Technical Specifications list a valve group which is isolated by a signal from the reactor building exhaust radiation - high instrumentation, but does not establish secondary containment isolation. The dampers which establish secondary containment isolation upon receipt of a signal from the reactor building exhaust radiation - high instrumentation will be listed in a plant procedure which will be specifically identified at a later date.

   There is no change in the design, operation, reliability, or testing requirements of existing plant systems. This change provides a more complete reference to existing plant equipment in an appropriate place in the plant procedures. Thus, no new or different accident situations are created.

3. The proposed change provides references to the appropriate valves covered by Item 2.a. This change adds a reference to the dampers which actually provide secondary containment isolation. There is no change in the response, capability, reliability, or testing requirements of plant systems. It does not change the design or capability of the primary containment, nor does it change the demands place upon this process barrier. Thus, there is no effect on the margin of safety.

Proposed Change Number 15

Place the operability requirements for Valve Group 6 on a separate line in Item 2.a, "Reactor Building Exhaust Radiation - High," in Table 3.3.2-1 under Secondary Containment Isolation on Page 3/4 3-13.

Basis

Currently, Item 2.a specifies operability requirements for Valve Group 6. The existing operability requirements apply only to the secondary containment isolation dampers, which are addressed in Proposed Change Number 14. Action Statement 20, "Be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours," applies to Valve Group 6, while Action 23, "Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour," applies to the secondary containment isolation dampers.

The proposed change places the Valve Group 6 information on a separate line with its own action statement (Action 20) under Item 2.a. This makes the information provided more complete and accurate. No change has been made to the design or operation of the instrumentation or equipment; only the way it is referenced in Item 2.a will be changed.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change provides additional information in Item 2.a to make it a more complete and accurate representation of the existing equipment and instrumentation. The logic associated with the instrumentation has not changed, nor has it design or operation. The proposed change references the appropriate operational conditions for Valve Group 6 under Item 2.a. The same operability requirements are listed for Group 6 in Item 1.g under primary containment isolation. Because this change provides additional information which will present a more accurate description of the existing instrumentation, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of any accident previously evaluated.

2. The proposed change will provide a more complete and accurate description of existing equipment and instrumentation. It does not represent a change in logic, nor does it reflect a change in the instrument design or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change has no effect on the design or operation of the equipment represented. It provides a more appropriate representation of existing equipment. Therefore, there is no impact on the margin of safety.

Proposed Change Number 16

Replace Instrument Number D12-RM-N010A,B with Instrument Number D12-RE-N010A,B in Item 2.a, "Reactor Building Exhaust Radiation - High," in Table 3.3.2-1 under Secondary Containment Isolation on Page 3/4 3-13.

Basis

A signal from the reactor building exhaust radiation - high instrumentation closes the secondary containment isolation dampers, thereby establishing secondary containment and minimizing the release of radiation from the reactor building during an accident. The dampers work in conjunction with the standby gas treatment system to establish a negative pressure in the reactor building and a controlled vent path through the standby gas treatment system filters and the stack. Secondary containment isolation would occur during a LOCA due to reactor vessel water level - low, level 2 or drywell pressure - high. This would minimize the release of radioactivity from the reactor building. Secondary containment isolation is also assumed to occur during a refueling accident due to a signal from the reactor building exhaust radiation - high instrumentation to contain radiation released in a fuel handling accident.

Item 2.a currently incorrectly references Instrument Number D12-RM-N010A,B as the reactor building exhaust radiation - high instrumentation. This instrument should be identified as D12-RE-N010A,B. The proposed change does not result from a plant modification; the existing instrument number is not referenced correctly. It does not represent a change to the instrumentation. The proposed change will result in the instrument loop being more clearly and completely identified in the Technical Specifications, and reduce the possibility of confusion concerning which instrument channel is subject to those Technical Specification requirements.

10CFR50.92 Evaluation

This change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will result in the reactor building exhaust radiation - high instrumentation loop being more completely and correctly identified. The nomenclature will be consistent with that used in other plant documents and programs. It does not represent any change to the devices which perform the specified function; it more clearly references existing instrumentation. This change will reduce misunderstanding about which devices are associated with the isolation function. Thus, there is no change in the probability of an accident, nor is there any change in the consequences of any accident.

2. The existing instrument number listed under Item 2.a is not consistent with other plant documents and programs. There is no change to the instrument it represents; this change will only revise the instrument number to match that listed in other documentation. This change may reduce misunderstanding about which devices are associated with the isolation function. Thus, this change does not create the possibility of a new accident.

3. The proposed change will result in the reactor building exhaust radiation - high instrumentation being more correctly identified in the Technical Specifications. The nomenclature will be consistent with that used in other plant documents and programs and, therefore, reduce confusion about which instrumentation is being referenced. There is no physical change to the instrumentation; only its number is being revised in the Technical Specifications. Therefore, this change has no impact on the margin of safety.

Proposed Change Number 17

Add Instrument Number D12-RM-K609A,B to Item 2.a, "Reactor Building Exhaust Radiation - High," in Table 3.3.2-1 under Secondary Containment Isolation on Page 3/4 3-13.

Basis

A signal from the reactor building exhaust radiation - high instrumentation closes the secondary containment isolation dampers, thereby establishing secondary containment isolation and minimizing the release of radiation from the reactor building during an accident. The dampers work in conjunction with the standby gas treatment system to establish a negative pressure in the reactor building and a controlled vent path through the standby gas treatment system filters and the stack. Secondary containment isolation would occur during a LOCA due to reactor vessel water level - low, level 2 or drywell pressure - high. This would minimize the release of radioactivity from the reactor building. Secondary containment isolation is also assumed to occur during a refueling accident due to a signal from the reactor building exhaust radiation - high instrumentation to contain radiation released in a fuel handling accident. Item 2.a currently lists only the radiation monitor D12-RM-N010A,B. It should also list the radiation monitor drawer D12-RM-K609A,B which is located in the control room.

The proposed change does not result from a plant modification; it is an addition of a reference to existing instrumentation which is currently associated with this system, but not listed. The instrumentation will continue to perform its intended function just as before; however, it will now be listed in its appropriate place in the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will result in the reactor building exhaust radiation - high instrumentation loop being more completely and correctly identified. The list of instruments associated with the reactor building exhaust radiation - high instrumentation will be more complete and consistent with that used in other plant documents and programs. It does not represent any change to the devices which perform the specified function; it more clearly references existing instrumentation. This change will reduce misunderstanding about which devices are associated with the isolation function. Thus, there is no change in the probability of an accident, nor is there any change in the consequences of any accident.

2. The existing instrument number listed under Item 2.a is not a complete representation of the instrumentation associated with the isolation instrumentation. There is no change to the instrumentation represented; this change will only add an additional instrument number to provide a more complete list of associated instruments. This change may reduce misunderstanding about which devices are associated with the isolation function. Thus, this change does not create the possibility of a new accident.

3. The proposed change will result in the reactor building exhaust radiation - high instrumentation being more correctly identified in the Technical Specifications. The instrumentation will be referenced more completely and thereby reduce confusion about which instrumentation is associated with the isolation signal. There is no physical change to the instrumentation; an additional reference is being added to provide a more complete description of the isolation instrumentation. Therefore, this change has no impact on the margin of safety.

Proposed Change Number 18

Delete Valve Group 7 from Item 2.b, "Drywell Pressure - High," in Table 3.3.2-1 under Secondary Containment Isolation on Page 3/4 3-13.

Basis

Currently, Technical Specification Table 3.6.3-1 identifies the reactor vessel head spray, residual heat removal (RHR) shutdown cooling supply, and RHR injection isolation valves as Group 8 valves. Table 3.3.2-1 indicates Group 8 valves isolate on reactor vessel water level - low, level 1 (LL1) and reactor steam dome pressure - high.

Table 3.3.2-1 also lists Group 7 valves as having isolation signals from reactor vessel water level - low, level 1, drywell pressure - high, and reactor steam dome pressure - high. However, Table 3.6.3-1 does not identify any valves as Group 7 valves.

The proposed change deletes the reference to Valve Group 7 under "Valve Groups Operated By Signal" in Item 2.b of Table 3.3.2-1. The following factors contributed to the confusion associated with creation of a Valve Group 7.

CUSTOM TO STANDARD TECHNICAL SPECIFICATION CONVERSION

In 1977, BSEP converted from the custom Technical Specification format to the standard format. The Standard Technical Specifications (STS) were requested for both units on August 22, 1977, and issued on November 23, 1977.

As part of this conversion, several changes were made to the valve groups to more completely and consistently describe the isolation signals. Valve Group 2 originally contained those valves which were assumed to isolate during a loss of coolant accident but were not located on high energy lines with their own leak detection. The isolation signals associated with these valves were LL1 and drywell pressure - high. There were also several subgroups within Group 2 that isolated on additional signals. In the conversion from custom to STS, new valve groups were created to recognize these distinctions. Group 6 was created for those valves originally in Group 2 that also isolated on reactor building vent high radiation. Group 8 was created for those valves originally in Group 2 associated with the RHR system that isolate on LL1 and reactor vessel steam dome pressure - high.

LOGIC DESIGN CHANGES

In 1975, General Electric issued Service Information Letter 131, which recommended several changes to the containment isolation logic. Experience had shown that the normal water level drops that occurred during scrams due to collapse of voids exceeded the LL1 setpoint. This caused unnecessary valve group isolations.

Three changes were recommended for BSEP to reduce the complexity of scram recoveries and to reduce demands on the operators caused by unnecessary system isolations and transients so they could concentrate on the scram itself and be more observant for unusual occurrences. These changes were:

1. Lower reactor water cleanup system isolation from LL1 to reactor vessel water level - low, level 2 (LL2),

2. Lower secondary containment isolation and standby gas treatment system initiation from LL1 to LL2,

3. Delete the drywell pressure - high isolation signals for RHR suction and discharge valves.

Plant modification and Technical Specification change packages were prepared to support these logic changes. The original amendment request for BSEP-2 was made on February 2, 1976. The BSEP-2 amendment request was submitted in custom format. On June 22, 1977, a submittal was made that indicated that the same changes were applicable to BSEP-1 as well, and requested that the original change be amended to include BSEP-1. The amendment for BSEP-2 was issued on October 12, 1977. The amendment for BSEP-1 was issued in STS format on April 4, 1979, after conversion to STS format.

Both requests associated with the logic change requested that the drywell pressure - high signal be deleted as an RHR isolation signal. However, the Technical Specification pages originally submitted with the BSEP-2 submittal failed to indicate that the drywell pressure - high signal was also to be deleted for the RHR head spray valves (E11-F022 and E11-F023). This change was encompassed in the logic change; however, the Technical Specification pages were not changed to reflect this.

Apparently, this failure to specifically identify the RHR reactor head spray valves as no longer having the drywell pressure - high signal associated with them, led those preparing the BSEP-2 custom to STS conversion package to assume that they should be treated differently (i.e., no longer as Group 2 valves, but not as Group 8 valves either, as they should have been). The isolation signals associated with the Group 7 valves were assumed to be LL1, reactor steam dome pressure - high and drywell pressure - high. Thus, when Valve Groups 6 and 8 were created in the BSEP-2 STS, Valve Group 7 was created specifically for the RHR reactor head spray valves solely because they were thought to still isolate on drywell pressure - high.

Valve Group 7 was created in Table 3.6.3-1 only for BSEP-2 in the original issuance of the STS. The BSEP-2 Technical Specifications now listed a Valve Group 7 for the E11-F022 and E11-F023 valves in Table 3.6.3-1, and Group 7 as a valve group operated by the LL1, reactor steam dome pressure - high and drywell pressure - high isolation signals in Table 3.2.3-1. When the BSEP-1 STS were issued, there was no reference to Group 7 listed anywhere because the logic modification had not yet been performed.

The BSEP-1 Technical Specification change associated with the logic change was developed from the initial logic change request for BSEP-2. As a result, the BSEP-1 amendment, which was an amendment to the STS, included the Group 7 references in Table 3.2.3-1 as a valve group operated by the LL1, reactor steam

dome pressure - high and drywell pressure - high signals, but did not identify Valve Group 7 and its associated valves in Table 3.6.3-1.

On September 7, 1982, an additional amendment request was submitted which stated that a typographical error had been made in the classification of the RHR head spray valves. The valves were not actually Group 7; they were to be Group 8 valves. This amendment was issued on March 6, 1984 and changed the classification of the valves in Table 3.6.3-1 from Group 7 to Group 8. This actually applied only to BSEP-2, since BSEP-1 never identified a Group 7 in Table 3.6.3-1; however, the amendment was issued for both units. This amendment request did not delete the superfluous references to Valve Group 7 in Table 3.2.3-1. Thus, these references need to be deleted.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change deletes references to a valve group that currently does not exist, and was never intended to exist. The valves once considered Group 7 valves, E11-F022 and E11-F023, are actually Group 8 valves, and isolate on the signals which isolate the Group 8 valves. There are no valves associated with Valve Group 7, and it has been determined that no valves should be associated with the current Valve Group 7. Since Group 7 does not currently provide any means of protection, this change does not affect the probability or consequences of an accident.

2. Valve Group 7 currently does not include any valves. Therefore, it can perform no safety function, nor hamper any existing safety function. Thus, its deletion does not create the possibility of a new or different kind of accident.

3. Valve Group 7 cannot provide any means of protection, because it does not currently include any valves. Its deletion does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, the margin of safety remains unchanged.

Proposed Change Number 19

Add reference to Footnote (k), "Secondary containment isolation dampers as listed in plant procedure ________." under the valve group column in Item 2.b, "Drywell Pressure - High," in Table 3.3.2-1 under Secondary Containment Isolation on Page 3/4 3-13.

Basis

A signal from the drywell pressure - high instrumentation closes the secondary containment isolation dampers, thereby establishing secondary containment isolation and minimizing the release of radiation from the reactor building during an accident. The dampers work in conjunction with the standby gas treatment system to establish a negative pressure in the reactor building and a controlled vent path through the standby gas treatment system filters and the stack. Secondary containment isolation would occur during a LOCA due to reactor vessel water level - low, level 2 or drywell pressure - high. This would minimize the release of radioactivity from the reactor building. Secondary containment isolation is also assumed to occur during a refueling accident due to a signal from the reactor building exhaust radiation - high instrumentation to contain radiation released in a fuel handling accident.

The current Technical Specifications reference Valve Groups 2, 6 and 7 under Item 2.b but do not reference the dampers which in fact establish secondary containment isolation. Deletion of Valve Group 7 is addressed in Proposed Change Number 18. The proposed change adds a reference to Footnote (k) to the current references to Valve Groups 2 and 6 in Item 2.b to provide the needed reference to the secondary containment isolation dampers.

Footnote (k) states, "Secondary containment isolation dampers as listed in plant procedure _______." The secondary containment isolation dampers are currently listed in Table 3.6.5.2-1 of the Technical Specifications. This table is being deleted, and the information relocated to plant procedures, as described elsewhere in this submittal.

Addition of the footnote reference is necessary because there is no reference to the secondary containment dampers in Item 2.b. This change will simplify the Technical Specifications, eliminate discrepancies, and reduce the potential for misinterpretation. The procedure which will list the secondary containment isolation dampers once they are removed from the Technical Specifications is currently being developed. The Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal.

10CFR50.92 Evaluation

This change does not involve a significant hazards consideration for the following reasons:

1. The proposed change provides a more accurate reference to the appropriate valves covered by Item 2.b under Secondary Containment Isolation. The current Technical Specifications list valve groups which are isolated by a signal from the drywell pressure - high instrumentation, but do not establish secondary containment isolation. The dampers which establish secondary containment isolation will be listed in procedure ______.

   There is no change in the design, operation, reliability, or testing requirements of existing plant systems. This change provides a more complete reference to existing plant equipment in its appropriate place in the table. Thus, there is no change in the probability of occurrence of an accident, nor is there any change in the consequences of any accident previously evaluated.

2. The proposed change provides a more accurate reference to the appropriate valves covered by Item 2.b. The current Technical Specifications list valve groups which are isolated by a signal from the drywell pressure - high instrumentation, but do not establish secondary containment isolation. The dampers which establish secondary containment isolation will be listed in a plant procedure which will be identified at a later date. The proposed change provides a reference in Item 2.b to the secondary containment damper listing.

   There is no change in the design, operation, reliability, or testing requirements of existing plant systems. This change provides a more complete reference to existing plant equipment in an appropriate place in plant procedures. Thus, no new or different types of accident situations are created.

3. The proposed change provides references to the appropriate valves covered by Item 2.b. This change adds a reference to the dampers which actually provide secondary containment. There is no change in the response, capability, reliability, or testing requirements of plant systems. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, there is no effect on the margin of safety.

Proposed Change Number 20

Place the operability requirements for Valve Groups 2 and 6 on a separate line under Item 2.b, "Drywell Pressure - High," under Secondary Containment Isolation on Page 3/4 3-13 and specify the correct requirements for the valve group.

Basis

Currently, Item 2.b specifies the same operability requirements for Valve Groups 2 and 6. The existing operability requirements apply only to the secondary containment isolation dampers, to which a reference is being addid via Proposed Change Number 19. Action Statement 20, "Be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours," applies to Valve Groups 2 and 6, while Action 23, "Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour," applies to the secondary containment isolation dampers.

The proposed change places the Valve Group 2 and 6 information on a separate line with its own action statement, Action 20 under Item 2.b. This makes the information provided more complete and accurate. No change has been made to the design or operation of the instrumentation or equipment; only the way it is referenced in Item 2.b will be changed.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change provides additional information in Item 2.b to make it a more complete and accurate representation of the existing equipment and instrumentation. The logic associated with the instrumentation has not changed, nor has it design or operation. The proposed change references the appropriate operational conditions for Valve Groups 2 and 6 under Item 2.b. The same operability requirements are listed for Groups 2 and 6 in Item 1.b under primary containment isolation. Because this change provides additional information which will present a more accurate description of the existing instrumentation, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of any accident previously evaluated.

2. The proposed change will provide a more complete and accurate description of existing equipment and instrumentation. It does not represent a change in logic, nor does it reflect a change in the instrument design or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change has no effect on the design or operation of the equipment represented. It provides a more appropriate representation of existing equipment. Therefore, there is no impact on the margin of safety.

Proposed Change Number 21

Add reference to Footnote (k) "Secondary containment isolation dampers as listed in plant procedure _______," under the valve group column in Item 2.c, "Reactor Vessel Water Level - Low, Level 2," in Table 3.3.2-1 under Secondary Containment Isolation on Page 3/4 3-13.

Basis

A signal from the reactor vessel water level - low, level 2 instrumentation closes the secondary containment isolation dampers, thereby establishing secondary containment isolation and minimizing the release of radiation from the reactor building during an accident. The dampers work in conjunction with the standby gas treatment system to establish a negative pressure in the reactor building and a controlled vent path through the standby gas treatment system filters and the stack. Secondary containment isolation would occur during a LOCA due to reactor vessel water level - low, level or drywell pressure - high isolation. This would minimize the release of radioactivity from the reactor building. Secondary containment isolation is also assumed to occur during a refueling accident due to a signal from the reactor building exhaust radiation - high instrumentation to contain radiation released in a fuel handling accident.

The current Technical Specifications reference Valve Groups 1 and 3 under Item 2.c but do not reference the dampers which in fact establish secondary containment isolation. There should also be a reference to the secondary containment isolation dampers, which serve to isolate secondary containment.

The proposed change adds a reference to Footnote (k) to the current references to Valve Groups 1 and 3 in Item 2.c. Footnote (k) states, "Secondary containment isolation dampers as listed in plant procedure _______." The secondary containment isolation dampers are currently listed in Table 3.6.5.2-1 of the Technical Specifications. This table is being deleted as described elsewhere in this submittal.

Addition of the footnote reference is necessary because there is no reference to the secondary containment dampers in Item 2.b. This change will simplify the Technical Specifications, eliminate discrepancies, and reduce the potential for misinterpretation. The procedure which will list the secondary containment isolation dampers once they are removed from the Technical Specifications is currently being developed. The Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change provides a more accurate reference to the appropriate valves covered by Item 2.b. The current Technical Specifications list valve groups which are isolated by a signal from the reactor vessel water level - low, level 2 instrumentation, but do not establish secondary containment. The equipment which establishes secondary containment isolation upon receipt of a signal from the reactor building exhaust radiation - high instrumentation is listed in plant procedure _____.

   There is no change in the design, operation, reliability, or testing requirements of existing plant systems. This change provides a more complete reference to existing plant equipment in its appropriate place in the table. Thus, there is no change in the probability of occurrence of an accident, nor is there any change in the consequences of any accident previously evaluated.

2. The proposed change provides a more accurate reference to the appropriate valves covered by Item 2.c. The current Technical Specifications list valve groups which are isolated by a signal from the reactor vessel water level - low, level 2 instrumentation, but do not establish secondary containment. The equipment which establishes secondary containment isolation upon receipt of a signal from the reactor vessel water level - low, level 2 instrumentation will be listed in a plant procedure which will be specifically identified at a later date. The proposed change provides a reference to the secondary containment damper table in Item 2.c.

   There is no change in the design, operation, reliability, or testing requirements of existing plant systems. This change provides a more complete reference to existing plant equipment in an appropriate place in plant procedures. Thus, no new or different types of accident situations are created.

3. The proposed change provides references to the appropriate valves covered by Item 2.c. This change adds a reference to the dampers which actually provide secondary containment isolation. There is no change in the response, capability, reliability, or testing requirements of plant systems. Thus, there is no effect on the margin of safety.

Proposed Change Number 22

Place the operability requirements for Valve Group 1 on a separate line under Item 2.c, "Reactor Vessel Water Level - Low, Level 2," under Secondary Containment Isolation in Table 3.3.2-1 on Page 3/4 3-13 and specify the correct requirements for the valve group (BSEP-1 only).

Basis

Currently, Item 2.c specifies the same operability requirements for Valve Groups 1 and 3. The existing operability requirements apply only to the secondary containment isolation dampers, which isolate the secondary containment. Action Statement 20, "Be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours," applies to Valve Group 1, while Action Statement 23, "Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour," applies to the secondary containment isolation dampers.

The proposed change places the Valve Group 1 information on a separate line with its own action statement, Action Statement 20, under Item 2.c. This makes the information provided more complete and accurate. No change has been made to the design or operation of the instrumentation or equipment; only the way it is referenced in Item 2.c will be changed.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change provides additional information in Item 2.c to make it a more complete and accurate representation of the existing equipment and instrumentation. The logic associated with the instrumentation has not changed, nor has it design or operation. The proposed change references the appropriate operational conditions for Valve Group 1 under Item 2.c. The same operability requirements are listed for Group 1 in Item 1.a.2 under primary containment isolation. Because this change provides additional information which will present a more accurate description of the existing instrumentation, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of any accident previously evaluated.

2. The proposed change will provide a more complete and accurate description of existing equipment and instrumentation. It does not represent a change in logic, nor does it reflect a change in the instrument design or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change has no effect on the design or operation of the equipment represented. It provides a more appropriate representation of existing equipment. Therefore, there is no impact on the margin of safety.

Proposed Change Number 23

Place the operability requirements for Valve Group 3 on a separate line under Item 2.c, "Reactor Vessel Water Level - Low, Level 2," under Secondary Containment Isolation in Table 3.3.2-1 on Page 3/4 3-13 and specify the correct requirements for the valve group.

Basis

Currently, Item 2.c specifies the same operability requirements for Valve Groups 1 and 3. The existing operability requirements apply only to the secondary containment isolation dampers, which isolate the secondary containment. Action Statement 24, "Isolate the reactor water cleanup system," applies to Valve Group 3, while Action Statement 23, "Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour," applies to the secondary containment isolation dampers.

The proposed change places the Valve Group 3 information on a separate line with its own action statement, Action Statement 24, under Item 2.c. This makes the information provided more complete and accurate. No change has been made to the design or operation of the instrumentation or equipment; only the way it is referenced in Item 2.c will be changed.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change provides additional information in Item 2.c to make it a more complete and accurate representation of the existing equipment and instrumentation. The logic associated with the instrumentation has not changed, nor has it design or operation. The proposed change references the appropriate operational conditions for Valve Group 3 under Item 2.c. The same operability requirements are listed for Group 3 in Item 3.e under primary containment isolation. Because this change provides additional information which will present a more accurate description of the existing instrumentation, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of any accident previously evaluated.

2. The proposed change will provide a more complete and accurate description of existing equipment and instrumentation. It does not represent a change in logic, nor does it reflect a change in the instrument design or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change has no effect on the design or operation of the equipment represented. It provides a more appropriate representation of existing equipment. Therefore, there is no impact on the margin of safety.

Proposed Change Number 24

Replace "and" with ";" in the instrument number listings in Item 2.c, "Reactor Vessel Water Level - Low, Level 2," in Table 3.3.2-1 under Secondary Containment Isolation on Page 3/4 3-13.

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table, and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability of the primary containment system nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 25

Replace Instrument Number G31-dFS-N603-1A,1B with Instrument Number G31-FDS-N603-1A,1B in Item 3.a, "Δ Flow - High," in Table 3.3.2-1 under Reactor Water Cleanup System Isolation on Page 3/4 3-13.

Basis

The isolation instrumentation listed in Item 3.a isolates the reactor water cleanup system supply valves in the event of a rupture of the reactor water cleanup system. This minimizes the amount of radioactivity, energy, and inventory lost from the reactor vessel.

Currently, the Technical Specifications list Instrument Number G31-dFS-N603-1A,1B under Item 3.a. The proposed change will revise that number to G31-FDS-N603-1A,1B. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference G31-dFS-N603-1A,1B with G31-FDS-N603-1A,1B. It does not change the device which performs the specified function; it only more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device, and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 3.a of Table 3.3.2-1. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 26

Replace Instrument Number G31-TS-N602A,B,C,D,E,F with Instrument Number G31-TDS-N602A,B,C,D,E,F in Item 3.c, "Area Ventilation Δ Temperature - High," in Table 3.3.2-1 under Reactor Water Cleanup System Isolation on Page 3/4 3-13.

Basis

The isolation instrumentation listed in Item 3.c isolates the reactor water cleanup system supply valves in the event of a rupture of the reactor water cleanup system. This minimizes the amount of radioactivity, energy, and inventory lost from the reactor vessel.

Currently, the Technical Specifications list Instrument Number G31-TS-N602A,B,C,D,E,F under Item 3.c. The proposed change will revise that number to G31-TDS-N602A,B,C,D,E,F. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference G31-TS-N602A,B,C,D,E,F with G31-TDS-N602A,B,C,D,E,F. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device, and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new or different type of accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 3.c of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 27

Place the operability requirements for Valve Group 1 on a separate line in Item 3.e, "Reactor Vessel Water Level - Low, Level 2," under Reactor Water Cleanup System Isolation on Page 3/4 3-14 and specify the correct requirements for the valve group (BSEP-1 only).

Basis

Currently, Item 3.e in the BSEP-1 Technical Specifications specifies the same operability requirements for Valve Groups 1 and 3. The existing operability requirements apply only to Group 3, which isolates the reactor water cleanup system. Action Statement 24, "Isolate the reactor water cleanup system," applies to Valve Group 3, while Action Statement 20, "Be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours," applies only to the Group 1 Valves. The Group 1 valves isolate the main steam lines and the main steam line drain valves on a signal from the reactor vessel water level - low, level 2 instrumentation.

The proposed change places the Valve Group 1 information on a separate line with its own action statement (Action 20) under Item 3.e. This makes the information provided more complete and accurate. No change has been made to the design or operation of the instrumentation or equipment; only the way it is referenced in Item 3.e will be changed.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change provides additional information in Item 3.e to make it a more complete and accurate representation of the existing equipment and instrumentation. The logic associated with the instrumentation has not changed, nor has its design or operation. The proposed change references the appropriate operational conditions for Valve Group 1 under Item 3.e. The same operability requirements are listed for Group 1 in Item 1.a.2 under primary containment isolation. Because this change provides additional information which will present a more accurate description of the existing instrumentation, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of any accident previously evaluated.

2. The proposed change will provide a more complete and accurate description of existing equipment and instrumentation. It does not represent a change in logic, nor does it reflect a change in the instrument design or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change has no effect on the design or operation of the equipment represented. It provides a more appropriate representation of existing equipment. Therefore, there is no impact on the margin of safety.

Proposed Change Number 28

Replace "and" with ";" in the instrument number listings in Item 3.e, "Reactor Vessel Water Level - Low, Level 2," in Table 3.3.2-1 under Reactor Water Cleanup System Isolation on Page 3/4 3-14.

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore there is no impact on the margin of safety.

Proposed Change Number 29

Add new Item 3.f, "Δ Flow - High - Time Delay Relay," and associated Instrument Number G31-R616C,D to Table 3.3.2-1 under Reactor Water Cleanup System Isolation on Page 3/4 3-14.

Basis

The reactor water cleanup (RWCU) system isolation instrumentation isolates the RWCU supply isolation valves in the event of a rupture of the RWCU system. This minimizes the amount of radioactivity, energy, and inventory lost from the reactor vessel. A potential rupture is sensed by any of the following signals: high delta supply/return RWCU flow, high RWCU area temperature, high delta supply/return ventilation temperature, or reactor vessel water level - low, level 2. The system also isolates on standby liquid control system initiation to prevent dilution of the poison injected due to cleanup from RWCU operation.

The current Technical Specifications do not specifically reference operability and surveillance requirements for the existing RWCU Δ flow - high time delay relays. The proposed change adds operability requirements for these relays. The proposed requirements are similar to those listed for Item 3.a since these relays are part of the Δ flow - high logic and have the same function. Initially, the time delay relays were considered to be a part of the isolation logic described in Item 3.a and were not listed separately in the table. These relays do not initiate any isolation signal; however, they are an important part of the instrumentation. It has been determined that they should be addressed separately in the table to avoid possible confusion.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes specific operability and surveillance requirements for instrumentation which is designed to detect an accident and not prevent one. These requirements are consistent with the accident analysis assumptions and are similar to those for other associated instrumentation. There is no change to the equipment or in how the equipment is operated; only references to existing instrumentation are being added to the table. Since the establishment of operability and surveillance requirements does not change the design or affect the operation of the instrumentation, there is no increase in the probability of an accident, nor is there any change in the consequences of any accident previously evaluated.

2. Specification of the correct operability and surveillance requirements provides additional assurance that the instrumentation will perform its intended safety function. It does not change the design or operation of any equipment or cause it to operate in a manner not previously assumed. It does not affect any equipment which could cause an accident. It only affects instrumentation designed to detect the occurrence of an accident. Thus, this change does not create the possibility of a new accident.

3. Specification of specific operability and surveillance requirements that are similar to those for other isolation instrumentation in this system will not change the operation of the affected equipment. There will be no change in the reliability, response characteristics, or setpoints of the instrumentation. Therefore, there is no change in the margin of safety.

Proposed Change Number 30

Replace "HPCI Steam Line High Flow Time Delay Relay" with "HPCI Steam Line Flow - High Time Delay Relay" in the title of Item 4.a.2 in Table 3.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-14.

Basis

The proposed change was made solely to provide consistency with other titles throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The instrumentation itself has not changed, nor has its design or operation. Only its title has been reworded to be consistent with other titles in the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the instrument design or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table. It does not involve a change in instrumentation, design, or operation. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 31

Add operability requirements for a new Valve Group 7 and Footnote (j) to the Valve Group column in Item 4.a.3, "HPCI Steam Supply Pressure - Low," in Table 3.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-14.

Basis

The Group 4 isolation valves consist of the high pressure coolant injection (HPCI) inboard and outboard steam line isolation valves and the HPCI torus suction isolation valves. These valves isolate in the event of a HPCI steam line break to mitigate the consequences of the break.

The HPCI system also has turbine exhaust vacuum breaker isolation valves (E41-F075 and E41-F079) on a vacuum relief line for the HPCI turbine exhaust. This line helps prevent the creation of a water column in the exhaust line. Preventing development of this column reduces the piping loads which could exist if the turbine was restarted. The valves isolate to establish primary containment on coincident HPCI steam line low pressure and drywell pressure - high signals. The low steam line pressure signal indicates HPCI is isolated or no longer needed because the reactor is depressurized. The drywell pressure - high signal indicates a LOCA exists and primary containment isolation is needed.

The current Technical Specifications specify operability and surveillance requirements for the Group 4 isolation valves but not for the vacuum breaker isolation valves.

The proposed change reflects the creation of a new Valve Group 7 which includes the HPCI turbine exhaust vacuum breakers E41-F075 and E41-F079. Operability and surveillance requirements for these valves are being added under Item 4.a.3 with a new Footnote (j) which states, "Valve isolation dependent upon low steam supply pressure coincident with high drywell pressure." This footnote recognizes the existing logic design and the need for both signals to be present for conditions to exist which indicate that the isolation safety function is necessary. Each of the two trip systems consists of one channel, and the minimum number of operable channels per trip system is one. The instrumentation is required to be operable in Operational Conditions 1, 2, and 3 since that is when primary containment integrity and HPCI operability are both required. In the event both trip systems do not meet the operability requirements, existing Action Statement 25, "Close the affected system isolation valves and declare the affected system inoperable," is specified. This is the conservative action if the logic is completely inoperable and is the same action specified for the Group 4 isolation instrumentation.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes specific operability and surveillance requirements for existing instrumentation which were not originally included in the Technical Specifications. The HPCI turbine exhaust vacuum breakers isolate on coincident HPCI steam line pressure - low and drywell pressure - high. The steam line pressure - low signal indicates HPCI is isolated or no longer needed because the reactor is depressurized, and the drywell pressure - high signal indicates a LOCA exists and primary containment isolation is needed.

   This change adds operability requirements for existing instrumentation to provide additional assurance that primary containment is isolated when necessary. No new equipment is being added, and existing equipment will perform its safety function in the same manner as before. These requirements are being added to the Technical Specifications for clarity and completeness. Thus, there is no change in the probability of an accident, nor is there any change in the consequences of any accident.

2. The proposed change adds operability requirements to the Technical Specifications for existing instrumentation. This will provide a more complete and accurate document, thereby enhancing primary containment isolation assurance. It does not provide an opportunity for any new accidents because no equipment, operability requirements, or setpoints are being altered.

3. The proposed change adds requirements for existing instrumentation to the Technical Specifications. It does not reflect a physical change to the plant systems; it only provides a more complete and accurate description of existing equipment. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, there is no impact on the margin of safety of the plant.

Proposed Change Number 32

Replace "and" with ";" in the instrument number listings in Item 4.a.5, "Bus Power Monitor," in Table 3.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-15.

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table, and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 33

Replace Instrument Number E51-dTS-N604C,D with Instrument Number E51-TDS-N604C,D in Item 4.a.8, "HPCI Steam Line Area Δ Temperature - High," in Table 3.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-15.

Basis

The isolation instrumentation listed in Item 4.a.8 isolates the high pressure coolant injection (HPCI) system should there be a HPCI steam line break.

Currently, the Technical Specifications list Instrument Number E51-dTS-N604C,D under Item 4.a.8. The proposed change will revise that number to E51-TDS-N604C,D. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference E51-dTS-N604C,D with E51-TDS-N604C,D. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 4.a.8 of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 34

Replace "Emergency Area Cooler Temperature - High" with "HPCI Equipment Area Temperature - High" in the title of Item 4.a.9 in Table 3.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-15.

Basis

The emergency area cooler temperature - high isolation signal isolates the Group 4 isolation valves in the event of a high pressure coolant injection (HPCI) steam line break to mitigate the consequences of such a break. The Group 4 isolation valves include the HPCI inboard and outboard steam line isolation valves and the HPCI torus suction isolation valves.

The current Technical Specifications have an inaccurate description of the trip function. The proposed change revises the description of the trip function from "Emergency Area Cooler Temperature - High" to "HPCI Equipment Area Temperature - High."

This change is necessary so that the trip function description more clearly reflects the actual area the instrumentation monitors. The existing description is a generic BWR term which does not relate to the actual BSEP design. The area the instrumentation monitors at BSEP is referred to in other plant documents and programs as the "HPCI Equipment Area." This change will reduce the potential for confusion by more clearly describing the actual location of the instrumentation.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will change the description of a trip function to more clearly identify the area the instrumentation monitors. The current wording, "Emergency Area Cooler," does not reflect the commonly used description of the area. "HPCI Equipment Area" more clearly describes the area. This change will reduce confusion concerning which area is being referred to.

   No changes to the referenced instrumentation are reflected by this change. The instrumentation will function exactly as before. Therefore, there is no change in the probability of an accident, nor is there any change in the consequences of an accident.

2. The proposed change revises the trip function description so the appropriate trip function is more clearly identified in the Technical Specifications. It does not change the required function of the instrumentation; it merely clarifies the description of an existing trip function. Thus, no new accident possibilities are created.

3. The proposed change provides a clarification in the description of an existing isolation function. It does not change the design, reliability, or function of any plant equipment; it merely references them more clearly. Thus, there is no impact on the margin of safety as a result of this change.

Proposed Change Number 35

Add new Item 4.a.10, "Drywell Pressure - High," and associated Instrument Numbers E11-PT-N011C,D and E11-PTS-N011C-2,D-2 to Table 3.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-15.

Basis

Valve Group 4 isolates the high pressure coolant injection (HPCI) inboard and outboard steam line isolation valves and the HPCI torus suction isolation valves in the event of a HPCI steam line break to mitigate the consequences of the break.

The HPCI system also has turbine exhaust vacuum breaker isolation valves (E41-F075 and E41-F079) on a vacuum relief line for the HPCI turbine exhaust. This line helps prevent the creation of a water column in the exhaust line. Preventing development of this column reduces the piping loads which could exist if the turbine is restarted. The valves isolate on coincident HPCI steam line pressure - low and drywell pressure - high signals to establish primary containment integrity. The steam line pressure - low signal indicates HPCI is isolated or no longer needed because the reactor is depressurized. The drywell pressure - high signal indicates a loss of coolatn accident (LOCA) exists and primary containment isolation is needed.

The current Technical Specifications specify operability and surveillance requirements for the Group 4 isolation valves but not for the vacuum breaker isolation valves. The proposed change reflects the creation of a new Valve Group 7, which includes the HPCI turbine exhaust vacuum breakers E41-F075 and E41-F079. Operability and surveillance requirements for the actuation instrumentation are added under Item 4.a.3 for the HPCI steam supply pressure - low instrumentation portion of the logic. A new Item 4.a.10 is being added for the drywell pressure - high portion of the logic. Both additions reference a new Footnote (j), which states, "Valve isolation depends upon low steam supply pressure coincident with high drywell pressure." This footnote recognizes the existing logic design and the need for both signals to be present for conditions to exist which indicate that the isolation safety function is necessary. The minimum number of operable channels per trip system is one, with each of the two trip systems consisting of one channel. The instrumentation is required to be operable in Operational Conditions 1, 2, and 3 since that is when primary containment integrity and HPCI operability are both required. In the event both trip systems do not meet the operability requirements, existing Action Statement 25, "Close the affected system isolation valves and declare the affected system inoperable," is specified. This is the conservative action if the logic is completely inoperable and is the same action specified for the Group 4 isolation instrumentation.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes specific operability and surveillance requirements for existing instrumentation which was not originally included in the Technical Specifications. The HPCI turbine exhaust vacuum breakers isolate on coincident HPCI steam line pressure - low and drywell pressure - high signal. The steam line pressure - low signal indicates HPCI is isolated or no longer needed because the reactor is depressurized, and the drywell pressure - high signal indicates a LOCA exists and primary containment isolation is needed.

   This change adds operability requirements for existing instrumentation to provide additional assurance that primary containment is isolated when necessary. No new equipment is being added, and existing equipment will perform its safety function in the same manner as before. These requirements are being added to the Technical Specifications for clarity and completeness. Thus, there is no change in the probability of an accident, nor is there any change in the consequences of any accident.

2. The proposed change adds operability requirements to the Technical Specifications for existing instrumentation. This will provide a more complete and accurate document, thereby enhancing primary containment isolation assurance. It does not provide an opportunity for any new accidents because no equipment, operability requirements, or setpoints are being altered.

3. The proposed change adds requirements for existing instrumentation to the Technical Specifications. It does not reflect a physical change to the plant systems; it only provides a more complete and accurate description of existing equipment. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, there is no impact on the margin of safety of the plant.

Proposed Change Number 36

Replace "RCIC Steam Line High Flow Time Delay Relay," with "RCIC Steam Line Flow - High Time Delay Relay" in the title of Item 4.b.2 in Table 3.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-16.

Basis

The proposed change was made solely to provide consistency with other titles throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The instrumentation itself has not changed, nor has its design or operation. Only its title has been reworded to be consistent with other titles in the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of any accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the instrument design or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table. It does not involve a change in instrumentation, design or operation. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 37

Add operability requirements for a new Valve Group 9 and Footnote (j) to the Valve Group column in Item 4.b.3, "RCIC Steam Supply Pressure - Low," in Table 3.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-16.

Basis

The Group 5 isolation valves include the reactor core isolation cooling (RCIC) inboard and outboard steam line isolation valves. These valves isolate in the event of a RCIC steam line break to mitigate the consequences of the break.

The RCIC system also has turbine exhaust vacuum breaker isolation valves (E51-F062 and E51-F066) on a vacuum relief line for the RCIC turbine exhaust. The line helps prevent the creation of a water column in the exhaust line. Preventing development of this column reduces the piping loads which could exist if the turbine is restarted. The valves isolate on coincident RCIC steam line pressure - low and drywell pressure - high signals to establish primary containment integrity. The steam line pressure - low signal indicates RCIC is isolated or no longer needed because the reactor is depressurized. The drywell pressure - high signal indicates a loss of coolant accident (LOCA) exists and primary containment isolation is needed.

The current Technical Specifications specify operability and maintenance requirements for the Group 5 isolation valves but not for the vacuum breaker isolation valves.

The proposed change reflects the creation of the new Valve Group 9, which includes the RCIC turbine exhaust vacuum breakers E51-F062 and E51-F066. Operability and surveillance requirements for the existing RCIC vacuum breaker-isolation valves are being added under Item 4.b.3 with a new Footnote (j) which states, "Valve isolation dependent upon low steam supply pressure coincident with high drywell pressure." This footnote recognizes the existing logic design and the need for both signals to be present for conditions to exist that require the isolation safety function. The minimum number of operable channels per trip system is one, with each of the two trip systems consisting of one channel. The instrumentation is required to be operable in Operational Conditions 1, 2, and 3 since that is when primary containment integrity and RCIC operability are both required. In the event both trip systems do not meet the operability requirements, existing Action Statement 25, "Close the affected system isolation valves and declare the affected system inoperable," is specified. This is the conservative action if the logic is completely inoperable and is the same action specified for the Group 5 isolation instrumentation.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes specific operability and surveillance requirements for existing instrumentation which was not originally included in the Technical Specifications. The RCIC turbine exhaust vacuum breakers isolate on coincident RCIC steam line pressure - low and drywell pressure - high. The steam line pressure - low signal indicates RCIC is isolated or no longer needed because the reactor is depressurized, and the drywell pressure - high signal indicates a LOCA exists and primary containment is needed.

   This change adds operability requirements for existing instrumentation to provide additional assurance that primary containment is isolated when necessary. No new equipment is being added, and existing equipment will perform its safety function in the same manner as before. Existing requirements are being added to the Technical Specifications for clarity and completeness. Thus, there is no change in the probability of an accident, nor is there any change in the consequences of any accident.

2. The proposed change adds operability requirements to the Technical Specifications for existing instrumentation. This will provide a more complete and accurate document, thereby enhancing primary containment isolation assurance. It does not provide an opportunity for any new accidents because no equipment, operability requirements, or setpoints are being altered. Only references to existing instrumentation are being added.

3. The proposed change adds requirements for existing instrumentation to the Technical Specifications. It does not reflect a physical change to the plant systems; it only provides a more complete and accurate description of existing equipment. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, there is no impact on the margin of safety of the plant.

Proposed Change Number 38

Replace "and" with ";" in the instrument number listing in Item 4.b.5, "Bus Power Monitor," in Table 3.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-16.

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table, and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 39

Replace Instrument Number E51-dTS-N604A,B with Instrument Number E51-TDS-N604A,B in Item 4.b.8, "RCIC Steam Line Area Δ Temperature - High," in Table 3.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-17.

Basis

The isolation instrumentation listed in Item 4.b.8 isolates the reactor core isolation cooling (RCIC) system should there be a RCIC steam line break.

Currently, the Technical Specifications list Instrument Number E51-dTS-N604A,B under Item 4.b.8. The proposed change will revise that number to E51-TDS-N604A,B. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference E51-dTS-N604A,B with E51-TDS-N604A,B. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device and makes its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 4.b.8 of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 40

Replace Instrument Number E51-dTS-N601A,B with Instrument Number E51-TDS-N601A,B in Item 4.b.10, "RCIC Equipment Room Δ Temperature - High," in Table 3.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-17.

Basis

The isolation instrumentation listed in Item 4.b.10 isolates the reactor core isolation cooling (RCIC) system should there be a RCIC steam line break.

Currently, the Technical Specifications list Instrument Number E51-dTS-N601A,B under Item 4.b.10. The proposed change will revise that number to E51-TDS-N601A,B. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference E51-dTS-N601A,B with E51-TDS-N601A,B. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 4.b.10 of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 41

Add new Item 4.b.11, "RCIC Steam Line Tunnel Temperature - High Time Delay Relay," to Table 3.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-17.

Basis

The reactor core isolation cooling (RCIC) steam line tunnel temperature - high time delay relay instrumentation isolates the Group 5 isolation valves in the event of a RCIC steam line break to mitigate the consequences of the break. The Group 5 valves include the RCIC inboard and outboard steam line isolation valves.

The current Technical Specifications do not specifically reference operability and surveillance requirements for the existing RCIC steam line tunnel high temperature time delay relays. The proposed change adds operability requirements for these relays. The proposed requirements are identical to those listed for Item 4.b.7 since the relays are part of that logic and have the same function. Initially, the time delay relays were considered to be a part of the isolation logic described in Item 4.b.7 and were not listed separately in the table. These relays do not initiate any isolation signal; however, they are an important part of the instrumentation and it has been determined that they should be addressed separately in the table to avoid possible confusion.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes specific operability and surveillance requirements for instrumentation that is designed to detect an accident; not prevent one. These requirements are consistent with the accident analysis assumptions and are similar to those for other associated instrumentation. There is no change to the equipment or in how the equipment is operated; only references to existing instrumentation are being added to the table. Since the establishment of operability and surveillance requirements does not change the design or affect the operation of the instrumentation, there is no increase in the probability of an accident, nor is there any change in the consequences of any accident previously evaluated.

2. Specification of the correct operability and surveillance requirements provides additional assurance that the instrumentation will perform its intended safety function. It does not change the design or operation of any equipment or cause it to operate in a manner not previously assumed. It does not affect any equipment which could cause an accident. It only affects instrumentation designed to detect the occurrence of an accident. Thus, this change does not create the possibility of a new accident.

3. Specifying the specific operability and surveillance requirements that are similar to the other isolation instrumentation for this system will not change the operation of the affected equipment. There will be no change in the reliability, response characteristics, or setpoints of the instrumentation. There is no change in the design or capability of the primary containment system, nor a change in the demands placed upon this process barrier. Therefore, there is no change in the margin of safety.

Proposed Change Number 42

Add new Item 4.b.12 "Drywell Pressure - High," and associated Instrument Numbers E11-PT-N011A,B and E11-PTS-N011A-2,B-2 to Table 3.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-17.

Basis

Valve Group 5 isolates the reactor core isolation cooling (RCIC) steam line to mitigate the consequences of a break. The Group 5 isolation valves include the RCIC inboard and outboard steam line isolation valves.

The RCIC system also has turbine exhaust vacuum breaker isolation valves (E51-F062 and E51-F066) on a vacuum relief line for the RCIC turbine exhaust. The line helps prevent the development of a water column in the exhaust line. Preventing development of this column reduces the piping loads which could exist if the turbine is restarted. The valves isolate on coincident RCIC steam line pressure - low and drywell pressure - high signals to establish primary containment isolation. The steam line pressure - low signal indicates RCIC is isolated or no longer needed because the reactor is depressurized. The drywell pressure - high signal indicates a loss of coolant accident (LOCA) exists and primary containment isolation is needed.

The current Technical Specifications specify operability and maintenance requirements for the Group 5 isolation valves but not for the vacuum breaker isolation valves. The proposed change reflects the creation of the new Valve Group 9, which includes the RCIC turbine exhaust vacuum breakers E51-F062 and E51-F066. Operability and surveillance requirements for the RCIC steam supply pressure - low portion of the logic is being added in Item 4.b.3. A new Item 4.b.12 for the drywell pressure - high portion of the logic has also been added. Both additions reference a new Footnote (j) which states, "Valve isolation dependent upon low steam supply pressure coincident with high drywell pressure." This footnote recognizes the existing logic design and the need for both signals to be present for conditions to exist which indicate that the isolation safety function is necessary. The minimum number of operable channels per trip system is one, with each of the two trip systems consisting of one channel. The instrumentation is required to be operable in Operational Conditions 1, 2, and 3 since that is when primary containment integrity and RCIC operability are both required. In the event both trip systems do not meet the operability requirements, existing Action Statement 25, "Close the affected system isolation valves and declare the affected system inoperable," is specified. This is the conservative action if the logic is completely inoperable and is the same action specified for the Group 5 isolation instrumentation.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes specific operability and surveillance requirements for existing instrumentation which was not originally included in the Technical Specifications. The RCIC turbine exhaust vacuum breakers isolate on coincident RCIC steam line low pressure and drywell pressure - high. The steam line pressure - low signal indicates RCIC is isolated or no longer needed because the reactor is depressurized, and the drywell pressure - high signal indicates a LOCA exists and primary containment isolation is needed.

   This change adds operability requirements for existing instrumentation to provide additional assurance that primary containment is isolated when necessary. No new equipment if being added, and existing equipment will perform its safety function in the same manner as before. These requirements are being added to the Technical Specifications for clarity and completeness. Thus, there is no change in the probability of an accident, nor is there any change in the consequences of any accident.

2. The proposed change adds operability requirements to the Technical Specifications for existing instrumentation. This will provide a more complete and accurate document, thereby enhancing primary containment isolation assurance. It does not provide an opportunity for any new accidents because no equipment, operability requirements, or setpoints are being altered.

3. The proposed change adds requirements for existing instrumentation to the Technical Specifications. It does not reflect a physical change to the plant systems; it only provides a more complete and accurate description of existing equipment. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, there is no impact on the margin of safety of the plant.

Proposed Change Number 43

Delete Valve Group 7 from the Valve Group column in Item 5.a, "Reactor Vessel Water Level - Low, Level 1," in Table 3.3.2-1 under Shutdown Cooling System Isolation on Page 3/4 3-17.

Basis

Currently, Technical Specification Table 3.6.3-1 identifies the reactor vessel head spray, residual heat removal (RHR) shutdown cooling supply, and RHR injection isolation valves as Group 8 valves. Table 3.3.2-1 indicates Group 8 valves isolate on reactor vessel water level - low, level 1 (LL1) and reactor steam dome pressure - high.

Table 3.3.2-1 also lists Group 7 valves as having isolation signals from reactor vessel water level - low, level 1, drywell pressure - high, and reactor steam dome pressure - high. However, Table 3.6.3-1 does not identify any valves as Group 7 valves.

The proposed change deletes the reference to Valve Group 7 under "Valve Groups Operated By Signal" in Item 1.a.1 of Table 3.3.2-1. The following factors contributed to the confusion associated with creation of a Valve Group 7.

CUSTOM TO STANDARD TECHNICAL SPECIFICATION CONVERSION

In 1977, BSEP converted from the custom Technical Specification format to the standard format. The Standard Technical Specifications (STS) were requested for both units on August 22, 1977, and issued on November 23, 1977.

As part of this conversion, several changes were made to the valve groups to more completely and consistently describe the isolation signals. Valve Group 2 originally contained those valves which were assumed to isolate during a loss of coolant accident but were not located on high energy lines with their own leak detection. The isolation signals associated with these valves were LL1 and drywell pressure - high. There were also several subgroups within Group 2 that isolated on additional signals. In the conversion from custom to STS, new valve groups were created to recognize these distinctions. Group 6 was created for those valves originally in Group 2 that also isolated on reactor building vent high radiation. Group 8 was created for those valves originally in Group 2 associated with the RHR system that isolate on LL1 and reactor vessel steam dome pressure - high.

LOGIC DESIGN CHANGES

In 1975, General Electric issued Service Information Letter 131, which recommended several changes to the containment isolation logic. Experience had shown that the normal water level drops that occurred during scrams due to collapse of voids exceeded the LL1 setpoint. This caused unnecessary valve group isolations.

Three changes were recommended for BSEP to reduce the complexity of scram recoveries and to reduce demands on the operators caused by unnecessary system isolations and transients so they could concentrate on the scram itself and be more observant for unusual occurrences. These changes were:

1. Lower reactor water cleanup system isolation from LL1 to reactor vessel water level - low, level 2 (LL2),

2. Lower secondary containment isolation and standby gas treatment system initiation from LL1 to LL2,

3. Delete the drywell pressure - high isolation signals for RHR suction and discharge valves.

Plant modification and Technical Specification change packages were prepared to support these logic changes. The original amendment request for BSEP-2 was made on February 2, 1976. The BSEP-2 amendment request was submitted in custom format. On June 22, 1977, a submittal was made that indicated that the same changes were applicable to BSEP-1 as well, and requested that the original change be amended to include BSEP-1. The amendment for BSEP-2 was issued on October 12, 1977. The amendment for BSEP-1 was issued in STS format on April 4, 1979, after conversion to STS format.

Both requests associated with the logic change requested that the drywell pressure - high signal be deleted as an RHR isolation signal. However, the Technical Specification pages originally submitted with the BSEP-2 submittal failed to indicate that the drywell pressure - high signal was also to be deleted for the RHR head spray valves (E11-F022 and E11-F023). This change was encompassed in the logic change; however, the Technical Specification pages were not changed to reflect this.

Apparently, this failure to specifically identify the RHR reactor head spray valves as no longer having the drywell pressure - high signal associated with them, led those preparing the BSEP-2 custom to STS conversion package to assume that they should be treated differently (i.e., no longer as Group 2 valves, but not as Group 8 valves either, as they should have been). The isolation signals associated with the Group 7 valves were assumed to be LL1, reactor steam dome pressure - high and drywell pressure - high. Thus, when valve Groups 6 and 8 were created in the BSEP-2 STS, Valve Group 7 was created specifically for the RHR reactor head spray valves solely because they were thought to still isolate on drywell pressure - high.

Valve Group 7 was created in Table 3.6.3-1 only for BSEP-2 in the original issuance of the STS. BSEP-2 now listed a Valve Group 7 for the E11-F022 and E11-F023 valves in Table 3.6.3-1, and Group 7 as a valve group operated by the LL1, reactor steam dome pressure - high and drywell pressure - high isolation signals in Table 3.2.3-1. When the BSEP-1 STS were issued, there was no reference to Group 7 listed anywhere because the logic modification had not yet been performed.

The BSEP-1 Technical Specification change associated with the logic change was developed from the initial logic change request for BSEP-2. As a result, the BSEP-1 amendment, which was an amendment to the STS, included the Group 7 references in Table 3.2.3-1 as a valve group operated by the LL1, reactor steam

dome pressure - high and drywell pressure - high signals, but did not identify Valve Group 7 and its associated valves in Table 3.6.3-1.

On September 7, 1982, an additional amendment request was submitted which stated that a typographical error had been made in the classification of the RHR head spray valves. The valves were not actually Group 7; they were to be Group 8 valves. This amendment was issued on March 6, 1984 and changed the classification of the valves in Table 3.6.3-1 from Group 7 to Group 8. This actually applied only to BSEP-2, since BSEP-1 never identified a Group 7 in Table 3.6.3-1; however, the amendment was issued for both units. This amendment request <u>did not</u> delete the superfluous references to Valve Group 7 in Table 3.2.3-1. Thus, these references need to be deleted.

<u>10CFR50.92 Evaluation</u>

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change deletes references to a valve group that does not exist, and was never intended to exist. The valves once considered Group 7 valves, E11-F022 and E11-F023, are actually Group 8 valves, and isolate on the same signals that isolate the Group 8 valves. There are no valves associated with the current Valve Group 7, and it has been determined that no valves should be associated with the current Valve Group 7. Since Group 7 does not currently provide any means of protection, this change does not affect the probability or consequences of an accident.

2. Valve Group 7 currently does not include any valves. Therefore, it can perform no safety function, nor hamper any existing safety function. Thus, its deletion does not create the possibility of a new or different kind of accident.

3. Valve Group 7 currently cannot provide any means of protection, because it does not currently include any valves. Its deletion does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, the margin of safety remains unchanged.

Proposed Change Number 44

Place the operability requirements for Valve Groups 2 and 6 on a separate line in Item 5.a "Reactor Vessel Water Level - Low, Level 1" in Table 3.3.2-1 on under Shutdown Cooling System Isolation on Page 3/4 3-17 and specify the correct requirements for the valve group.

Basis

Currently, Item 5.a specifies the same operability requirements for Valve Groups 2, 6, and 8. The existing operability requirements apply only to Group 8, which isolates shutdown cooling system. Action Statement 27, "Deactivate the shutdown cooling supply and reactor vessel head spray isolation valves in the closed position until the reactor steam dome pressure is within the specified limits," applies to the Group 8 Valves while Action Statement 20, "Be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours," applies to Valve Groups 2 and 6.

The proposed change places the Valve Group 2 and 6 information on a separate line with its own action statement (Action 20) under Item 5.a. This makes the information provided more complete and accurate. No change has been made to the design or operation of the instrumentation or equipment; only the way it is referenced in Item 5.a will be changed.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change provides additional information in Item 5.a to make it a more complete and accurate representation of the existing equipment and instrumentation. The logic associated with the instrumentation has not changed, nor has its design or operation. The proposed change references the appropriate operational conditions for Valve Groups 2 and 6 under Item 5.a. The same operability requirements are listed for Groups 2 and 6 in Item 1.a.1 under primary containment isolation. Because this change provides additional information which will present a more accurate description of the existing instrumentation, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of any accident previously evaluated.

2. The proposed change will provide a more complete and accurate description of existing equipment and instrumentation. It does not represent a change in logic, nor does it reflect a change in the instrument design or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change has no effect on the design or operation of the equipment represented. It provides a more appropriate representation of existing equipment. Therefore, there is no impact on the margin of safety.

Proposed Change Number 45

Delete Valve Group 7 from the Valve Group column in Item 5.b, "Reactor Steam Dome Pressure - High," in Table 3.3.2-1 under Shutdown Cooling System Isolation on Page 3/4 3-17.

Basis

Currently, Technical Specification Table 3.6.3-1 identifies the reactor vessel head spray, residual heat removal (RHR) shutdown cooling supply, and RHR injection isolation valves as Group 8 valves. Table 3.3.2-1 indicates Group 8 valves isolate on reactor vessel water level - low, level 1 (LL1) and reactor steam dome pressure - high.

Table 3.3.2-1 also lists Group 7 valves as having isolation signals from reactor vessel water level - low, level 1, drywell pressure - high, and reactor steam dome pressure - high. However, Table 3.6.3-1 does not identify any valves as Group 7 valves.

The proposed change deletes the reference to Valve Group 7 under "Valve Groups Operated By Signal" in Item 1.a.1 of Table 3.3.2-1. The following factors contributed to the confusion associated with creation of a Valve Group 7.

CUSTOM TO STANDARD TECHNICAL SPECIFICATION CONVERSION

In 1977, BSEP converted from the custom Technical Specification format to the standard format. The Standard Technical Specifications (STS) were requested for both units on August 22, 1977, and issued on November 23, 1977.

As part of this conversion, several changes were made to the valve groups to more completely and consistently describe the isolation signals. Valve Group 2 originally contained those valves which were assumed to isolate during a loss of coolant accident but were not located on high energy lines with their own leak detection. The isolation signals associated with these valves were LL1 and drywell pressure - high. There were also several subgroups within Group 2 that isolated on additional signals. In the conversion from custom to STS, new valve groups were created to recognize these distinctions. Group 6 was created for those valves originally in Group 2 that also isolated on reactor building vent high radiation. Group 8 was created for those valves originally in Group 2 associated with the RHR system that isolate on LL1 and reactor vessel steam dome pressure - high.

LOGIC DESIGN CHANGES

In 1975, General Electric issued Service Information Letter 131, which recommended several changes to the containment isolation logic. Experience had shown that the normal water level drops that occurred during scrams due to collapse of voids exceeded the LL1 setpoint. This caused unnecessary valve group isolations.

Three changes were recommended for BSEP to reduce the complexity of scram recoveries and to reduce demands on the operators caused by unnecessary system isolations and transients so they could concentrate on the scram itself and be more observant for unusual occurrences. These changes were:

1. Lower reactor water cleanup system isolation from LL1 to reactor vessel water level - low, level 2 (LL2),

2. Lower secondary containment isolation and standby gas treatment system initiation from LL1 to LL2,

3. Delete the drywell pressure - high isolation signals for RHR suction and discharge valves.

Plant modification and Technical Specification change packages were prepared to support these logic changes. The original amendment request for BSEP-2 was made on February 2, 1976. The BSEP-2 amendment request was submitted in custom format. On June 22, 1977, a submittal was made that indicated that the same changes were applicable to BSEP-1 as well, and requested that the original change be amended to include BSEP-1. The amendment for BSEP-2 was issued on October 12, 1977. The amendment for BSEP-1 was issued in STS format on April 4, 1979, after conversion to STS format.

Both requests associated with the logic change requested that the drywell pressure - high signal be deleted as an RHR isolation signal. However, the Technical Specification pages originally submitted with the BSEP-2 submittal failed to indicate that the drywell pressure - high signal was also to be deleted for the RHR head spray valves (E11-F022 and E11-F023). This change was encompassed in the logic change; however, the Technical Specification pages were not changed to reflect this.

Apparently, this failure to specifically identify the RHR reactor head spray valves as no longer having the drywell pressure - high signal associated with them, led those preparing the BSEP-2 custom to STS conversion package to assume that they should be treated differently (i.e., no longer as Group 2 valves, but not as Group 8 valves either, as they should have been). The isolation signals associated with the Group 7 valves were assumed to be LL1, reactor steam dome pressure - high and drywell pressure - high. Thus, when valve Groups 6 and 8 were created in the BSEP-2 STS, Valve Group 7 was created specifically for the RHR reactor head spray valves solely because they were thought to still isolate on drywell pressure - high.

Valve Group 7 was created in Table 3.6.3-1 only for BSEP-2 in the original issuance of the STS. BSEP-2 now listed a Valve Group 7 for the E11-F022 and E11-F023 valves in Table 3.6.3-1, and Group 7 as a valve group operated by the LL1, reactor steam dome pressure - high and drywell pressure - high isolation signals in Table 3.2.3-1. When the BSEP-1 STS were issued, there was no reference to Group 7 listed anywhere because the logic modification had not yet been performed.

The BSEP-1 Technical Specification change associated with the logic change was developed from the initial logic change request for BSEP-2. As a result, the BSEP-1 amendment, which was an amendment to the STS, included the Group 7 references in Table 3.2.3-1 as a valve group operated by the LL1, reactor steam

dome pressure - high and drywell pressure - high signals, but did not identify Valve Group 7 and its associated valves in Table 3.6.3-1.

On September 7, 1982, an additional amendment request was submitted which stated that a typographical error had been made in the classification of the RHR head spray valves. The valves were not actually Group 7; they were to be Group 8 valves. This amendment was issued on March 6, 1984 and changed the classification of the valves in Table 3.6.3-1 from Group 7 to Group 8. This actually applied only to BSEP-2, since BSEP-1 never identified a Group 7 in Table 3.6.3-1; however, the amendment was issued for both units. This amendment request <u>did not</u> delete the superfluous references to Valve Group 7 in Table 3.2.3-1. Thus, these references need to be deleted.

<u>10CFR50.92 Evaluation</u>

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change deletes references to a valve group that does not exist, and was never intended to exist. The valves once considered Group 7 valves, E11-F022 and E11-F023, are actually Group 8 valves, and isolate on the signals which isolate the Group 8 valves. There are no valves associated with Valve Group 7, and it has been determined that no valves should be associated with the current Valve Group 7. Since Group 7 does not currently provide any means of protection, this change does not affect the probability or consequences of an accident.

2. Valve Group 7 currently does not include any valves. Therefore, it can perform no safety function, nor hamper any existing safety function. Thus, its deletion does not create the possibility of a new or different kind of accident.

3. Valve Group 7 currently cannot provide any means of protection, because it does not currently include any valves. Its deletion does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, the margin of safety remains unchanged.

Proposed Change Number 46

Add reference to a new Footnote (h) for Valve Group 8 in the Valve Group column of Item 5.b, "Reactor Steam Dome Pressure - High," in Table 3.3.2-1 under Shutdown Cooling System Isolation on Page 3/4 3-17.

Basis

The Group 8 isolation valves include those isolation valves in the residual heat removal (RHR) system associated with operation in the shutdown cooling mode (the reactor pressure vessel head spray, shutdown cooling supply, and RHR injection valves). These valves isolate automatically on reactor vessel water level - low, level 2 to minimize the reactor inventory loss should there be a malfunction or rupture of the RHR system while operating in the shutdown cooling mode. The RHR injection valves have a reactor low pressure permissive associated with this isolation signal to limit its operation to only while in shutdown cooling to prevent any impact on the emergency core cooling system (ECCS) injection function. There is also high reactor pressure closure logic associated with the reactor pressure vessel head spray and shutdown cooling supply valves to prevent over-pressurization of the RHR system by the reactor.

The current Technical Specifications imply that reactor steam dome pressure - high is an isolation signal for all Group 8 isolation valves. The proposed change adds a reference to a new Footnote (h), which states "Does not isolate E11-F015A,B," for the Group 8 valves. The E11-F015A,B valves are the RHR injection valves. The reactor steam dome pressure - high isolation signal functions to prevent over-pressurization of the RHR system by the reactor. This isolation signal is not necessary for the RHR injection valves E11-F015A,B because its setpoint is 140 psig and RHR is required to inject up to 410 psig. This logic would, therefore, detrimentally affect the ECCS function. Over-pressurization protection for these valves is provided by the ECCS injection logic. This logic prevents opening the injection pathway if reactor pressure exceeds 410 psig, thereby eliminating the need for the isolation signal specified by Item 5.b.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed revision to the Technical Specifications does not change the approved design or operation of the RHR injection valves. The necessary over-pressurization protection is provided not by this isolation signal, but by the ECCS injection logic itself. The ECCS logic provides the necessary protection and fulfills the design requirement. Thus, this correction to the Technical Specifications does not change the reliability or function of equipment designed to protect against over-pressurization; it more clearly defines the proper actions for the RHR injection valves. Therefore, there is no increase in the probability of any accident previously evaluated. And, since there is no change in mitigative function of the instrumentation, there is no change in the consequences of any accident.

2. The proposed change does not eliminate any safety function; it recognizes that protection against excessive pressurization of the RHR injection line is provided by other instrumentation. There is no change in the design or operation of the equipment and the existing equipment provides the necessary protection. Therefore, no new accident possibilities are created.

3. This change establishes operability and surveillance requirements that are consistent with the accident analysis assumptions. There is no change in the setpoints or required operating characteristics of the instrumentation. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, there is no impact on the margin of safety.

Proposed Change Number 47

Replace Instrument Number B32-PS-N018A,B with Instrument Number B32-PS-N018A-1,B in Item 5.b, "Reactor Steam Dome Pressure - High," in Table 3.3.2-1 under Shutdown Cooling System Isolation on Page 3/4 3-17.

Basis

The isolation instrumentation listed in Item 5.b isolates the shutdown cooling system should there be a malfunction or rupture of the residual heat removal system while operating in shutdown cooling mode. This minimizes reactor inventory loss.

Currently, the Technical Specifications list Instrument Number B32-PS-N018A,B under Item 5.b. The proposed change will revise that number to B32-PS-N018A-1,B. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference B32-PS-N018A,B with B32-PS-N018A-1,B. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new or different accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 5.b Table 3.3.2-1. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 48

Put Footnotes a through g in parentheses on Pages 3/4 3-18 and 3/4 3-19, (BSEP-2 only) and double space footnotes (BSEP-1 only).

Basis

The proposed change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnotes has not changed unless noticed elsewhere in this enclosure. The footnotes have been put into parentheses to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnotes. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Proposed Change Number 49

Replace current Footnote (a), which states, "See Specification 3.6.3.1, Table 3.6.3.1-1 for valves in each valve group," with "Refer to plant procedure _________ for valves in each valve group," on Page 3/4 3-18.

Basis

Technical Specification Section 3.6 addresses containment systems. Table 3.6.3-1 provides a partial listing of the automatic primary containment isolation valves (PCIVs). The Brunswick Plant Technical Specifications currently require the following surveillances:

> Specification 4.6.3.1 requires that primary containment isolation valves be demonstrated operable by cycling and isolation time testing following maintenance, repair, or replacement work.
>
> Specification 4.6.3.2 requires verification of isolation valve operability through the application of an isolation test signal to verify the valve actuates properly. This testing is required every 18 months.
>
> Specification 4.6.3.3 requires verification of the isolation time for power-operated and automatic valves.
>
> Specification 4.6.4.4 requires that reactor instrumentation system isolation valves be demonstrated operable by cycling the valve every 18 months.
>
> Specification 4.6.1.2.d requires Type B and C leak testing of primary containment isolation valves.

The Company has completed a review of the primary containment isolation system (PCIS) Technical Specifications and determined that certain changes to Table 3.6.3-1 are needed to accurately reflect the as-built design of the primary containment isolation system. However, rather than propose changes to the table, CP&L believes it is more appropriate to request removal of Table 3.6.3-1 from the Technical Specifications and replace references to the table with references to valve listings contained in plant procedures (the Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal). Our reasons for this determination are described below.

First, removal of the valve listings from the Technical Specifications is appropriate in the context of Technical Specification reform. The recent industry and NRC efforts to improve the Technical Specifications have identified component listings as items for removal from the Technical Specifications because they do not directly contribute to plant safety. In addition, the level of detail of such listings has resulted in frequent requests for license amendments. This process has resulted in inherent inaccuracies in the Technical Specifications due to the delays in receiving license amendments brought about by the current regulatory environment. The NRC has already recommended and approved removal of the analogous snubber listing from the Technical Specifications. In addition, tables relating to fire protection devices and containment penetration protection circuit breakers have been removed from the Technical Specifications for other dockets. For example, listings relating to containment isolation valves, snubbers, containment penetration protection devices, motor operated valve thermal overload

protection, the reactor vessel material surveillance program, engineered safety features actuation system instrumentation response times, and reactor trip system instrumentation response times have been removed from the Technical Specifications for the Shearon Harris Plant and references to plant procedures incorporated. The use of component listings through reference to plant procedures ensures that timely information is maintained that reflects recent plant modifications.

Second, changes to these plant procedures are strictly controlled under 10CFR50.59. In addition, the Technical Specifications requirement to maintain primary containment integrity (Technical Specification 3.6.1.1), as well as ASME Code Section XI testing requirements (Technical Specification 4.0.5) and 10CFR50, Appendix J requirements (Technical Specification 3.6.1.2) are maintained. Only the listing of valves would be subject to licensee control under 10CFR50.59. Plant procedures are auditable and subject to inspection by the NRC.

Third, appropriate mechanisms and controls exist to ensure correct testing for the valves through plant procedures. As discussed above, these referenced procedure(s) will incorporate valve list information developed to meet the requirements of Technical Specifications 4.6.3.1, 4.6.3.2, 4.6.3.3 and 4.6.3.4. In addition, Appendix J of 10CFR50 (implemented through Technical Specification 3/4.6.1.2) specifies leak rate test requirements for containment isolation valves, and thus maintenance of a list of such valves. Section XI of the ASME Code (implemented through Technical Specification 4.0.5) requires once per cycle testing of valve closure and timing for certain of the valves currently listed in Table 3.6.3-1. In addition, Technical Specification 4.3.2.2 requires once per cycle performance of a logic system functional test for most of the valves currently listed in Table 3.6.3-1. All the above requirements are appropriately integrated into plant procedure(s) which are subject to change only under requirements of 10CFR50.59. These plant procedures cover testing (to or beyond the present Technical Specification requirements) for the valves currently included in Table 3.6.3-1. A complete listing of containment isolation valves and required closure times is also provided as Table 6.2.4-2 to the Brunswick Updated Final Safety Analysis Report. References to plant procedures will replace references to Table 3.6.3-1 in the text of the Technical Specifications. Specific closure time testing requirements for the main steam isolation valves will be maintained in the Technical Specifications under the existing Specification 3/4.4.7.

<u>10CFR50.92 Evaluation</u>

The proposed change does not involve a significant hazards consideration for the following reasons:

1. Removal of the primary containment isolation valve listing from the Technical Specifications is appropriate in the context of Technical Specification reform. Recent industry and NRC efforts to improve the Technical Specifications have identified component listings as an item for removal because they do not directly contribute to plant safety. The use of component listings through reference to plant procedures will ensure that timely information is maintained that reflects recent plant modifications. Changes to these procedures will be controlled under the requirements of 10CFR50.59. Primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specifications 3.6.1.1

and 3.6.3 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3 (and thus the proper identification, control, and periodic testing of the appropriate containment isolation valves). Relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not change their design or operation, nor does it affect any of the accident analyses; therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. As discussed in Item 1 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3 (and thus the proper identification, control, and periodic testing of the appropriate containment isolation valves). The design and operation of these valves does not change. Thus, relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. As discussed in Items 1 and 2 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3. Furthermore, the relocation of the primary containment isolation valve listing to plant procedures will clarify the Technical Specifications and ensure a more up to date and accurate primary containment isolation valve listing, thereby avoiding possible operator confusion. There is no change in the containment design or its pressure retaining capability. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Proposed Change Number 50

Delete Footnote (h) from Page 3/4 3-19 (BSEP-2 only).

Basis

Footnote (h) was added to the Technical Specifications via Amendment 131 on December 10, 1986 to support the hydrogen injection test which took place in January, 1987. This was a one-time test; therefore, the footnote is no longer applicable or necessary for normal operation.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change deletes a footnote which no longer applies. The footnote was added to support a one-time, hydrogen injection test, which was completed on January 5, 1987. No additional testing is planned. Thus, the proposed change has no effect on the probability of an accident, nor does it affect the consequences of any accident.

2. The referenced footnote no longer applies to BSEP-2. The hydrogen injection test was successfully completed on January 5, 1987. Thus, this footnote is no longer necessary, and deletion of it will not create the possibility of a new or different type of accident.

3. Footnote (h) was added to support a one-time hydrogen injection test, which was completed on January 5, 1987. No additional testing is planned; therefore, the footnote no longer applies and should be deleted. This deletion has no impact on the margin of safety.

Proposed Change Number 51

Add new Footnote (h), "Does not isolate E11-F015A,B," to the footnote on Page 3/4 3-19.

Basis

The Group 8 isolation valves include those isolation valves in the residual heat removal (RHR) system associated with operation in the shutd[illegible] cooling mode (the reactor pressure vessel head spray, shutdown cooling suppl[illegible]d RHR injection valves). These valves isolate automatically on reactor v[illegible] water level - low, level 2 to minimize the reactor inventory loss should there be a malfunction or rupture of the RHR system while operating in the shutdown cooling mode. The RHR injection valves have a reactor low pressure permissive associated with this isolation signal to limit its operation to only while in shutdown cooling to prevent any impact on the emergency core cooling system (ECCS) injection function. There is also high reactor pressure closure logic associated with the reactor pressure vessel head spray and shutdown cooling supply valves to prevent over-pressurization of the RHR system by the reactor.

The current Technical Specifications imply that reactor steam dome pressure - high is an isolation signal for all Group 8 isolation valves. The proposed change adds a reference to a new Footnote (h) which states, "Does not isolate E11-F015A,B," for the Group 8 valves. The E11-F015A,B valves are the RHR injection valves. The reactor steam dome pressure - high isolation signal functions to prevent over-pressurization of the RHR system by the reactor. This isolation signal is not necessary for the RHR injection valves E11-F015A,B because their setpoint is 140 psig and RHR is required to inject up to 410 psig. This logic would, therefore, detrimentally affect the ECCS function. Over-pressurization protection for these valves is provided by the ECCS injection logic. This logic prevents opening the injection pathway if reactor pressure exceeds 410 psig, thereby eliminating the need for the isolation signal specified by Item 5.b.

10CFR50.92 Evaluation

1. The proposed revision to the Technical Specifications does not change the approved design or operation of the RHR injection valves. The necessary over-pressurization protection is provided not by this isolation signal, but by the ECCS injection logic itself. The ECCS logic provides the necessary protection and fulfills the design requirement. Thus, this correction to the Technical Specifications does not change the reliability or function of equipment designed to protect against over-pressurization; it more clearly defines the proper actions for the RHR injection valves. Therefore, there is no increase in the probability of any accident previously evaluated. And, since there is no change in mitigative function of the instrumentation, there is no change in the consequences of any accident.

2. The proposed change does not eliminate any safety function; it recognizes that protection against excessive pressurization of the RHR injection line is provided by other instrumentation. There is no change in the design or operation of the equipment and the existing equipment provides the necessary protection. Therefore, no new accident possibilities are created.

3. This change establishes operability and surveillance requirements consistent with the accident analysis assumptions. There is no change in the setpoints or required operating characteristics of the instrumentation. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, there is no impact on the margin of safety.

Proposed Change Number 52

Add new Footnote (i), "Does not isolate B32-F019 or B32-F020," to the footnote list on Page 3/4 3-19.

Basis

The reactor water sample line valves, B32-F019 and B32-F020, isolate on reactor vessel water level - low and main steam line radiation - high. The remaining Group 1 isolation signals have no bearing on the reactor water sample line; they would not detect a reactor water sample line rupture, nor would isolation of the reactor water sample line serve to mitigate or prevent a main steam line break outside containment. This is because of the small diameter of the sample line, and the fact that it is routed separately from the main steam lines.

The current Technical Specifications incorrectly indicate that the B32-F019 and B32-F020 reactor water sample isolation valves isolate automatically on all of the Group 1 isolation signals. It should show that these valves isolate only on reactor vessel water level - low or main steam line radiation - high.

This change adds a reference to a new Footnote (i) to Items 1.c.2, 1.c.3, 1.c.4 (BSEP-2 only), 1.d, 1.e, and 1.f. The footnote states "Does not isolate B32-F019 or B32-F020." This would correct the current information which implies that the B32-F019 and B32-F020 valves isolate on main steam line pressure - low, main steam line flow - high, main steam line tunnel temperature - high, condenser vacuum - low, and turbine building area temperature - high signals.

This change does not represent any physical changes to the instruments themselves; it only provides a more adequate description of the existing instrumentation logic and its function.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The Group 1 isolation signals function to prevent, detect and/or mitigate a main steam line break outside containment. The only signals that isolate the reactor water sample line are the reactor vessel water level - low or main steam line radiation - high. The remaining Group 1 signals would not detect a need to isolate the reactor water sample line, nor would isolation of this line mitigate or prevent a main steam line break outside containment.

   The proposed change does not represent a change in instrumentation or logic; it merely provides a more adequate description of the existing instrumentation and its function. The existing design of the system is more clearly defined by the proposed change thereby reducing confusion about the operation of the system. It is sufficient for accomplishing the required safety function. Therefore, there is no change in the probability of occurrence or the consequences of a pipe break outside containment, which is the only accident to which this change could apply.

2. Failure of the reactor water sample line is bounded by the main steam line break accident. The Group 1 isolation signals, with the exception of the reactor vessel water level - low and main steam line radiation - high signals, would not detect and are not needed to isolate the reactor water sample line when necessary. No instrumentation is being changed; it is merely being referenced in a more accurate manner. Therefore, the possibility of a new accident is not created by this change.

3. The Group 1 isolation signals, except for reactor vessel water level - low and main steam line radiation - high, would not detect the need to isolate the reactor water sample line. Isolation of the line would not mitigate the consequences of a main steam line break. These signals do not cause isolation of the reactor water sample line, and never have. Therefore, these isolation signals, when associated with Valves B32-F019 and B32-F020, do not accomplish any safety function. Thus, they do not contribute to the margin of safety and their elimination from the Technical Specifications would not change the margin of safety. The automatic isolation of the reactor water sample line on the existing two trips are the ones which are necessary and sufficient to ensure isolation of Valves B32-F019 and B32-F020 when required.

Proposed Change Number 53

Add new Footnote (j), "Valve isolation depends upon low steam supply pressure coincident with high drywell pressure," to the footnote list on Page 3/4 3-19.

Basis

Footnote (j) reflects the existing logic design associated with the reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) turbine exhaust vacuum breaker isolation valves and the need for both the steam supply pressure - low and drywell pressure - high signals to be present for conditions to exist for the isolation safety function to be necessary.

The current Technical Specifications do not contain operability requirements for the turbine exhaust vacuum breaker isolation valves, which are described by two new valve groups; Valve Groups 7 and 9, respectively. They are being added to the Technical Specifications as described in Items 4.a.3 and 4.b.3 of Table 3.3.2-1. This footnote applies only to those valves, and is being added in conjunction with those changes.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes specific operability and surveillance requirements for existing instrumentation not originally included in the Technical Specifications. The HPCI and RCIC turbine exhaust vacuum breakers isolate on coincident HPCI or RCIC steam line low pressure and drywell pressure - high signals. This footnote reflects this information.

   The proposed change adds operability requirements to the Technical Specifications for existing instrumentation to provide additional assurance that primary containment is isolated when necessary. No new equipment is being added and the existing equipment will perform its safety function in the same manner as before. Thus, there is no change in the probability or consequences of an accident.

2. The proposed change adds operability requirements to the Technical Specifications for existing instrumentation. This will provide a more complete and accurate document, thereby enhancing primary containment isolation assurance. It does not provide an opportunity for any new accidents because no equipment, operability requirements, or setpoints are being altered.

3. The proposed change adds requirements for existing instrumentation to the Technical Specifications. It does not reflect a physical change to the plant systems; it only provides a more complete and accurate description of existing equipment. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, there is no impact on the margin of safety of the plant.

Proposed Change Number 54

Add new Footnote (k), "Secondary containment isolation dampers as listed in plant procedure ________," to the footnote list on Page 3/4 3-19.

Basis

A secondary containment isolation signal closes the secondary containment isolation dampers, thereby establishing secondary containment isolation and minimizing the release of radiation from the reactor building during an accident. The dampers work in conjunction with the standby gas treatment system to establish a negative pressure in the reactor building and a controlled vent path through the standby gas treatment system filters and the stack. Secondary containment isolation would occur during a loss of coolant accident (LOCA) due to a reactor vessel water level - low, level 2 or drywell pressure - high signal. This would minimize the release of radioactivity from the reactor building. Secondary containment isolation is also assumed to occur during a refueling accident due to a signal from the reactor building exhaust radiation - high instrumentation to contain radiation released in a fuel handling accident.

Current, Technical Specification Items 2.a, 2.b, and 2.c do not reference the dampers which, in fact, establish secondary containment. A reference to Footnote (k) which states, "Secondary containment isolation dampers as listed in plant procedure ______,"is being added to each of these items to provide the needed reference to the secondary containment isolation dampers.

The secondary containment isolation dampers are currently listed in Table 3.6.5.2-1 of the Technical Specifications. This table is being deleted and the information relocated in a plant procedure, as described elsewhere in this submittal.

Addition of this footnote is necessary because there is no reference to the secondary containment isolation dampers under the secondary containment isolation section of Table 3.3.2-1. This change will simplify the Technical Specifications, eliminate discrepancies, and reduce the potential for misinterpretation. The plant procedure which will list the secondary containment isolation dampers once they are removed from the Technical Specifications is currently being developed. The Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazard for the following reasons:

1. The proposed change provides a more accurate reference to the appropriate secondary containment isolation dampers covered by Items 2.a, 2.b and 2.c. The current Technical Specifications list valve groups which are isolated by the appropriate signals, but do not establish secondary containment isolation. The dampers which establish secondary containment isolation will be listed in a plant procedure which will be identified at a later date.

There is no change in the design, operation, reliability, or testing requirements of existing plant systems. This change provides a more complete reference to existing plant equipment in its appropriate place in the table. Thus, there is no change in the probability of occurrence of an accident, nor is there any change in the consequences of any accident previously evaluated.

2. The proposed change provides a more accurate reference to the appropriate valves covered by Items 2.a, 2.b and 2.c. The current Technical Specifications list valve groups which do not establish secondary containment isolation. The dampers which establish secondary containment isolation will be listed in a plant procedure which will be identified at a later date.

   There is no change in the design, operation, reliability, or testing requirements of existing plant systems. This change provides a more complete reference to existing plant equipment in an appropriate place in plant procedures. Thus, no new or different types of accident situations are created.

3. The proposed change provides references to the appropriate dampers covered by Items 2.a, 2.b and 2.c. This change adds a reference to the dampers which actually provide secondary containment isolation. There is no change in the response, capability, reliability, or testing requirements of plant systems. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, there is no effect on the margin of safety.

Proposed Change Number 55

Replace Footnote * with Footnote (a) in Item 1.a, "Reactor Vessel Water Level" in Table 3.3.2-2 under Primary Containment Isolation on Page 3/4 3-20 (BSEP-1 only).

Basis

The proposed change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnotes has not changed unless specified elsewhere in this enclosure. The footnote has been changed from * to (a) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnote. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Proposed Change Number 56

Delete Item 1.a.2, "Reactor Vessel Water Level - Low, Level 2," from Table 3.3.2-2 on Page 3/4 3-20 and re-label Item 1.a.3 as Item 1.a.2 (BSEP-2 only).

The Technical Specification change request submitted on September 29, 1987, as supplemented on October 14, 1987 and November 24, 1987, revised the reactor vessel water level trip function for the Valve Group 1 isolation valves from reactor vessel water level - low, level 2 to reactor vessel water level - low, level 3. This resulted in only Valve Group 3 being actuated by the reactor vessel water level - low, level 2 trip function. Valve Group 3 isolates the reactor water cleanup system and is addressed specifically in Item 3.e for the reactor vessel water level - low level 2 instrumentation. The proposed change deletes Item 1.a.2 because the instrumentation no longer actuates any valve groups that need to be addressed under Item 1. This change does not represent any physical change to the design or operation of any systems. It only more accurately describes the trip function associated with the Group 3 valves.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change deletes information which no longer belongs under Item 1. Via a previous submittal, the trip function for the Group 1 isolation valves was revised from reactor vessel water level - low, level 2 to reactor vessel water level - low, level 3. The information associated with the remaining valve group actuated by that trip function, Valve Group 3, is more appropriately referenced in Item 3.e, which describes the instrumentation that actuates reactor water cleanup system isolation. The proposed change does not reflect a change to the design or operation of the instrumentation and valve groups; it only clarifies existing information in the table. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change clarifies the table by placing the information associated with the reactor vessel water level - low, level 2 instrumentation in its appropriate place. It does not reflect a change in the design or operation of the system. Therefore, it does not create the possibility of a new or different accident.

3. The proposed change does not reflect a change to the design or operation of any equipment. It merely provides the applicable references to the reactor vessel water level - low, level 2 instrumentation in the appropriate place in the table. Therefore, there is no impact on the margin of safety.

Proposed Change Number 57

Replace "and" with ";" in the instrument number listings in Item 1.a.2, "Reactor Vessel Water Level - Low, Level 2," (BSEP-1 only) and "Reactor Vessel Water Level - Low, Level 3" (BSEP-2 only) in Table 3.3.2-2 under Primary Containment Isolation on Page 3/4 3-20.

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table, and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability or the primary containment isolation system, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 58

Delete Footnote (b) from Item 1.c.1, "Main Steam Line Radiation - High," in Table 3.3.2-2 under Primary Containment Isolation on Page 3/4 3-20 (BSEP-2 only).

Basis

Footnote (b) was added to the Technical Specifications via Amendment 131 on December 10, 1986 to support the hydrogen injection test which took place in January, 1987. This was a one-time test; therefore, the footnote is no longer applicable or necessary for normal operation.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change deletes a footnote which no longer applies. The footnote was added to support a one-time, hydrogen injection test, which was completed on January 5, 1987. No additional testing is planned. Thus, the proposed change has no effect on the probability of an accident, nor does it affect the consequences of any accident.

2. The referenced footnote no longer applies to BSEP-2. The hydrogen injection test was successfully completed on January 5, 1987. Thus, this footnote is no longer necessary, and deletion of it will not create the possibility of a new or different type of accident.

3. Footnote (b) was added to support a one-time hydrogen injection test, that was completed on January 5, 1987. No additional testing is planned; therefore, the footnote no longer applies and should be deleted. Thus, this deletion has no impact on the margin of safety.

Proposed Change Number 59

Add new Item 1.g, "Reactor Building Exhaust Radiation - High," and associated Instrument Number D12-RM-K609A,B to Table 3.3.2-2 under Primary Containment Isolation on Page 3/4 3-21.

Basis

The reactor building exhaust radiation - high isolation signal causes the Group 6 isolation valves, which include the containment atmospheric control valves, and certain containment atmospheric monitoring and post-accident sampling system valves, to close during a loss of coolant accident (LOCA). This minimizes the amount of radiation released from primary containment under LOCA conditions. This signal is not of primary importance for Group 6 isolation; the reactor vessel water level - low, level 2 and drywell pressure - high signals provide Group 6 isolation signals much earlier in a LOCA scenario. These two signals directly detect a LOCA, while the reactor building exhaust radiation - high instrumentation detects radiation released from primary containment to the reactor building. It does not directly detect a LOCA and will isolate Group 6 much later than the other two signals will.

Item 1 provides information concerning isolation of the main steam lines, drywell drains, and the containment atmospheric control and monitoring systems. The main steam lines are isolated by Group 1 valves, the drywell drains and associated systems by the Group 2 valves, and the containment atmospheric control and monitoring systems by the Group 6 valves.

The new Item 1.g provides information relating to the reactor building exhaust radiation - high instrumentation trip setpoints. This instrumentation consists of two channels; one per trip system, either of which can initiate a Group 6 isolation. The radiation monitors listed are located in the reactor building exhaust plenum and monitor normal HVAC effluent from secondary containment in the reactor building.

Item 1.g provides the appropriate setpoint requirements for isolation of the Group 6 valves on a signal from the reactor building exhaust radiation - high instrumentation. They are the same as those signals specified under Item 2.a since the same instrumentation actuates both groups.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes trip setpoint requirements for instrumentation which is designed to detect accident conditions, not prevent an accident. These requirements are consistent with the accident analysis assumptions and are the same as for other equipment actuated by this instrumentation. There is no change in how the equipment is operated. This change establishes trip setpoint requirements for existing equipment and does not change the design or affect the operation of the instrumentation. Thus, there is no increase in the probability of an accident.

The proposed change adds trip setpoint requirements which are appropriate for the Group 6 isolation valves and their related safety function. The existing requirements in the Technical Specifications are listed under the Secondary Containment section, and are not directly applicable to the Group 6 valves. The reactor building exhaust radiation - high signal is not depended upon for Group 6 isolation for any design basis accident; thus, there is no impact on the consequences of any accident.

2. Specifying the correct trip setpoint requirements provides additional assurance that the instrumentation will perform its intended safety function. It does not change the design or operation of any equipment or cause it to operate in a manner not previously evaluated. It does not affect any equipment which could cause an accident. It only affects existing instrumentation designed to detect accident conditions. Therefore, it does not create the possibility of a new or different type of accident.

3. Specifying the correct trip setpoint requirements does not affect any equipment which could cause an accident. It only affects instrumentation designed to detect radioactivity in secondary containment and is not depended upon by this valve group for mitigating any design basis accident. Therefore, it is not a factor in the margin of safety. The instrumentation is designed to detect and mitigate the design basis accidents is not affected by this change and will have the same reliability and response characteristics.

Proposed Change Number 60

Replace Instrument Number D12-RM-N010A,B with Instrument Number D12-RM-K609A,B in Item 2.a, "Reactor Building Exhaust Radiation - High," in Table 3.3.2-2 under Secondary Containment Isolation on Page 3/4 3-21.

Basis

The secondary containment isolation dampers isolate the reactor building HVAC system to establish secondary containment under accident conditions to minimize the release of radiation from the reactor building. They work in conjunction with the standby gas treatment system to establish a negative pressure in the reactor building an a controlled vent path through the standby gas treatment system filters and the stack. The dampers isolate on reactor vessel water level - low, level 2, drywell pressure - high, or reactor building exhaust vent radiation - high isolation signals.

The current Technical Specifications incorrectly references Instrument D12-RM-N010A,B as the instrument which provides a reactor building exhaust radiation - high isolation signal. This instrument is a radiation detector and does not itself have a setpoint. The setpoint is associated with the radiation monitor drawer for this instrument loop. This is designated by instrument number D12-RM-K609A,B. The devices which perform the designated function are not altered in any way by this change; the appropriate instrumentation is more clearly identified in the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change revises an instrument number reference so the appropriate instrumentation is referenced in the Technical Specifications. It does not change the devices which perform the function; it more completely and clearly references them. This change may reduce the possibility of misinterpretation of the intent of the Technical Specifications. Therefore, there is no increase in the probability of any accident previously evaluated, nor is there any change in the consequences of any accident previously evaluated.

2. The proposed change corrects an instrument number reference to provide consistency with other plant documentation and provide reference to the appropriate instrumentation. The instruments themselves will perform as before to mitigate the consequences of an accident. Therefore, no new accident possibilities are created.

3. The proposed change does not physically change any plant equipment; it merely references the appropriate instrumentation in its appropriate place in the Technical Specifications. There is no change in the operability, reliability, or testing requirements of plant systems; therefore, there is no impact on the margin of safety.

Proposed Change Number 61

Replace Footnote * with (a) in Item 2.c, "Reactor Vessel Water Level - Low, Level 2," in Table 3.3.2-2 under Secondary Containment Isolation on Page 3/4 3-21.

Basis

This change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnotes has not changed unless noticed elsewhere in this enclosure. The footnote has been changed from * to (a) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnotes. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnote. Therefore, there is no impact on the margin of safety.

Proposed Change Number 62

Replace "and" with ";" in the instrument number listings in Item 2.c, "Reactor Vessel Water Level - Low, Level 2," in Table 3.3.2-2 under Secondary Containment Isolation on Page 3/4 3-21.

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table, and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability of the primary containment system nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 63

Replace Instrument Number G31-dFS-N603-1A,1B with Instrument Number G31-FDS-N603-1A,1B in Item 3.a, "Δ Flow - High," in Table 3.3.2-2 under Reactor Water Cleanup System Isolation on Page 3/4 3-22.

Basis

The isolation instrumentation listed in Item 3.a isolates the reactor water cleanup system supply valves in the event of a rupture of the reactor water cleanup system. This minimizes the amount of radioactivity, energy, and inventory lost from the reactor vessel.

Currently, the Technical Specifications list Instrument Number G31-dFS-N603-1A,1B under Item 3.a. The proposed change will revise that number to G31-FDS-N603-1A,1B. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference G31-dFS-N603-1A,1B with G31-FDS-N603-1A,1B. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device, and makes its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new or different type of accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 3.a of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment system nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 64

Replace "Area Ventilation Temperature Δ Temp - High" with "Area Ventilation Δ Temperature - High" in the title of Item 3.c in Table 3.3.2-2 under Reactor Water Cleanup System Isolation on Page 3/4 3-22.

Basis

The reactor water cleanup system (RWCU) isolation actuation instrumentation isolates the RWCU supply isolation valves in the event of a rupture of the RWCU system. This minimizes the amount of radioactivity, energy and inventory lost from the reactor vessel under such conditions. A potential rupture is sensed by the following signals; high delta supply/return RWCU flow, high RWCU area temperature, high delta supply/return ventilation temperature, or reactor vessel water level - low level 2. The system also isolates on standby liquid control system (SLC) initiation to prevent dilution of the poison injected by SLC. The high delta supply/return RWCU flow instrumentation measures the difference between the supply and return flow rate and assumes an excessive difference indicates a rupture. A 45 second time delay is provided for the Δ flow - high instrumentation to prevent spurious isolation signals which can result when starting a RWCU pump or changing the flow path.

The current Technical Specifications have an administrative error; the word "Temperature" is use twice in the title of Item 3.c. Only one reference to the temperature is necessary. The title of Item 3.c should be "Area Ventilation Δ Temperature - High." This change does not reflect a change to the design, operation, reliability, or testing requirements of the instrumentation; it only provides a more concise definition of the trip function.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change replaces the title of Item 3.c with one that provides a better description of the instrumentation. This change does not involve any change to the operability, reliability, or testing requirements of the instrumentation involved; it merely provides a more appropriate description of its function. Therefore, there is no increase in the probability of any accident previously evaluated, nor is there any change in the consequences of any accident.

2. The proposed change does not involve any physical changes to plant systems. It provides a better description in the Technical Specifications of the actual function of the area delta temperature monitoring instrumentation. The possibility of misinterpretation of the function of the instrumentation will be reduced. The equipment will still function in the same manner. Therefore, there are no new accident possibilities created.

3. The proposed change provides a more clear, concise description of the actual function of the area delta temperature monitoring instrumentation. It does not physically alter any plant instrumentation and, therefore, does not impact the margin of safety.

Proposed Change Number 65

Replace Instrument Number G31-TS-N602A,B,C,D,E,F with Instrument Number G31-TDS-N602A,B,C,D,E,F in Item 3.c, "Area Ventilation Δ Temperature - High," in Table 3.3.2-2 under Reactor Water Cleanup System Isolation on Page 3/4 3-22.

Basis

The isolation instrumentation listed in Item 3.c isolates the reactor water cleanup system supply valves in the event of a rupture of the reactor water cleanup system. This minimizes the amount of radioactivity, energy, and inventory lost from the reactor vessel.

Currently, the Technical Specifications list Instrument Number G31-TS-N602A,B,C,D,E,F under Item 3.c. The proposed change will revise that number to G31-TDS-N602A,B,C,D,E,F. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference G31-TS-N602A,B,C,D,E,F with G31-TDS-N602A,B,C,D,E,F. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new or different type of accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 3.c of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment system nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 66

Replace "and" with ";" in the instrument number listings in Item 3.e, "Reactor Vessel Water Level - Low, Level 2," in Table 3.3.2-2 under Reactor Water Cleanup System Isolation on Page 3/4 3-22.

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table, and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability of the primary containment system nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 67

Replace * with (a) in Item 3.e, "Reactor Vessel Water Level - Low, Level 2," in Table 3.3.2-2 under Reactor Water Cleanup System Isolation on Page 3/4 3-22.

Basis

The proposed change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnote has not changed unless noticed elsewhere in this enclosure. The footnote has been changed from * to (a) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnotes. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Proposed Change Number 68

Add new Item 3.f, "Δ Flow - High - Time Delay Relay," and associated Instrument Number B31-R616C,D to Table 3.3.2-2 under Reactor Water Cleanup System Isolation on Page 3/4 3-22.

Basis

The reactor water cleanup (RWCU) system isolation instrumentation isolates the RWCU supply isolation valves in the event of a rupture of the RWCU system. This minimizes the amount of radioactivity, energy, and inventory lost from the reactor vessel. A potential rupture is sensed by any of the following signals: high delta supply/return RWCU flow, high RWCU area temperature, high delta supply/return ventilation temperature, or reactor vessel water level - low, level 2. The system also isolates on standby liquid control system initiation to prevent dilution of the poison injected due to cleanup from RWCU operation.

The current Technical Specifications do not specifically reference setpoint requirements for the existing time delay relays in the RWCU high delta flow logic. The proposed change adds setpoint requirements for these relays which reflect the established value and are in agreement with the accident analysis. Initially, the time delay relays were considered to be a part of the isolation logic described in Item 3.a and were not listed separately in the table. These relays do not initiate any isolation signal; however, they are an important part of the instrumentation. It has been determined that they should be addressed separately in the table to avoid possible confusion.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes setpoint requirements for instrumentation which is designed to detect an accident and not prevent one. These requirements are consistent with the accident analysis assumptions. There is no change to the equipment or in how the equipment is operated; only references to existing instrumentation are being added to the table. Since the establishment of operability and surveillance requirements does not change the design or affect the operation of the instrumentation, there is no increase in the probability of an accident, nor is there any change in the consequences of any accident previously evaluated.

2. Specification of the correct operability and surveillance requirements provides additional assurance that the instrumentation will perform its intended safety function. It does not change the design or operation of any equipment or cause it to operate in a manner not previously assumed. It does not affect any equipment which could cause an accident. It only affects instrumentation designed to detect the occurrence of an accident. Thus, this change does not create the possibility of a new accident.

3. Specification of setpoint requirements that reflect the established value and are consistent with the accident analysis does not affect or change the operation of the affected equipment. It does not change the design or capability of the primary containment system nor does it change the demands placed upon this process barrier. Therefore, there is no change in the margin of safety.

Proposed Change Number 69

Replace "HPCI Steam Line High Flow Time Delay Relay" with "HPCI Steam Line Flow - High Time Delay Relay" under Core Standby Cooling Systems Isolation in the title of Item 4.a.2 on Page 3/4 3-22.

Basis

The proposed change was made solely to provide consistency with other titles throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The instrumentation itself has not changed, nor has its function, design, or operation. Only its title has been reworded to be consistent with other titles in the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table. It does not involve a change in instrumentation, function, design, or operation. Therefore, there is no impact on the margin of safety.

Proposed Change Number 70

Replace "and" with ";" in the instrument number listings in Item 4.a.5, "Bus Power Monitor," in Table 3.3.2-2 under Core Standby Cooling Systems Isolation on Page 3/4 3-23.

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table, and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability of the primary containment system nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 71

Replace Instrument Number E51-dTS-N604C,D with Instrument Number E51-TDS-N604C,D in Item 4.a.8, "HPCI Steam Line Area Δ Temperature - High," in Table 3.3.2-2 under Core Standby Cooling Systems Isolation on Page 3/4 3-23.

Basis

The isolation instrumentation listed in Item 4.a.8 isolates the high pressure coolant injection (HPCI) system should there be a HPCI steam line break.

Currently, the Technical Specifications list Instrument Number E51-dTS-N604C,D under Item 4.a.8. The proposed change will revise that number to E51-TDS-N604C,D. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference E51-dTS-N604C,D with E51-TDS-N604C,D. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 4.a.8 of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment system nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 72

Replace "Emergency Area Cooler Temp - High" with "HPCI Equipment Area Temperature - High" in the title of Item 4.a.9 in Table 3.3.2-2 under Core Standby Cooling Systems Isolation on Page 3/4 3-23.

Basis

The emergency area cooler temperature - high isolation signal isolates the Group 4 isolation valves in the event of a high pressure coolant injection (HPCI) steam line break to mitigate the consequences of such a break. The Group 4 isolation valves include the HPCI inboard and outboard steam line isolation valves and the HPCI torus suction isolation valves.

The current Technical Specifications have an inaccurate description of the trip function. The proposed change revises the description of the trip function from "Emergency Area Cooler Temperature - High" to "HPCI Equipment Area Temperature - High."

This change is necessary so that the trip function description more clearly reflects the actual area the instrumentation monitors. The existing description is a generic BWR term which does not relate to the actual BSEP design. The area the instrumentation monitors at BSEP is referred to in other plant documents and programs as the "HPCI Equipment Area." This change will reduce the potential for confusion by more clearly describing the actual location of the instrumentation.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will change the description of a trip function to more clearly identify the area the instrumentation monitors. The current wording, "Emergency Area Cooler," does not reflect the commonly used description of the area. "HPCI Equipment Area" more clearly describes the area. This change will reduce confusion concerning which area is being referred to.

   No changes to the referenced instrumentation are being reflected by this change. The instrumentation will function exactly as before. Therefore, there is no change in the probability of an accident, nor is there any change in the consequences of an accident.

2. The proposed change revises the trip function so the appropriate trip function is clearly identified in the Technical Specifications. It does not change the required function of the instrumentation; it merely clarifies the description of an existing trip function. Thus, no new accident possibilities are created.

3. The proposed change provides a clarification in the description of an existing isolation function. It does not change the design, reliability, or function of any plant equipment; it merely references them more clearly. Thus, there is no impact on the margin of safety as a result of this change.

Proposed Change Number 73

Add setpoint requirements for a new Item 4.a.10, "Drywell Pressure - High," and associated Instrument Number E11-PTS-N011C-2,D-2 to Table 3.3.2-2 under HPCI System Isolation on Page 3/4 3-23.

Basis

Valve Group 4 isolates the high pressure coolant injection (HPCI) inboard and outboard steam line isolation valves and the HPCI torus suction isolation valves in the event of a HPCI steam line break to mitigate the consequences of the break.

The HPCI system also has turbine exhaust vacuum breaker isolation valves (E41-F075 and E41-F079) on a vacuum relief line for the HPCI turbine exhaust. This line helps prevent the creation of a water column in the exhaust line. Preventing development of this column reduces the piping loads which could exist if the turbine is restarted. The valves isolate on coincident HPCI steam line pressure - low and drywell pressure - high signals to establish primary containment. The low steam line pressure signal indicates HPCI is isolated or no longer needed because the reactor is depressurized. The drywell pressure - high signal indicates a loss of coolant accident (LOCA) exists and primary containment isolation is needed.

The current Technical Specifications specify setpoint requirements for the Group 4 isolation valve logic but not for the vacuum breaker isolation valve logic. The proposed change reflects the creation of a new Valve Group 7, which includes the HPCI turbine exhaust vacuum breakers E41-F075 and E41-F079. Setpoint requirements for the actuation instrumentation are provided under Item 4.a.3 for the HPCI steam supply pressure - low instrumentation portion of the logic. A new Item 4.a.10 is being added for the drywell pressure - high portion of the logic to specify the appropriate setpoints. These values are consistent with the accident analysis assumptions and are the same as for other LOCA detection instrumentation.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes setpoint requirements for existing instrumentation which was not originally included in the Technical Specifications to provide additional assurance that primary containment is isolated when necessary. These requirements are consistent with the accident analysis and are the same as for other LOCA detection instrumentation and established values for this instrumentation. No new equipment is being added and existing equipment will perform its safety function in the same manner as before. Therefore, there is no impact on the consequences of any accident, nor is there any change in the probability of any accident.

2. The proposed change adds references to existing requirements to the Technical Specifications. This will provide a more complete and accurate document, thereby enhancing primary containment isolation assurance. It does not provide an opportunity for any new accidents because no equipment, operability requirements, or setpoints are being altered. Only references to existing instrumentation are being added.

3. The proposed change adds requirements for existing instrumentation to the Technical Specifications. It does not reflect a physical change to the plant systems; it only provides a more complete and accurate description of existing equipment. Thus, there is no impact on the margin of safety of the plant.

Proposed Change Number 74

Replace "RCIC Steam Line High Flow Time Delay Relay" with "RCIC Steam Line Flow - High Time Delay Relay" in the title of Item 4.b.2 in Table 3.3.2-2 under Core Standby Cooling Systems Isolation on Page 3/4 3-24.

Basis

The proposed change was made solely to provide consistency with other titles throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The instrumentation itself has not changed, nor has its function, design, or operation. Only its title has been reworded to be consistent with other titles in the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table. It does not involve a change in instrumentation, function, design, or operation. It does not change the design or capability of the primary containment system nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 75

Replace "and" with ";" in the instrument number listings in Table 3.3.2-2 under Item 4.b.5, "Bus Power Monitor," under Core Standby Cooling System Isolation on Page 3/4 3-24.

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table, and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability of the primary containment system nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 76

Replace Instrument Number E51-dTS-N604A,B with Instrument Number E51-TDS-N604A,B in Item 4.b.8, "RCIC Steam Line Area Δ Temperature - High," in Table 3.3.2-2 under Core Standby Cooling Systems Isolation on Page 3/4 3-24.

Basis

The isolation instrumentation listed in Item 4.b.8 isolates the reactor core isolation cooling (RCIC) system should there be a RCIC steam line break.

Currently, the Technical Specifications list Instrument Number E51-dTS-N604A,B under Item 4.b.8. The proposed change will revise that number to E51-TDS-N604A,B. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference E51-dTS-N604A,B with E51-TDS-N604A,B. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 4.b.8 of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment system nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 77

Replace Instrument Number E51-dTS-N601A,B with Instrument Number E51-TDS-N601A,B in Item 4.b.10, "RCIC Equipment Room Δ Temperature - High," in Table 3.3.2-2 under Core Standby Cooling Systems Isolation on Page 3/4 3-25.

Basis

The isolation instrumentation listed in Item 4.b.10 isolates the reactor core isolation cooling (RCIC) system should there be a RCIC steam line break.

Currently, the Technical Specifications list Instrument Number E51-dTS-N601A,B under Item 4.b.10. The proposed change will revise that number to E51-TDS-N601A,B. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference E51-dTS-N601A,B with E51-TDS-N601A,B. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 4.b.10 of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment system nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 78

Add new Item 4.b.11, "RCIC Steam Line Tunnel Temperature - High Time Delay Relay," and associated Instrument Number E51-KC-M602A,B to Table 3.3.2-2 under Core Standby Cooling Systems Isolation on Page 3/4 3-25.

Basis

The reactor core isolation cooling (RCIC) steam line tunnel temperature - high time delay relay instrumentation isolates the Group 5 isolation valves in the event of a RCIC steam line break to mitigate the consequences of the break. The Group 5 valves include the RCIC inboard and outboard steam line isolation valves.

The current Technical Specifications do not specifically reference setpoint requirements for the existing time delay relays in the RCIC steam line tunnel temperature - high logic. The proposed change adds requirements for these relays which reflect the established value and are in agreement with the accident analyses. Initially, the time delay relays were considered to be a part of the isolation logic described in Item 4.b.7 and were not listed separately in the table. These relays do not initiate any isolation signal; however, they are an important part of the instrumentation, and it has been determined that they should be addressed separately in the table to avoid possible confusion.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes specific setpoint requirements for instrumentation which is designed to detect an accident and not prevent one. These requirements are consistent with the accident analysis assumptions. There is no change to the equipment or in how the equipment is operated; only references to existing instrumentation are being added to the table. Since the establishment of operability and surveillance requirements does not change the design or affect the operation of the instrumentation, there is no increase in the probability of an accident, nor is there any change in the consequences of any accident previously evaluated.

2. Specification of the correct setpoint requirements provides additional assurance that the instrumentation will perform its intended safety function. It does not change the design or operation of any equipment or cause it to operate in a manner not previously assumed. It does not affect any equipment which could cause an accident. It only affects instrumentation designed to detect the occurrence of an accident. Thus, this change does not create the possibility of a new accident.

3. Specification of specific setpoint requirements which reflect the established value and are consistent with the accident analysis does not change the operation of the affected equipment. There will be no change in the reliability, response characteristics, or setpoints of the instrumentation. There is no change in the design or capability of the primary containment system, nor is there any change the demands placed upon this process barrier. Therefore, there is no change in the margin of safety.

Proposed Change Number 79

Add setpoint requirements for new Item 4.b.12, "Drywell Pressure - High," and associated Instrument Number E11-PTS-N011A-2,B-2 to Table 3.3.2-2 under Core Standby Cooling Systems Isolation on Page 3/4 3-25.

Basis

Valve Group 5 isolates the reactor core isolation cooling (RCIC) steam line to mitigate the consequences of a break in that line. The Group 5 isolation valves include the RCIC inboard and outboard steam line isolation valves.

The RCIC system also has turbine exhaust vacuum breaker isolation valves (E51-F062 and E51-F066) on a vacuum relief line for the RCIC turbine exhaust. The line helps prevent the de elopment of a water column in the exhaust line. Preventing development of this column reduces the piping loads which could exist if the turbine is restarted. The valves isolate on coincident RCIC steam line pressure - low and drywell pressure - high signals to establish primary containment isolation. The steam line pressure - low signal indicates RCIC is isolated or no longer needed because the reactor is depressurized. The drywell pressure - high signal indicates a loss of coolant accident (LOCA) exists and primary containment isolation is needed.

The current Technical Specifications specify setpoint requirements for the Group 5 isolation valve logic but not for the vacuum breaker isolation valve logic. The proposed change reflects the creation of the new Valve Group 9, which includes the RCIC turbine exhaust vacuum breakers E51-F062 and E51-F066.

The isolation logic associated with these vacuum breakers requires a signal from both the drywell pressure - high instrumentation and the RCIC steam supply pressure - low instrumentation. Item 4.b.12 addresses the drywell pressure - high part of the logic; Item 4.b.3 addresses the RCIC steam supply pressure - low portion. Table 3.3.2-1 provides more information about the operability requirements of these valves in Items 4.b.3 and 4.b.12. The proposed change addressed here adds the trip setpoint information associated with the drywell pressure - high instrumentation.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes the trip setpoint values for the drywell pressure - high instrumentation which was not originally included in the Technical Specifications. This change is being made to address the addition of the RCIC turbine exhaust vacuum breakers, which isolate on coincident RCIC steam line pressure - low and drywell pressure - high, to the Technical Specifications via the addition of Valve Group 9. The steam line pressure - low signal indicates RCIC is isolated or no longer needed because the reactor is depressurized, and the drywell pressure - high signal indicates a LOCA exists and primary containment isolation is needed.

This change adds setpoint requirements for existing instrumentation to provide additional assurance that primary containment is isolated when necessary. No new equipment is being added, and existing equipment will perform its safety function in the same manner as before. These requirements are being added to the Technical Specifications for clarity and completeness. Thus, there is no change in the probability of an accident; nor is there any change in the consequences of any accident.

2. The proposed change adds setpoint requirements to the Technical Specifications for existing instrumentation. This will provide a more complete and accurate document, thereby enhancing primary containment isolation assurance. It does not provide an opportunity for any new accidents because no equipment, operability requirements, or setpoints are being altered.

3. The proposed change adds requirements for existing instrumentation to the Technical Specifications. It does not reflect a physical change to the plant systems; it only provides a more complete and accurate description of existing equipment. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, there is no impact on the margin of safety of the plant.

Proposed Change Number 80

Replace Footnote * with (a) in Item 5.a, "Reactor Vessel Water Level - Low, Level 1," in Table 3.3.2-2 under Shutdown Cooling System Isolation on Page 3/4 3-25 (BSEP-1 only).

Basis

The proposed change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnote has not changed unless noticed elsewhere in this enclosure. The footnote * has been replaced with (a) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnote. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnote. Therefore, there is no impact on the margin of safety.

Proposed Change Number 81

Replace Instrument Number B32-PS-N018A,B with Instrument Number B32-PS-N018A-1,B in Item 5.b, "Reactor Steam Dome Pressure - High," in Table 3.3.2-2 under Shutdown Cooling System Isolation on Page 3/4 3-25.

Basis

The isolation instrumentation listed in Item 5.b isolates the shutdown cooling system should there be a malfunction or rupture of the residual heat removal system while operating in shutdown cooling mode. This minimizes reactor inventory loss.

Currently, the Technical Specifications list Instrument Number B32-PS-N018A,B under Item 5.b. The proposed change will revise that number to B32-PS-N018A-1,B. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference B32-PS-N018A,B with B32-PS-N018A-1,B. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device and makes its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new accident.

3. The proposed change will more clearly identify an instrument that currently exists and is listed under Item 5.b of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 82

Replace Footnote * with Footnote (a) on the bottom of Page 3/4 3-25 (BSEP-1 Only).

Basis

This change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnote has not changed unless noticed elsewhere in this enclosure. The footnote * has been changed to (a) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnotes. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Proposed Change Number 83

Delete Footnote (b) from the footnote table on existing Page 3/4 3-21a (BSEP-2 only).

Basis

Footnote (b) was added to the Technical Specifications via Amendment 131 on December 10, 1986 to support the hydrogen injection test which took place in January, 1987. This was a one-time test; therefore, the footnote is no longer applicable or necessary for normal operation.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change deletes a footnote which no longer applies. The footnote was added to support a one-time, hydrogen injection test, which was completed on January 5, 1987. No additional testing is planned. Thus, the proposed change has no effect on the probability of an accident, nor does it affect the consequences of any accident.

2. The referenced footnote no longer applies to BSEP-2. The hydrogen injection test was successfully completed on January 5, 1987. Thus, this footnote is no longer necessary, and deletion of it will not create the possibility of a new or different type of accident.

3. Footnote (b) was added to support a one-time hydrogen injection test, which was completed on January 5, 1987. No additional testing is planned; therefore, the footnote no longer applies and should be deleted. This deletion has no impact on the margin of safety.

Proposed Change Number 84

Relocate Footnote (a), "Vessel water levels refer to REFERENCE LEVEL ZERO," from the bottom of existing Page 3/4 3-21a to the bottom of new Page 3/4 3-25 (BSEP-2 only).

Basis

This footnote is being relocated from existing Page 3/4 3-21a to the bottom of new Page 3/4 3-25 for convenience. Since Footnote (b) is being deleted as described elsewhere in this submittal, there is no longer any need for a separate footnote table. The content of the footnote has not changed.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change relocates Footnote (a) from existing Page 3/4 3-21a to the bottom of Page 3/4 3-25. The content of the footnote remains the same. This change is being made solely for convenience. Therefore, there is no change in the probability of any accident, nor is there any change in the consequences of any accident.

2. The proposed change is administrative in nature. It does not change the content of the affected footnote; it only relocates it to the bottom of Page 3/4 3-25 because a separate footnote table is no longer necessary due to deletion of another footnote. Therefore, there are no new accident possibilities created.

3. The proposed change is administrative in nature; it is being made for consistency and convenience. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, it has no impact on the margin of safety.

Proposed Change Number 85

Replace reference to Footnote # with (d) and add Footnote (a) to the response time column heading on Table 3.3.2-3 on Pages 3/4 3-26 through 3/4 3-29.

Basis

Table 3.3.2-3 specifies the response time requirements for the isolation instrumentation. It ensures that the time it takes for the instrumentation to detect and initiate valve closure is within the allowed time assumed by the accident analysis.

The isolation response time is the sum of the instrument response time and the valve isolation time. For AC powered valves, the instrument response time also includes the emergency diesel generator start time since power to these valves must be re-established after a loss of off-site power before the valves can stroke closed. The instrument response time, therefore, is the greater of the following: 1) the measured response time of the instrument itself, including any time delay relays, or 2) the start time of the emergency diesel generators.

The current Technical Specifications provide an explanation of the requirements for determining instrumentation response time in Footnotes (a) and (#), which state:

(a) The isolation system instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Isolation system instrumentation response time specified includes the delay for diesel generator starting assumed in the accident analysis.

(#) Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to isolation time shown in Table 3.6.3-1 and Table 3.6.5.2-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

Footnote (#) is currently specified in the header for the RESPONSE TIME column of Table 3.3.2-3, which implies that it applies to all isolation instrumentation. Footnote (a) is currently listed only with Items 4.a.1 and 4.b.1, which are concerned with the HPCI and RCIC steam line flow - high instrumentation.

The proposed change moves the reference to Footnote (a) from just Items 4.a.1 and 4.b.1 to the header for the RESPONSE TIME column. Footnote (a) is being revised to state:

(a) The isolation system instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Isolation system instrumentation response time specified includes any delay for diesel generator starting assumed in the accident analysis.

By changing the wording to "any delay for diesel generator starting...," this note may be applied to all of the trip function instrumentation. It also recognizes that the response time must also include an allowance for the diesel generator start time if closure of the associated isolation valves is dependent upon the diesel generators. The diesel generator start time is a factor in the response times for most of the isolation valves, not just the HPCI and RCIC high steam flow trip functions. It does not apply to the main steam isolation valves (MSIVs) since they are air operated and have a shorter isolation time assumed in the accident analysis.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change adds a footnote which indicates that the diesel generator start time is a factor in determining the response times for most isolation instrumentation. This change does not reflect a change to the plant design, operation, surveillance or testing; it merely clarifies and expands upon existing requirements and information. Therefore, there is no increase in the probability of an accident, nor is there any change in the consequences of an accident.

2. The proposed change provides a clarification to the requirements for determining response times for AC powered valves. This reflects existing information, and does not reflect a change in plant design. Therefore, the proposed change does not create the possibility of any new or different kinds of accidents.

3. The proposed change does not reflect a change to the design of the plant. It more clearly states existing requirements, and will potentially reduce confusion and misinterpretation of the requirements. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, the margin of safety is not decreased.

Proposed Change Number 86

Delete Item 1.a.2, "Reactor Vessel Water Level - Low, Level 2," trip function from Table 3.3.2-3 and re-label Item 1.a.3 to accommodate this deletion on Page 3/4 3-26 (BSEP-2 only).

Basis

The Technical Specification change request submitted on September 29, 1987, as supplemented on October 14, 1987 and November 24, 1987, revised the reactor vessel water level trip function for the Valve Group 1 isolation valves from reactor vessel water level - low, level 2 to reactor vessel water level - low, level 3. This resulted in only Valve Group 3 being actuated by the reactor vessel water level - low, level 2 trip function. Valve Group 3 isolates the reactor water cleanup system and is addressed specifically in Item 3.e for the reactor vessel water level - low level 2 instrumentation. The proposed change deletes Item 1.a.2 because the instrumentation no longer actuates any valve groups that need to be addressed under Item 1. This change does not represent any physical change to the design or operation of any systems. It only more accurately describes the trip function associated with the Group 3 valves.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change deletes information which no longer belongs under Item 1. Via a previous submittal, the trip function for the Group 1 isolation valves was revised from reactor vessel water level - low, level 2 to reactor vessel water level - low, level 3. The information associated with the remaining valve group actuated by that trip function, Valve Group 3, is more appropriately referenced in Item 3.e, which describes the instrumentation that actuates reactor water cleanup system isolation. The proposed change does not reflect a change to the design or operation of the instrumentation and valve groups; it only clarifies existing information in the table. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change clarifies the table by placing the information associated with the reactor vessel water level - low, level 2 instrumentation in its appropriate place. It does not reflect a change in the design or operation of the system. Therefore, it does not create the possibility of a new or different accident.

3. The proposed change does not reflect a change to the design or operation of any equipment. It merely provides the applicable references to the reactor vessel water level - low, level 2 instrumentation in the appropriate place in the table. Therefore, there is no impact on the margin of safety.

Proposed Change Number 87

Replace Footnote * with Footnote (c) in Item 1.a.2, "Reactor Vessel Water Level - Low, Level 2," (BSEP-1 only) and "Reactor Vessel Water Level - Low, Level 3 (BSEP-2 only) in Table 3.3.2-2 under Primary Containment Isolation on Page 3/4 3-26.

Basis

This change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnote has not changed unless noticed elsewhere in this enclosure. The footnote has been changed from * to (c) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnotes. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Proposed Change Number 88

Add "$\leq 13^{(h)}$" under the response time column for Item 1.a.2, "Reactor Vessel Water Level - Low, Level 2," (BSEP-1 only) and "<13" under the response time column for Item 1.a.2, "Reactor Vessel Water Level - Low, Level 3" (BSEP-2 only) in Table 3.3.2-3 under Primary Containment Isolation or Page 3/4 3-26 .

Basis

Item 1.a.2 specifies the response time requirements for the reactor vessel water level - low, level 2 instrumentation that actuates the valve group isolations. It ensures the time it takes for the instrumentation to detect and initiate valve closure is within the allowed time assumed by the accident analysis. These limits, therefore, ensure the instrumentation reacts quickly enough to limit the amount of radioactivity, reactor inventory, and energy lost from containment to less than that determined by the accident analysis.

The isolation system response time is the sum of the instrumentation response time and the valve isolation time. For AC powered valves, the instrumentation response time also includes the emergency diesel generator start time since the power to these valves must be re-established after the loss of off-site power before the valves can stroke closed. The instrumentation response time is, therefore, the greater of the following: 1) the measured response time of the instrumentation by itself, including any time delay relays, or 2) the start time of the emergency diesel generators.

The current Technical Specifications specify an instrumentation response time of 1.0 second for the trip function instrumentation of Item 1.a.2. A reference to Footnote * which states, "Isolation actuation instrumentation response time only," is also provided. This footnote indicates that any time delay needed for diesel generator starting is not subject to the response time requirement since the requirement applies only to the instrumentation.

The trip function covered by Item 1.a.2 actuates both AC powered valves and the main steam isolation valves (MSIVs). The MSIVs are not dependent on AC power for closure. The response times currently specified are appropriate for the MSIVs, but not for the other valves actuated by this instrumentation since they are dependent upon the diesel generators. The proposed change establishes a 13 second response time requirement for the valves other than the MSIVs to accommodate the diesel generator start time. A new Footnote (h) is added with the new 13 second response time which states, "Isolation system instrumentation response time for associated valves except MSIVs." The proposed change also revises Footnote *, which is associated with the 1.0 second response time, to Footnote (c) which states, "Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed."

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change clarifies the existing response time requirements for the reactor vessel water level - low, level 2 instrumentation. The response time currently listed, 1.0 seconds, actually applies only to the main steam isolation valves. The new additional response time listed 13 seconds, applies to the AC powered valves which rely on the diesel generators. The 13 seconds includes the diesel generator start times

   The MSIV closure time is the most critical because it represents the worst case pathway for inventory, energy, and radiation loss. The response time for this instrumentation has not changed. The design, operation, reliability, or testing requirements of plant systems have not changed. The instrumentation will function in the same manner as it currently does; the response times for the referenced information will be more completely and accurately represented in the Technical Specifications and remain in agreement with the accident analysis assumptions. Thus, there is no increase in the probability of an accident; nor is there a change in the consequences of an accident.

2. The proposed change is a clarification of existing information. The MSIVs have a response time of 1.0 second. Other valves which are associated with the reactor vessel water level - low, level 2 instrumentation are AC powered. Therefore, diesel generator delays should be used in calculation of the response times for those valves. This information is now referenced in the Technical Specifications, making it a more complete and accurate document. No changes are being made to the design, operation, or testing requirements of the existing instrumentation; the proposed change only provides a reference to existing information. Therefore, it does not create the possibility of a new or different type of accident.

3. The proposed change does not change the design, operation, or testing requirements of any existing instrumentation or equipment. It only provides a reference to existing information which was not previously referenced in the Technical Specifications. Thus, there is no impact on the margin of safety.

Proposed Change Number 89

Replace "and" with ";" in the instrument number listings in Item 1.a.2. "Reactor Vessel Water Level - Low, Level 2," (BSEP-1 only) and "Reactor Vessel Water Level - Low, Level 3" (BSEP-2 only) in Table 3.3.2-3 under Primary Containment Isolation on Page 3/4 3-26.

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table, and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 90

Replace Footnote * with Footnote (c) in Item 1.c.1, "Main Steam Line Radiation - High," in Table 3.3.2-3 under Primary Containment Isolation on Page 3/4 3-26.

Basis

This change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnote has not changed unless noticed elsewhere in this enclosure. The footnote has been changed from * to (c) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnotes. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Proposed Change Number 91

Add "$\leq 13^{(h)}$" under the response time column for Item 1.c.1, "Main Steam Line Radiation - High," in Table 3.3.2-3 under Primary Containment Isolation on Page 3/4 3-26.

Basis

Item 1.c.1 specifies the response time requirements for the main steam line radiation - high instrumentation that actuates the valve group isolations. It ensures the time it takes for the instrumentation to detect and initiate valve closure is within the allowed time assumed by the accident analysis. These limits, therefore, ensure the instrumentation reacts quickly enough to limit the amount of radioactivity lost from containment to less than that determined by the accident analysis.

The isolation system response time is the sum of the instrumentation response time and the valve isolation time. For AC powered valves, the instrumentation response time also includes the emergency diesel generator start times since the power to these valves must be re-established after the loss of off-site power before the valves can stroke closed. The instrumentation response time is, therefore, the greater of the following: 1) the measured response time of the instrumentation by itself, including any time delay relays, or 2) the start time of the emergency diesel generators.

The current Technical Specifications specify an instrumentation response time of 1.0 seconds for the trip function instrumentation of Item 1.c.1. A reference to Footnote * which states, "Isolation actuation instrumentation response time only," is also provided. This footnote indicates that any time delay needed for diesel generator starting is not subject to the response time requirement since the requirement applies only to the instrumentation.

The trip function covered by Item 1.c.1 actuates both AC powered valves and the main steam isolation valves (MSIVs). The MSIVs are not dependent on AC power for closure. The response times currently specified are appropriate for the MSIVs, but not for the other valves actuated by this instrumentation since they are dependent upon the diesel generators. The proposed change establishes a 13 second response time requirement for the valves other than the MSIVs to accommodate the diesel generator start time. A new Footnote (h) is added with the new 13 second response time which states, "Isolation system instrumentation response time for associated valves except MSIVs." The proposed change also revises Footnote *, which is associated with the 1.0 second response time, to Footnote (c) which states, "Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed."

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change clarifies the existing response time requirements for the main steam line radiation - high instrumentation. The response time currently listed, 1.0 second, actually applies only to the main steam isolation valves. The new additional response time listed, 13 seconds, applies to the AC powered valves which rely on the diesel generators. The 13 seconds includes the diesel generator start times.

   The MSIV closure time is the most critical because it represents the worst case pathway for inventory, energy, and radiation loss. The response time for this instrumentation has not changed. The design, operation, reliability, or testing requirements of plant systems has not changed. The instrumentation will function in the same manner as it currently does; the response times for the referenced information will be more completely and accurately represented in the Technical Specifications and remain in agreement with the accident analysis assumptions. Thus, there is no increase in the probability of an accident, nor is there a change in the consequences of an accident.

2. The proposed change is a clarification of existing information. The MSIVs have a response time of 1.0 second. Other valves associated with the main steam line radiation - high instrumentation are AC powered. Therefore, diesel generator delays should be utilized in calculation of the response times for those valves. This information is now referenced in the Technical Specifications, making it a more complete and accurate document. No changes are being made to the design, operation, or testing requirements of the existing instrumentation; the proposed change only provides reference to existing information. Therefore, it does not create the possibility of a new or different type of accident.

3. The proposed change does not change the design, operation, or testing requirements of any existing instrumentation or equipment. It only provides reference to existing information which was not previously referenced in the Technical Specifications. Thus, there is no impact on the margin of safety.

Proposed Change Number 92

Replace Footnote * with Footnote (c) in Item 1.c.3, "Main Steam Line Flow - High," in Table 3.3.2-3 under Primary Containment Isolation on Page 3/4 3-26.

Basis

This change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnote has not changed unless noticed elsewhere in this enclosure. The footnote has been changed from * to (c) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnotes. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Proposed Change Number 93

Add "$\leq 13^{(h)}$" under the response time column for Item 1.c.3, "Main Steam Line Flow - High," in Table 3.3.2-3 under Primary Containment Isolation on Page 3/4 3-26.

Basis

Item 1.c.3 specifies the response time requirements for the main steam line flow - high instrumentation that actuates the valve group isolations. It ensures the time it takes for the instrumentation to detect and initiate valve closure is within the allowed time assumed by the accident analysis. These limits, therefore, ensure the instrumentation reacts quickly enough to limit the amount of radioactivity, reactor inventory, and energy lost from containment to less than that determined by the accident analysis.

The isolation system response time is the sum of the instrumentation response time and the valve isolation time. For AC powered valves, the instrumentation response time also includes the emergency diesel generator start time since the power to these valves must be re-established after the loss of off-site power before the valves can stroke closed. The instrumentation response time is, therefore, the greater of the following; 1) the measured response time of the instrumentation by itself, including any time delay relays, or 2) the start time of the emergency diesel generators.

The current Technical Specifications specify an instrumentation response time of 0.5 seconds for the trip function instrumentation of Item 1.c.3. A reference to Footnote * which states, "Isolation actuation instrumentation response time only," is also provided. This footnote indicates that any time delay needed for diesel generator starting is not subject to the response time requirement since the requirement applies only to the instrumentation.

The trip function covered by Item 1.c.3 actuates both AC powered valves and the main steam isolation valves (MSIVs). The MSIVs are not dependent on AC power for closure. The response times currently specified are appropriate for the MSIVs, but not for the other valves actuated by this instrumentation since they are dependent upon the diesel generators. The proposed change establishes a 13 second response time requirement for the valves other than the MSIVs to accommodate the diesel generator start time. A new footnote (h) is added with the new 13 second response time. The footnote states, "Isolation system instrumentation response time for associated valves except MSIVs." The proposed change also revises Footnote *, which is associated with the 0.5 second response time, to Footnote (c) which states, "Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed."

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change clarifies the existing response time requirements for the main steam line flow - high instrumentation. The response time currently listed, 0.5 seconds, actually applies only to the main steam isolation valves. The new additional response time listed, 13 seconds, applies to the AC powered valves which rely on the diesel generators. The 13 seconds includes the diesel generator start times.

   The MSIV closure time is the most critical because it represents the worst case pathway for inventory, energy, and radiation loss. The response time for this instrumentation has not changed. The design, operation, reliability, or testing requirements of plant systems has not changed. The instrumentation will function in the same manner as it currently does; the response times for the referenced information will be more completely and accurately represented in the Technical Specifications and remain in agreement with the accident analysis assumptions. Thus, there is no increase in the probability of an accident; nor is there a change in the consequences of an accident.

2. The proposed change is a clarification of existing information. The MSIVs have a response time in this situation of 0.5 seconds. Other valves which are associated with the main steam line radiation - high instrumentation are AC powered. Therefore, diesel generator delays should be utilized in calculation of the response times for those valves. This information is now referenced in the Technical Specifications, making it a more complete and accurate document. No changes are being made to the design, operation, or testing requirements of the existing instrumentation; the proposed change only provides reference to existing information. Therefore, it does not create the possibility of a new or different type of accident.

3. The proposed change does not change the design, operation, or testing requirements of any existing instrumentation or equipment. It only provides reference to existing information which was not previously referenced in the Technical Specifications. Thus, there is no impact on the margin of safety.

Proposed Change Number 94

Replace Footnote * with Footnote (c) in Item 1.c.4, "Main Steam Line Flow - High," in Table 3.3.2-3 under Primary Containment Isolation on Page 3/4 3-26 (**BSEP-2 Only**).

Basis

This change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnote has not changed unless noticed elsewhere in this enclosure. The footnote has been changed from * to (c) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnotes. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Proposed Change Number 95

Add "<13(h)" under the response time column for Item 1.c.4, "Main Steam Line Flow - High," in Table 3.3.2-3 under Primary Containment Isolation on Page 3/4 3-26 (BSEP-2 only).

Basis

Item 1.c.4 specifies the response time requirements for the main steam line flow - high instrumentation that actuates the valve group isolations while in the startup mode. It ensures the time it takes for the instrumentation to detect and initiate valve closure is within the allowed time assumed by the accident analysis. These limits, therefore, ensure the instrumentation reacts quickly enough to limit the amount of radioactivity, reactor inventory, and energy lost from containment to less than that determined by the accident analysis.

The isolation system response time is the sum of the instrumentation response time and the valve isolation time. For AC powered valves, the instrumentation response time also includes the emergency diesel generator start time since the power to these valves must be re-established after the loss of off-site power before the valves can stroke closed. The instrumentation response time is, therefore, the greater of the following; 1) the measured response time of the instrumentation by itself, including any time delay relays, or 2) the start time of the emergency diesel generators.

The current Technical Specifications specify an instrumentation response time of 0.5 seconds for the trip function instrumentation of Item 1.c.4. A reference to Footnote * which states, "Isolation actuation instrumentation response time only," is also provided. This footnote indicates that any time delay needed for diesel generator starting is not subject to the response time requirement since the requirement applies only to the instrumentation.

The trip function covered by Item 1.c.4 actuates both AC powered valves and the main steam isolation valves (MSIVs). The MSIVs are not dependent on AC power for closure. The response times currently specified are appropriate for the MSIVs, but not for the other valves actuated by this instrumentation since they are dependent upon the diesel generators. The proposed change establishes a 13 second response time requirement for the valves other than the MSIVs to accommodate the diesel generator start time. A new Footnote (h) is added with the new 13 second response time which states, "Isolation system instrumentation response time for associated valves except MSIVs." The proposed change also revises Footnote *, which is associated with the 0.5 second response time, to (c) which states, "Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed."

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change clarifies the existing response time requirements for the main steam line flow - high instrumentation. The response time currently listed, 0.5 seconds, actually applies only to the main steam isolation valves. The new additional response time listed, 13 seconds, applies to the AC powered valves which rely on the diesel generators. The 13 seconds includes the diesel generator start times.

   The MSIV closure time is the most critical because it represents the worst case pathway for inventory, energy, and radiation loss. The response time for this instrumentation has not changed. The design, operation, reliability, or testing requirements of plant systems has not changed. The instrumentation will function in the same manner as it currently does; the response times for the referenced information will be more completely and accurately represented in the Technical Specifications and remain in agreement with the accident analysis assumptions. Thus, there is no increase in the probability of an accident, nor is there a change in the consequences of an accident.

2. The proposed change is a clarification of existing information. The MSIVs have a response time in this situation of 0.5 seconds. Other valves which are associated with the main steam line radiation - high instrumentation are AC powered. Therefore, diesel generator delays should be utilized in calculation of the response times for those valves. This information is now referenced in the Technical Specifications, making it a more complete and accurate document. No changes are being made to the design, operation, or testing requirements of the existing instrumentation; the proposed change only provides reference to existing information. Therefore, it does not create the possibility of a new or different type of accident.

3. The proposed change does not change the design, operation, or testing requirements of any existing instrumentation or equipment. It only provides reference to existing information which was not previously referenced in the Technical Specifications. Thus, there is no impact on the margin of safety.

Proposed Change Number 96

Add new Item 1.g, "Reactor Building Exhaust Radiation - High," and associated Instrument Numbers D12-RE-N010A,B and D12-RM-K609A,B to Table 3.3.2-3 under Primary Containment Isolation on Page 3/4 3-27.

Basis

The reactor building exhaust radiation - high isolation signal causes the Group 6 isolation valves, which include the containment atmospheric control valves, and certain containment atmospheric monitoring and post-accident sampling system valves, to close during a loss of coolant accident (LOCA). This minimizes the amount of radiation released from primary containment under LOCA conditions. This signal is not of primary importance for Group 6 isolation; the reactor vessel water level - low, level 2 and drywell pressure - high signals provide Group 6 isolation signals much earlier in a LOCA scenario. These two signals directly detect a LOCA, while the reactor building exhaust radiation - high instrumentation detects radiation released from primary containment to the reactor building. It does not directly detect a LOCA and will isolate Group 6 much later than the other two signals will.

Item 1 provides information concerning isolation of the main steam lines, drywell drains, and the containment atmospheric control and monitoring systems. The main steam lines are isolated by Group 1 valves, the drywell drains and associated systems by the Group 2 valves, and the containment atmospheric control and monitoring systems by the Group 6 valves.

The new Item 1.g provides information relating to the reactor building exhaust radiation - high instrumentation response times. This instrumentation consists of two channels; one per trip system, either of which can initiate a Group 6 isolation. The radiation monitors listed are located in the reactor building exhaust plenum and monitor normal HVAC effluent from secondary containment in the reactor building.

Item 1.g provides the appropriate response time requirements for isolation of the Group 6 valves on a signal from the reactor building exhaust radiation - high instrumentation. A response time requirement is not necessary since the Group 6 valves closed by this trip function are isolated by two other diverse signals which directly detect an accident. This signal is of secondary importance and not depended upon. Its response time, therefore, is not critical or the basis for any accident analysis.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes response time requirements for instrumentation which is not depended upon to detect accident conditions, or prevent an accident. These requirements are consistent with the accident analysis assumptions. There is no change in how the equipment is operated. This change establishes operability and surveillance requirements for existing equipment and does not change the design or affect the operation of the instrumentation. Thus, there is no increase in the probability of an accident.

   The proposed change adds response time requirements which are appropriate for the Group 6 isolation valves and their related safety function. The existing requirements in the Technical Specifications are listed under the Secondary Containment section, and are not directly applicable to the Group 6 valves. The reactor building exhaust radiation - high signal is not depended upon for Group 6 isolation for any design basis accident; thus, there is no impact on the consequences of any accident.

2. Specifying the correct response time requirements provides additional assurance that the instrumentation will perform its intended safety function. It does not change the design or operation of any equipment or cause it to operate in a manner not previously evaluated. It does not affect any equipment which could cause an accident. It only affects existing instrumentation designed to detect accident conditions. Therefore, it does not create the possibility of a new accident.

3. Specifying the correct response time requirements does not affect any equipment which could cause an accident. It only affects instrumentation designed to detect radioactivity in secondary containment and is not depended upon by this valve group for mitigating any design basis accident. Therefore, it is not a factor in the margin of safety. The instrumentation designed to detect and mitigate the design basis accidents is not affected by this change and will have the same reliability and response characteristics.

Proposed Change Number 97

Replace Instrument Number D12-RM-N010A,B with Instrument Number D12-RE-N010A,B in Item 2.a, "Reactor Building Exhaust Radiation - High," in Table 3.3.2-1 under Secondary Containment Isolation on Page 3/4 3-27.

Basis

A signal from the reactor building exhaust radiation - high instrumentation closes the secondary containment isolation dampers, thereby establishing secondary containment and minimizing the release of radiation from the reactor building during an accident. The dampers work in conjunction with the standby gas treatment system to establish a negative pressure in the reactor building and a controlled vent path through the standby gas treatment system filters and the stack. Secondary containment isolation would occur during a loss of coolatn accident (LOCA) due to reactor vessel water level - low, level 2 or drywell pressure - high. This would minimize the release of radioactivity from the reactor building. Secondary containment isolation is also assumed to occur during a refueling accident due to a signal from the reactor building exhaust radiation - high instrumentation to contain radiation released in a fuel handling accident.

Item 2.a currently incorrectly references Instrument Number D12-RM-N010A,B as the reactor building exhaust radiation - high instrumentation. This instrument should be identified as D12-RE-N010A,B. The proposed change does not result from a plant modification; the existing instrument number is not referenced correctly. It does not represent a change to the instrumentation. The proposed change will result in the instrument loop being more clearly and completely identified in the Technical Specifications, and reduce the possibility of confusion concerning which instrument channel is subject to those Technical Specification requirements.

10CFR50.92 Evaluation

This change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will result in the reactor building exhaust radiation - high instrumentation loop being more completely and correctly identified. The nomenclature will be consistent with that used in other plant documents and programs. It does not represent any change to the devices which perform the specified function; it more clearly references existing instrumentation. This change will reduce misunderstanding about which devices are associated with the isolation function. Thus, there is no change in the probability of an accident, nor is there any change in the consequences of any accident.

2. The existing instrument number listed under Item 2.a is not consistent with other plant documents and programs. There is no change to the instrument it represents; this change will only revise the instrument number to match that listed in other documentation. This change may reduce misunderstanding about which devices are associated with the isolation function. Thus, this change does not create the possibility of a new accident.

3. The proposed change will result in the reactor building exhaust radiation - high instrumentation being more correctly identified in the Technical Specifications. The nomenclature will be consistent with that used in other plant documents and programs and, therefore, reduce confusion about which instrumentation is being referenced. There is no physical change to the instrumentation; only its number is being revised in the Technical Specifications. Therefore, this change has no impact on the margin of safety.

Proposed Change Number 98

Add Instrument Number D12-RM-K609A,B to Item 2.a, "Reactor Building Exhaust Radiation - High," in Table 3.3.2-1 under Secondary Containment Isolation on Page 3/4 3-27.

Basis

A signal from the reactor building exhaust radiation - high instrumentation closes the secondary containment isolation dampers, thereby establishing secondary containment isolation and minimizing the release of radiation from the reactor building during an accident. The dampers work in conjunction with the standby gas treatment system to establish a negative pressure in the reactor building and a controlled vent path through the standby gas treatment system filters and the stack. Secondary containment isolation would occur during a loss of coolant accident (LOCA) due to a reactor vessel water level - low, level 2 or drywell pressure - high signal. This would minimize the release of radioactivity from the reactor building. Secondary containment isolation is also assumed to occur during a refueling accident due to a signal from the reactor building exhaust radiation - high instrumentation to contain radiation released in a fuel handling accident. Item 2.a currently lists only the radiation monitor D12-RM-N010A,B. It should also list the radiation monitor drawer D12-RM-K609A,B which is located in the control room.

The proposed change does not result from a plant modification; it is an addition of a reference to existing instrumentation which is currently associated with this system, but not listed. The instrumentation will continue to perform its intended function just as before; however, it will now be listed in its appropriate place in the Technical Speciffcations.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will result in the reactor building exhaust radiation - high instrumentation loop being more completely and correctly identified. The list of instruments associated with the reactor building exhaust radiation - high instrumentation will be more complete and consistent with that used in other plant documents and programs. It does not represent any change to the devices which perform the specified function; it more clearly references existing instrumentation. This change will reduce misunderstanding about which devices are associated with the isolation function. Thus, there is no change in the probability of an accident, nor is there any change in the consequences of any accident.

2. The existing instrument number listed under Item 2.a is not a complete representation of the instrumentation associated with the isolation instrumentation. There is no change to the instrumentation represented; this change will only add an additional instrument number to provide a more complete list of associated instruments. This change may reduce misunderstanding about which devices are associated with the isolation function. Thus, this change does not create the possibility of a new accident.

3. The proposed change will result in the reactor building exhaust radiation - high instrumentation being more correctly identified in the Technical Specifications. The instrumentation will be referenced more completely and thereby reduce confusion about which instrumentation is associated with the isolation signal. There is no physical change to the instrumentation; an additional reference is being added to provide a more complete description of the isolation instrumentation. Therefore, this change has no impact on the margin of safety.

Proposed Change Number 99

Replace "and" with ";" in the instrument number listings in Item 2.c, "Reactor Vessel Water Level - Low, Level 2," in Table 3.3.2-3 under Secondary Containment Isolation on Page 3/4 3-27.

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table, and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 100

Replace Footnote * with Footnote (c) in Item 2.c, "Reactor Vessel Water Level - Low, Level 2," in Table 3.3.2-3 under Secondary Containment Isolation on Page 3/4 3-27 (BSEP-1 only).

Basis

This change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnote has not changed unless noticed elsewhere in this enclosure. The footnote has been changed from * to (a) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnotes. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Proposed Change Number 101

Add "$\leq 13^{(h)}$" under the response time column for Item 2.c, "Reactor Vessel Water Level - Low, Level 2," in Table 3.3.2-3 under Secondary Containment Isolation on Page 3/4 3-27 (BSEP-1 only).

Basis

Item 2.c specifies the response time requirements for the reactor vessel water level - low, level 2 instrumentation that actuates the secondary containment isolation function. There are also other valve groups actuated by that instrumentation. It ensures that the time it takes for the instrumentation to detect and initiate valve closure is within the allowed time assumed by the accident analysis. These limits, therefore, ensure the instrumentation reacts quickly enough to limit the amount of radioactivity lost from secondary containment as assumed by the accident analysis.

The current Technical Specifications specify an instrumentation response time of 1.0 second for the trip function instrumentation of Item 2.c. A reference to Footnote * which states, "Isolation actuation instrumentation response time only," is also provided. This footnote indicates that any time delay needed for diesel generator starting is not subject to the response time requirement since the requirement applies only to the instrumentation.

The trip function covered by Item 2.c actuates both AC powered valves and the main steam isolation valves (MSIVs). The response times currently specified are appropriate for the MSIVs, but not for the other valves, or the secondary containment isolation dampers actuated by this instrumentation. The proposed change establishes a 13 second response time requirement for the dampers. This item is consistent with the accident analysis assumptions and is sufficient to initiate secondary containment before any release of radioactivity following a loss of coolant accident (LOCA). The correct response time requirements for the MSIVs are specified under Item 1.a.2.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change clarifies the existing response time requirements for the reactor vessel water level - low, level 2 instrumentation. The response time currently listed, 1.0 seconds, actually applies only to the main steam isolation valves. The new additional response time of 13 seconds is appropriate for the secondary containment isolation dampers.

The MSIV closure time is the most critical because it represents the worst case pathway for inventory, energy, and radiation loss. The response time for this instrumentation has not changed. The design, operation, reliability, or testing requirements of plant systems have not changed. The instrumentation will function in the same manner as it currently does; the response times for the referenced information will be more completely and accurately represented in the Technical Specifications and remain in agreement with the accident analysis assumptions. Thus, there is no increase in the probability of an accident, nor is there a change in the consequences of an accident.

2. The proposed change is a clarification of existing information. There is no change in the design or operation of the systems. Containment isolation will still occur as assumed in the accident analysis. Since the equipment will still operate and perform its safety function as before, the change does not create the possibility of a new or different type of accident.

3. The proposed change does not change the design or operation requirements of any existing instrumentation or equipment. It only provides a reference to existing information which was not previously specified in the Technical Specifications. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, there is no impact on the margin of safety.

Proposed Change Number 102

Replace "≤1.0*" with "≤13" under the response time column for Item 2.c, "Reactor Vessel Water Level - Low, Level 2," in Table 3.3.2-3 under Secondary Containment Isolation on Page 3/4 3-27 (BSEP-2 only).

Basis

Item 2.c specifies the response time requirements for the reactor vessel water level - low, level 2 instrumentation that actuates the secondary containment isolation function. There are also other valve groups actuated by that instrumentation. It ensures the time it takes for the instrumentation to detect and initiate valve closure is within the allowed time assumed by the accident analysis. These limits, therefore, ensure the instrumentation reacts quickly enough to limit the amount of radioactivity lost from secondary containment as assumed by the accident analysis.

The current Technical Specifications specify an instrumentation response time of 1.0 second for the trip function instrumentation of Item 2.c. A reference to Footnote * which states, "Isolation actuation instrumentation response time only," is also provided. This footnote indicates that any time delay needed for diesel generator starting is not subject to the response time requirement since the requirement applies only to the instrumentation.

The trip function covered by Item 2.c actuates only the AC powered valves and not the main steam isolation valves (MSIVs) since the trip setpoint for the MSIVs has been changed from reactor vessel water level - low, level 2 to reactor vessel water level - low, level 3 (see the September 29, 1987 submittal referenced in the cover letter). The response time currently specified are appropriate for the MSIVs, but not for the other valves, or the secondary containment isolation dampers actuated by this instrumentation. The proposed change establishes a 13 second response time requirement for the secondary containment isolation dampers. This item is consistent with the accident analysis assumptions and is sufficient to initiate secondary containment before any release of radioactivity following a LOCA. The correct response time requirements for the MSIVs are specified under Item 1.a.2.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change clarifies the existing response time requirements for the reactor vessel water level - low, level 2 instrumentation. The response time currently listed, 1.0 seconds, actually applies only to the main steam isolation valves. The new response time of 13 seconds is appropriate for the secondary containment isolation dampers.

The MSIV closure time is the most critical because it represents the worst case pathway for inventory, energy, and radiation loss. The response time for this instrumentation has not changed, and it is addressed in Item 1.a.2, "Reactor Vessel Water Level - Low, Level 3." The design, operation, reliability, or testing requirements of plant systems have not changed. The instrumentation will function in the same manner as it currently does; the response times for the referenced information will be more completely and accurately represented in the Technical Specifications and remain in agreement with the accident analysis assumptions. Thus, there is no increase in the probability of an accident, nor is there a change in the consequences of an accident.

2. The proposed change is a clarification of existing information. There is no change in the design or operation of the systems. Containment isolation will still occur as assumed in the accident analysis. Since the equipment will still operate and perform its safety function as before, the change does not create the possibility of a new or different type of accident.

3. The proposed change does not change the design or operation requirements of any existing instrumentation or equipment. It only provides a reference to existing information which was not previously specified in the Technical Specifications. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, there is no impact on the margin of safety.

Proposed Change Number 103

Replace Instrument Number G31-dFS-N603-1A,1B with Instrument Number G31-FDS-N603-1A,1B in Item 3.a, "Δ Flow - High," in Table 3.3.2-3 under Reactor Water Cleanup System Isolation on Page 3/4 3-27.

Basis

The isolation instrumentation listed in Item 3.a isolates the reactor water cleanup system supply valves in the event of a rupture of the reactor water cleanup system. This minimizes the amount of radioactivity, energy, and inventory lost from the reactor vessel.

Currently, the Technical Specifications list Instrument Number G31-dFS-N603-1A,1B under Item 3.a. The proposed change will revise that number to G31-FDS-N603-1A,1B. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference G31-dFS-N603-1A,1B with G31-FDS-N603-1A,1B. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device, and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 3.a of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 104

Replace "≤13" with "≤45" under the response time column and add reference to Footnote (g) in Item 3.a, "Δ Flow - High," in Table 3.3.2-3 under Reactor Water Cleanup System Isolation on Page 3/4 3-27.

Basis

The reactor water cleanup (RWCU) system isolation actuation instrumentation isolates the RWCU supply isolation valves in the event of a rupture of the RWCU system to minimize the amount of radioactivity, energy, and inventory lost from the reactor vessel. A potential rupture is sensed by any of the following signals: high delta supply/return RWCU flow, high RWCU area temperature, high delta supply/return ventilation temperature, or reactor vessel water level - low, level 2. The system also isolates on standby liquid control system initiation to prevent dilution of the sodium pentaborate solution injected by standby liquid control. The high delta supply/return RWCU flow instrumentation measures the difference between the supply and return flow rates and assumes an excessive difference indicates a rupture. A 45 second time delay is provided for the Δ flow - high instrumentation to prevent spurious isolation signals which can result when starting an RWCU pump or changing the flow path.

The current Technical Specifications specify a response time of 13 seconds for the Δ flow - high trip function. This value does not include the time delay added by time delay relay G31-R616C,D. This relay currently exists; however, its 45 second time delay is not included the response time for this instrumentation. The proposed change revises the response time from 13 seconds to 45 seconds, and adds a new reference to a new Footnote (g) which states "Includes time delay added by the time delay relay G31-R616C,D." The time delay begins upon receipt of the signal, as does the diesel generator start time. Thus, they run in parallel, and the most conservative response time to be listed is the one associated with the time delay caused by the time delay relays.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The Δ flow - high instrumentation response time currently listed in Item 3.a does not show the time delay resulting from the existing time delay relays in the logic. The original design of the plant includes a response time of 45 seconds for this instrumentation. The Technical Specifications are being revised to reflect this. There is no change to existing instrumentation, nor is there new instrumentation added. Thus, there is no increase in the possibility of an accident, nor is there any change in the consequences of any accident.

2. The proposed change revises the Technical Specifications to reflect the correct, existing information. The current information does not include information relating to the time delay relays. By providing this information, no new accident situations are created because it merely represents the existing design of the plant. No physical changes are being made to the system.

3. The proposed change does not impact the way in which the referenced information operates, nor does it change the design or testing requirements of the system. It merely references existing information in a more complete, correct manner. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, it has no impact on the margin of safety.

Proposed Change Number 105

Replace Instrument Number G31-TS-N602-A,B,C,D,E,F with Instrument Number G31-TDS-N602-A,B,C,D,E,F in Item 3.c, "Area Ventilation Δ Temperature - High," in Table 3.3.2-3 under Reactor Water Cleanup System Isolation on Page 3/4 3-27.

Basis

The isolation instrumentation listed in Item 3.c isolates the reactor water cleanup system supply valves in the event of a rupture of the reactor water cleanup system. This minimizes the amount of radioactivity, energy, and inventory lost from the reactor vessel.

Currently, the Technical Specifications list Instrument Number G31-TS-N602-A,B,C,D,E,F under Item 3.c. The proposed change will revise that number to G31-TDS-N602-A,B,C,D,E,F. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference G31-TS-N602-A,B,C,D,E,F with G31-TDS-N602-A,B,C,D,E,F. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device, and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new or different type of accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 3.c of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 106

Replace "and" with ";" in the instrument number listings in Item 3.e, "Reactor Vessel Water Level - Low, Level 2," in Table 3.3.2-3 under Reactor Water Cleanup System Isolation on Page 3/4 3-27 (BSEP-1 only).

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table, and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 107

Replace Footnote * with Footnote (c) in Item 3.e, "Reactor Vessel Water Level - Low, Level 2," in Table 3.3.2-3 under Primary Containment Isolation on Page 3/4 3-27 (BSEP-1 only).

Basis

This change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnote has not changed unless noticed elsewhere in this enclosure. The footnote has been changed from * to (c) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnotes. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Proposed Change Number 108

Add "$\leq 13^{(h)}$" in Item 3.e, "Reactor Vessel Water Level - Low, Level 2," in Table 3.3.2-3 under Secondary Containment Isolation on Page 3/4 3-27 (BSEP-1 only).

Basis

Item 3.e specifies the response time requirements for the reactor vessel water level - low, level 2 instrumentation that actuates Valve Group 3 isolations. It ensures the time it takes for the instrumentation to detect and initiate valve closure is within the allowed time assumed by the accident analysis. These limits, therefore, ensure the instrumentation reacts quickly enough to limit the amount of radioactivity, reactor inventory, and energy lost from containment to less than that determined by the accident analysis.

The isolation system response time is the sum of the instrumentation response time and the valve isolation time. For AC powered valves, the instrumentation response time also includes the emergency diesel generator start time since the power to these valves must be re-established after the loss of off-site power before the valves can stroke closed. The instrumentation response time is, therefore, the greater of the following: 1) the measured response time of the instrumentation by itself, including any time delay relays, or 2) the start time of the emergency diesel generators.

The current Technical Specifications specify an instrumentation response time of 1.0 second for the trip function instrumentation of Item 3.e. A reference to Footnote * which states, "Isolation actuation instrumentation response time only," is also provided. This footnote indicates that any time delay needed for diesel generator starting is not subject to the response time requirement since the requirement applies only to the instrumentation.

The trip function covered by Item 3.e actuates both AC powered valves and the main steam isolation valves (MSIVs). The MSIVs are not dependent on AC power for closure. The response times currently specified are appropriate for the MSIVs, but not for the other valves actuated by this instrumentation. The proposed change establishes a 13 second response time requirement for the valves other than the MSIVs. This time is consistent with the accident analysis assumptions and is sufficient to initiate isolation before any significant release of radioactivity following a LOCA. There is no response time requirement related to an RWCU line break as the trip function is not depended upon for mitigating that event. The correct response time requirements for the MSIVs are specified under Item 1.a.2.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change clarifies the existing response time requirements for the reactor vessel water level - low, level 2 instrumentation. The response time currently listed, 1.0 seconds, actually applies only to the main steam isolation valves. The new additional response time listed, 13 seconds, applies to the AC powered valves which rely on the diesel generators. The 13 seconds includes the diesel generator start times.

   The MSIV closure time is the most critical because it represents the worst case pathway for inventory, energy, and radiation loss. The response time for this instrumentation has not changed. The design, operation, reliability, or testing requirements of plant systems have not changed. The instrumentation will function in the same manner as it currently does; the response times for the referenced information will be more completely and accurately represented in the Technical Specifications and remain in agreement with the accident analysis assumptions. Thus, there is no increase in the probability of an accident, nor is there a change in the consequences of an accident.

2. The proposed change is a clarification of existing information. There is no change in the design or operation of the system. Containment isolation will still occur as assumed in the accident analysis. Since the equipment will still operate as before, this change does not create the possibility of a new or different type of accident.

3. The proposed change does not change the design or operation of any existing instrumentation or equipment. It only provides a reference to existing information which was not previously specified in the Technical Specifications. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, there is no impact on the margin of safety.

Proposed Change Number 109

Replace "$\leq$1.0*" with "$\leq$13" in Item 3.e, "Reactor Vessel Water Level - Low, Level 2," in Table 3.3.2-3 under Secondary Containment Isolation on Page 3/4 3-27 (BSEP-2 only).

Basis

Item 3.e specifies the response time requirements for the reactor vessel water level - low, level 2 instrumentation that actuates Valve Group 3 isolations. It ensures the time it takes for the instrumentation to detect and initiate valve closure is within the allowed time assumed by the accident analysis. These limits, therefore, ensure the instrumentation reacts quickly enough to limit the amount of radioactivity, reactor inventory, and energy lost from containment to less than that determined by the accident analysis.

The isolation system response time is the sum of the instrumentation response time and the valve isolation time. For AC powered valves, the instrumentation response time also includes the emergency diesel generator start time since the power to these valves must be re-established after the loss of off-site power before the valves can stroke closed. The instrumentation response time is, therefore, the greater of the following: 1) the measured response time of the instrumentation by itself, including any time delay relays, or 2) the start time of the emergency diesel generators.

The current Technical Specifications specify an instrumentation response time of 1.0 second for the trip function instrumentation of Item 3.e. A reference to Footnote * which states, "Isolation actuation instrumentation response time only," is also provided. This footnote indicates that any time delay needed for diesel generator starting is not subject to the response time requirement since the requirement applies only to the instrumentation.

The trip function covered by Item 3.e actuates only the AC powered valves and not the main steam isolation valves (MSIVs) since the trip setpoint for the MSIVs has been changed from reactor vessel water level - low, level 2 to reactor vessel water level - low, level 3 (see the September 29, 1987 submittal discussed in the cover letter). The MSIVs are not dependent on AC power for closure. The response times currently specified are appropriate for the MSIVs, but not for the other valves actuated by this instrumentation. The proposed change establishes a 13 second response time requirement which is the correct isolation time for the valves other than the MSIVs. This time is consistent with the accident analysis assumptions and is sufficient to initiate isolation before any significant release of radioactivity following a LOCA. There is no response time requirement related to an RWCU line break as the trip function is not depended upon for mitigating that event. The correct response time requirements for the MSIVs are specified under Item 1.a.2.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change clarifies the existing response time requirements for the reactor vessel water level - low, level 2 instrumentation. The response time currently listed, 1.0 seconds, actually applies only to the main steam isolation valves. The new response time listed, 13 seconds, applies to the AC powered valves which rely on the diesel generators. The 13 seconds includes the diesel generator start times.

   The MSIV closure time is the most critical because it represents the worst case pathway for inventory, energy, and radiation loss. The response time for this instrumentation has not changed and is addressed in Item 1.a.2, "Reactor Vessel Water Level - Low, Level 3." The design, operation, reliability, or testing requirements of plant systems have not changed. The instrumentation will function in the same manner as it currently does; the response times for the referenced information will be more completely and accurately represented in the Technical Specifications and remain in agreement with the accident analysis assumptions. Thus, there is no increase in the probability of an accident, nor is there a change in the consequences of an accident.

2. The proposed change is a clarification of existing information. There is no change in the design or operation of the system. Containment isolation will still occur as assumed in the accident analysis. Since the equipment will still operate as before, this change does not create the possibility of a new or different type of accident.

3. The proposed change does not change the design or operation of any existing instrumentation or equipment. It only provides a reference to existing information which was not previously specified in the Technical Specifications. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, there is no impact on the margin of safety.

Proposed Change Number 110

Add new Item 3.f, "Δ Flow - High - Time Delay Relay," and associated Instrument Number G31-R616C,D to Table 3.3.2-3 in Table 3.3.2-3 under Reactor Water Cleanup System Isolation on Page 3/4 3-27.

Basis

The reactor water cleanup (RWCU) system isolation instrumentation isolates the RWCU supply isolation valves in the event of a rupture of the RWCU system. This minimizes the amount of radioactivity, energy, and inventory lost from the reactor vessel. A potential rupture is sensed by any of the following signals: high delta supply/return RWCU flow, high RWCU area temperature, high delta supply/return ventilation temperature, or reactor vessel water level - low, level 2. The system also isolates on standby liquid control system initiation to prevent dilution of the poison injected due to cleanup from RWCU operation.

The current Technical Specifications do not specifically reference response time requirements for the existing time delay relays in the RWCU Δ flow - high logic. Initially, the time delay relays were considered to be a part of the isolation logic described in Item 3.a and were not listed separately in the table. These relays do not initiate any isolation signal; however, they are an important part of the instrumentation. It has been determined that they should be addressed separately in the table to avoid possible confusion.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change addresses response time requirements for instrumentation which is designed to detect an accident and not prevent one. These requirements are consistent with the accident analysis assumptions and are similar to those for other associated instrumentation. There is no change to the equipment or in how the equipment is operated; only references to existing instrumentation are being added to the table. Since the establishment of operability and surveillance requirements does not change the design or affect the operation of the instrumentation, there is no increase in the probability of an accident, nor is there any change in the consequences of any accident previously evaluated.

2. Specification of the correct response time requirements provides additional assurance that the instrumentation will perform its intended safety function. It does not change the design or operation of any equipment or cause it to operate in a manner not previously assumed. It does not affect any equipment which could cause an accident. It only affects instrumentation designed to detect the occurrence of an accident. Thus, this change does not create the possibility of a new accident.

3. Specification of specific response time requirements that is consistent with the other isolation instrumentation for this system will not affect or change the operation of the affected equipment. There will be no change in the reliability, response characteristics, or setpoints of the instrumentation. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, there is no change in the margin of safety.

Proposed Change Number 111

Replace Footnote ## with Footnote (e) in Item 4.a.1, "HPCI Steam Line Flow - High," in Table 3.3.2-3 under Core Standby Cooling Systems Isolation on Page 3/4 3-28.

Basis

This change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnote has not changed unless noticed elsewhere in this enclosure. The footnote has been changed from ## to (e) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnotes. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Proposed Change Number 112

Delete reference to Footnote (a) in Item 4.a.1, "HPCI Steam Line Flow - High," in Table 3.3.2-3 under Core Standby Cooling Systems Isolation on Page 3/4 3-28.

Basis

Table 3.3.2-3 specifies the response time requirements for the isolation instrumentation. It ensures that the time it takes for the instrumentation to detect and initiate valve closure is within the allowed time assumed by the accident analysis.

The isolation response time is the sum of the instrument response time and the valve isolation time. For AC powered valves, the instrument response time also includes the emergency diesel generator start time since power to these valves must be re-established after a loss of off-site power before the valves can stroke closed. The instrument response time, therefore, is the greater of the following: 1) the measured response time of the instrument itself, including any time delay relays, or 2) the start time of the emergency diesel generators.

The current Technical Specifications provide an explanation of the requirements for determining instrumentation response time in Footnote (a), which states:

(a) The isolation system instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Isolation system instrumentation response time specified includes the delay for diesel generator starting assumed in the accident analysis.

Footnote (a) is currently listed only with Items 4.a.1 and 4.b.1, which are concerned with the high pressure coolant injection (HPCI) and reactor core isolation coolint (RCIC) steam line flow - high instrumentation.

The proposed change moves the reference to Footnote (a) from just Items 4.a.1 and 4.b.1 to the header for the RESPONSE TIME column. Footnote (a) is being revised to state:

(a) The isolation system instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Isolation system instrumentation response time specified includes any delay for diesel generator starting assumed in the accident analysis.

By changing the wording to "any delay for diesel generator starting...," this footnote may be applied to all of the trip function instrumentation. It also recognizes that the response time must also include an allowance for the diesel generator start time if closure of the associated isolation valves is dependent upon the diesel generators. The diesel generator start time is a factor in the response times for most of the isolation valves, not just the HPCI and RCIC high steam flow trip functions. It does not apply to

the main steam isolation valves (MSIVs) since they are air operated and have a shorter isolation time assumed in the accident analysis.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change moves the reference to Footnote (a) from Item 4.a.1 to the column header, which indicates that the diesel generator start time is a factor in determining the response times for most isolation instrumentation. This change does not reflect a change to the plant design, operation, surveillance or testing; it merely clarifies and expands upon existing requirements and information. Therefore, there is no increase in the probability of an accident, nor is there any change in the consequences of an accident.

2. The proposed change provides a clarification to the requirements for determining response times for AC powered valves. This reflects existing information, and does not reflect a change in plant design. Therefore, the proposed change does not create the possibility of any new or different kinds of accidents.

3. The proposed change does not reflect a change to the design of the plant. It more clearly states existing requirements, and will potentially reduce confusion and misinterpretation of the requirements. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, the margin of safety is not decreased.

Proposed Change Number 113

Replace "HPCI Steam Line High Flow Time Delay Relay" with "HPCI Steam Line Flow - High Time Delay Relay" in the title of Item 4.a.2 in Table 3.3.2-3 under Core Standby Cooling Systems Isolation on Page 3/4 3-28.

Basis

The proposed change was made solely to provide consistency with other titles throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The instrumentation itself has not changed, nor has its design or operation. Only its title has been reworded to be consistent with other titles in the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the instrument design or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table. It does not involve a change in instrumentation, function, design, or operation. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 114

Replace "and" with ";" in the instrument number listings in Item 4.a.5, "Bus Power Monitor," in Table 3.3.2-3 under Core Standby Systems Isolation on Page 3/4 3-28.

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table, and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 115

Replace "HPCI Steam Line Area" with "HPCI Steam Line Area Δ Temperature - High" in the title of Item 4.a.8 in Table 3.3.2-3 under Core Standby Cooling Systems Isolation on Page 3/4 3-28.

Basis

The high pressure coolant injection (HPCI) system isolation instrumentation isolates the Group 4 isolation valves, which include the HPCI inboard and outboard steam line isolation valves and the HPCI torus suction isolation valves, in the event of a rupture of the HPCI system. This mitigates the consequences of such a break.

The current Technical Specifications have an inaccurate description of the trip function. The trip function actually monitors the difference between the HPCI area inlet and outlet air temperatures. The current Technical Specifications imply that it monitors only the area temperature. Item 4.a.7, "HPCI Steam Line Ambient Temperature - High," provides the information for the area temperature, and lists the correct operability and surveillance requirements for that instrumentation. Item 4.a.8 actually performs the area Δ temperature - high monitoring function; thus, it should have a trip function that accurately describes its function.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change replaces the title of Item 4.a.8 with one that is more applicable to the function of the instrument. The instrumentation described in Item 4.a.8 performs the area Δ temperature - high monitoring function, while Item 4.a.7 performs the area temperature - high monitoring function. This change does not involve any change to the operability, reliability, or testing requirements of the instrumentation involved; it merely provides a more comprehensive description of its function. Therefore, there is no increase in the probability of any accident previously evaluated, nor is there any change in the consequences of any accident.

2. The proposed change does not involve any physical changes to any plant systems. It provides a better description in the Technical Specifications of the actual function of the area Δ temperature monitoring instrumentation. The possibility of misinterpretation of the function of the instrumentation will be reduced. Therefore, there is no new accident possibility created.

3. The proposed change provides a more clear, concise description of the actual function of the area Δ temperature monitoring instrumentation. It does not physically alter any plant instrumentation and, therefore, does not impact the margin of safety.

Proposed Change Number 116

Replace Instrument Number E51-dTS-N604C,D with Instrument Number E51-TDS-N604C,D in Item 4.a.8, "HPCI Steam Line Area Δ Temperature - High," in Table 3.3.2-3 under Core Standby Cooling Systems Isolation on Page 3/4 3-28.

Basis

The isolation instrumentation listed in Item 4.a.8 isolates the high pressure coolant injection (HPCI) system should there be a HPCI steam line break.

Currently, the Technical Specifications list Instrument Number E51-dTS-N604C,D under Item 4.a.8. The proposed change will revise that number to E51-TDS-N604C,D. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference E51-dTS-N604C,D with E51-TDS-N604C,D. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 4.a.8 of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 117

Replace "Emergency Area Cooler Temperature - High" with "HPCI Equipment Area Temperature - High" in Item 4.a.9 in Table 3.3.2-3 under Core Standby Cooling Systems Isolation on Page 3/4 3-28.

Basis

The emergency area cooler temperature - high isolation signal isolates the Group 4 isolation valves in the event of a high pressure coolant injection (HPCI) steam line break to mitigate the consequences of such a break. The Group 4 isolation valves include the HPCI inboard and outboard steam line isolation valves and the HPCI torus suction isolation valves.

The current Technical Specifications have an inaccurate description of the trip function. The proposed change revises the description of the trip function from "Emergency Area Cooler Temperature - High" to "HPCI Equipment Area Temperature - High."

This change is necessary so that the trip function description more clearly reflects the actual area the instrumentation monitors. The existing description is a generic BWR term which does not relate to the actual BSEP design. The area the instrumentation monitors at BSEP is referred to in other plant documents and programs as the "HPCI Equipment Area." This change will reduce the potential for confusion by more clearly describing the actual location of the instrumentation.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will change the description of a trip function to more clearly identify the area the instrumentation monitors. The current wording, "Emergency Area Cooler," does not reflect the commonly used description of the area. "HPCI Equipment Area" more clearly describes the area. This change will reduce confusion concerning which area is being referred to.

   No changes to the referenced instrumentation are being reflected by this change. The instrumentation will function exactly as before. Therefore, there is no change in the probability of an accident, nor is there any change in the consequences of an accident.

2. The proposed change revises the trip function description so the appropriate trip function is clearly identified in the Technical Specifications. It does not change the required function of the instrumentation; it merely clarifies the description of an existing trip function. Thus, no new accident possibilities are created.

3. The proposed change provides a clarification in the description of an existing isolation function. It does not change the design, reliability, or function of any plant equipment; it merely references them more clearly. Thus, there is no impact on the margin of safety as a result of this change.

Proposed Change Number 118

Add new Item 4.a.10, "Drywell Pressure - High," and associated Instrument Numbers E11-PT-N011C,D and E11-PTS-N011C-2,D-2 to Table 3.3.2-3 under Core Standby Cooling Systems Isolation on Page 3/4 3-28.

Basis

Valve Group 4 isolates the high pressure coolant injection (HPCI) inboard and outboard steam line isolation valves and the HPCI torus suction isolation valves in the event of a HPCI steam line break to mitigate the consequences of the break.

The HPCI system also has turbine exhaust vacuum breaker isolation valves (E41-F075 and E41-F079) on a vacuum relief line for the HPCI turbine exhaust. This line helps prevent the creation of a water column in the exhaust line. Preventing development of this column reduces the piping loads which could exist if the turbine is restarted. The valves isolate on coincident HPCI steam line pressure - low and drywell pressure - high signals to establish primary containment. The steam line pressure - low signal indicates HPCI is isolated or no longer needed because the reactor is depressurized. The drywell pressure - high signal indicates a loss of coolant accident (LOCA) exists and primary containment isolation is needed.

The current Technical Specifications specify operability and surveillance requirements for the Group 4 isolation valves but not for the vacuum breaker isolation valves. The proposed change reflects the creation of a new Valve Group 7, which includes the HPCI turbine exhaust vacuum breakers E41-F075 and E41-F079. Operability and surveillance requirements for the actuation instrumentation are addressed under Item 4.a.3 for the HPCI steam supply pressure - low instrumentation portion of the logic. This change adds a new Item 4.a.10 which addresses the response time requirements for the existing instrumentation for the drywell pressure - high trip function associated with the valves in Group 7. No response time requirement is necessary for this instrumentation because they are on a closed line outside containment and this line does not communicate with any system outside containment. Closure of the Group 7 valves is not depended upon to limit the release of radioactivity from primary containment during a LOCA, so the response time of the instrumentation is not critical or a factor in the accident analysis.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes the response time requirements for the existing drywell pressure - high instrumentation which was not originally included in the Technical Specifications. The HPCI turbine exhaust vacuum breakers isolate on coincident HPCI steam line pressure - low and drywell pressure - high. The steam line pressure - low signal indicates HPCI is isolated or no longer needed because the reactor is depressurized, and the drywell pressure - high signal indicates a LOCA exists and primary containment isolation is needed.

This change adds response time requirements for existing instrumentation to provide additional assurance that primary containment is isolated when necessary. No new equipment is being added, and existing equipment will perform its safety function in the same manner as before. Existing requirements are being added to the Technical Specifications for clarity and completeness. Thus, there is no change in the probability of an accident, nor is there any change in the consequences of any accident.

2. The proposed change adds response time requirements to the Technical Specifications for existing instrumentation. This will provide a more complete and accurate document, thereby enhancing primary containment isolation assurance. It does not provide an opportunity for occurrence of any new accidents because no equipment, operability requirements, or setpoints are being altered.

3. The proposed change adds requirements for existing instrumentation to the Technical Specifications. It does not reflect a physical change to the plant systems; it only provides a more complete and accurate description of existing equipment. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, there is no impact on the margin of safety of the plant.

Proposed Change Number 119

Replace Footnote ### with Footnote (f) in Item 4.b.1, "RCIC Steam Line Flow - High," in Table 3.3.2-3 under Core Standby Cooling Systems Isolation on Page 3/4 3-29.

Basis

This change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnotes has not changed unless noticed elsewhere in this enclosure. The footnote has been changed from ### to (f) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnotes. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Proposed Change Number 120

Delete reference to Footnote (a) in Item 4.b.1, "RCIC Steam Line Flow - High," in Table 3.3.2-3 under Core Standby Cooling Systems Isolation on Page 3/4 3-29.

Basis

Table 3.3.2-3 specifies the response time requirements for the isolation instrumentation. It ensures that the time it takes for the instrumentation to detect and initiate valve closure is within the allowed time assumed by the accident analysis.

The isolation response time is the sum of the instrument response time and the valve isolation time. For AC powered valves, the instrument response time also includes the emergency diesel generator start time since power to these valves must be re-established after a loss of off-site power before the valves can stroke closed. The instrument response time, therefore, is the greater of the following: 1) the measured response time of the instrument itself, including any time delay relays, or 2) the start time of the emergency diesel generators.

The current Technical Specifications provide an explanation of the requirements for determining instrumentation response time in Footnote (a), which states:

> (a) The isolation system instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Isolation system instrumentation response time specified includes the delay for diesel generator starting assumed in the accident analysis.

Footnote (a) is currently listed only with Items 4.a.1 and 4.b.1, which are concerned with the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) steam line flow - high instrumentation.

The proposed change moves the reference to Footnote (a) from just Items 4.a.1 and 4.b.1 to the header for the RESPONSE TIME column. Footnote (a) is being revised to state:

> (a) The isolation system instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Isolation system instrumentation response time specified includes any delay for diesel generator starting assumed in the accident analysis.

By changing the wording to "any delay for diesel generator starting...," this footnote may be applied to all of the trip function instrumentation. It also recognizes that the response time must also include an allowance for the diesel generator start time if closure of the associated isolation valves is dependent upon the diesel generators. The diesel generator start time is a factor in the response times for most of the isolation valves, not just the HPCI and RCIC steam flow - high trip functions. It does not apply

to the main steam isolation valves (MSIVs) since they are air operated and have a shorter isolation time assumed in the accident analysis.

<u>10CFR50.92 Evaluation</u>

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change moves the reference to Footnote (a) from Item 4.b.1 to the column header, which indicates that the diesel generator start time is a factor in determining the response times for most isolation instrumentation. This change does not reflect a change to the plant design, operation, surveillance or testing; it merely clarifies and expands upon existing requirements and information. Therefore, there is no increase in the probability of an accident, nor is there any change in the consequences of an accident.

2. The proposed change provides a clarification to the requirements for determining response times for AC powered valves. This reflects existing information, and does not reflect a change in plant design. Therefore, the proposed change does not create the possibility of any new or different kinds of accidents.

3. The proposed change does not reflect a change to the design of the plant. It more clearly states existing requirements, and will potentially reduce confusion and misinterpretation of the requirements. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, the margin of safety is not decreased.

Proposed Change Number 121

Replace "RCIC Steam Line High Flow Time Delay Relay" with "RCIC Steam Line Flow - High Time Delay Relay" in the title of Item 4.b.2 on Table 3.3.2-3 under Core Standby Cooling Systems Isolation on Page 3/4 3-29.

Basis

The proposed change was made solely to provide consistency with other titles throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The instrumentation itself has not changed, nor has its function, design, or operation. Only its title has been reworded to be consistent with other titles in the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table. It does not involve a change in instrumentation, function, design, or operation. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 122

Replace "and" with ";" in the instrument number listings in Item 4.b.5, "Bus Power Monitor," in Table 3.3.2-3 under Core Standby Cooling Systems Isolation on Page 3/4 3-29.

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table, and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 123

Replace Instrument Number E51-dTS-N604A,B with Instrument Number E51-TDS-N604A,B in Item 4.b.8, "RCIC Steam Line Area Δ Temperature - High," in Table 3.3.2-3 under Core Standby Cooling Systems Isolation on Page 3/4 3-29.

Basis

The isolation instrumentation listed in Item 4.b.8 isolates the reactor core isolation cooling (RCIC) system should there be a RCIC steam line break.

Currently, the Technical Specifications list Instrument Number E51-dTS-N604A,B under Item 4.b.8. The proposed change will revise that number to E51-TDS-N604A,B. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference E51-dTS-N604A,B with E51-TDS-N604A,B. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 4.b.8 of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 124

Replace "Emergency Area Cooler Temperature - High" with "RCIC Equipment Room Ambient Temperature - High" in the title of Item 4.b.9 in Table 3.3.2-3 under Core Standby Cooling Systems Isolation on Page 3/4 3-29.

Basis

The emergency area cooler temperature - high isolation signal isolates the Group 5 isolation valves in the event of a reactor core isolation cooling (RCIC) steam line break to mitigate the consequences of such a break. The Group 5 isolation valves include the RCIC inboard and outboard steam line isolation valves.

The current Technical Specifications have an inaccurate description of the trip function. The proposed change revises the description of the trip function from "Emergency Area Cooler Temperature - High" to "RCIC Equipment Room Ambient Temperature - High."

This change is necessary so that the trip function description more clearly reflects the actual area the instrumentation monitors. The existing description is a generic BWR term which does not relate to the actual BSEP design. The area the instrumentation monitors at BSEP is referred to in other plant documents and programs as the "RCIC Equipment Area." This change will reduce the potential for confusion by more clearly describing the actual location of the instrumentation.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will change the description of a trip function to more clearly identify the area the instrumentation monitors. The current wording, "Emergency Area Cooler," does not reflect the commonly used description of the area. "RCIC Equipment Area" more clearly describes the area. This change will reduce confusion concerning which area is being referred to.

   No changes to the referenced instrumentation are being reflected by this change. The instrumentation will function exactly as before. Therefore, there is no change in the probability of an accident, nor is there any change in the consequences of an accident.

2. The proposed change revises the trip function so the appropriate trip function is clearly identified in the Technical Specifications. It does not change the required function of the instrumentation; it merely clarifies the description of an existing trip function. Thus, no new accident possibilities are created.

3. The proposed change provides a clarification in the description of an existing isolation function. It does not change the design, reliability, or function of any plant equipment; it merely references them more clearly. Thus, there is no impact on the margin of safety as a result of this change.

Proposed Change Number 125

Replace Instrument Number E51-dTS-N601A,B with Instrument Number E51-TDS-N601A,B in Item 4.b.10, "RCIC Equipment Room Δ Temperature - High," in Table 3.3.2-3 under Core Standby Cooling Systems Isolation on Page 3/4 3-29.

Basis

The isolation instrumentation listed in Item 4.b.10 isolates the reactor core isolation cooling (RCIC) system should there be a RCIC steam line break.

Currently, the Technical Specifications list Instrument Number E51-dTS-N601A,B under Item 4.b.10. The proposed change will revise that number to E51-TDS-N601A,B. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference E51-dTS-N601A,B with E51-TDS-N601A,B. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 4.b.10 of the table. The change does not represent any changes to the instrument, its function or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 126

Add new Item 4.b.11, "RCIC Steam Line Tunnel Temperature - High Time Delay Relay," and associated Instrument Number E51-KC-M602A,B to Table 3.3.2-3 under Core Standby Cooling Systems Isolation on Page 3/4 3-29.

Basis

The reactor core isolation cooling (RCIC) steam line tunnel temperature - high, time delay relay instrumentation isolates the Group 5 isolation valves in the event of a RCIC steam line break to mitigate the consequences of the break. The Group 5 valves include the RCIC inboard and outboard steam line isolation valves.

The current Technical Specifications do not specifically reference operability and surveillance requirements for the existing RCIC steam line tunnel high temperature time delay relays. The proposed change adds response time requirements for these relays. A specific response time requirement is not needed for these time delay relays since they are part of the logic under Item 4.b.7 and are included within its response time requirements. Initially, the time delay relays were considered to be a part of the isolation logic described in Item 4.b.7 and were not listed separately in the table. These relays do not initiate any isolation signal; however, they are an important part of the instrumentation, and it has been determined that they should be addressed separately in the table to avoid possible confusion.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change addresses response time requirements for instrumentation which is designed to detect an accident and not prevent one. These requirements are consistent with the accident analysis assumptions and are similar to those for other associated instrumentation. There is no change to the equipment or how the equipment is operated; only references to existing instrumentation are being added to the table. Since the establishment of response time requirements does not change the design or affect the operation of the instrumentation, there is no increase in the probability of an accident, nor is there any change in the consequences of any accident previously evaluated.

2. Specification of the correct response time requirements provides additional assurance that the instrumentation will perform its intended safety function. It does not change the design or operation of any equipment or cause it to operate in a manner not previously assumed. It does not affect any equipment which could cause an accident. It only affects instrumentation designed to detect the occurrence of an accident. Thus, this change does not create the possibility of a new accident.

3. Specification of specific response time requirements that are consistent with the other isolation instrumentation for this system will not affect or change the operation of the affected equipment. There will be no change in the reliability, response characteristics, or setpoints of the instrumentation. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, there is no change in the margin of safety.

Proposed Change Number 127

Add new Item 4.b.12, "Drywell Pressure - High," and associated Instrument Numbers E11-PT-N011A,B and E11-PTS-N011A-2,B-2 to Table 3.3.2-3 under Core Standby Cooling Systems Isolation on Page 3/4 3-29.

Basis

Valve Group 5 isolates the reactor core isolation coolin (RCIC) steam line to mitigate the consequences of a break. The Group 5 isolation valves include the RCIC inboard and outboard steam line isolation valves.

The RCIC system also has turbine exhaust vacuum breaker isolation valves (E51-F062 and E51-F066) on a vacuum relief line for the RCIC turbine exhaust. The line helps prevent the development of a water column in the exhaust line. Preventing development of this column reduces the piping loads which could exist if the turbine is restarted. The valves isolate on coincident RCIC steam line pressure - low and drywell pressure - high signals to establish primary containment isolation. The steam line pressure - low signal indicates RCIC is isolated or no longer needed because the reactor is depressurized. The drywell pressure - high signal indicates a LOCA exists and primary containment isolation is needed.

The current Technical Specifications specify operability and maintenance requirements for the Group 5 isolation valves but not for the vacuum breaker isolation valves. The proposed change reflects the creation of the new Valve Group 9, which includes the RCIC turbine exhaust vacuum breakers E51-F062 and E51-F066.

The isolation logic associated with these vacuum breakers requires a signal from both the drywell pressure - high instrumentation and the RCIC steam supply pressure - low instrumentation. This change adds a new item to address the response time requirements for the existing instrumentation for the drywell pressure - high trip function associated with the existing valves which form the new Valve Group 9. No response time requirement is necessary for this instrumentation because they are on a closed line outside containment and this line does not communicate with any system outside containment. Closure of the Group 9 valves is not depended upon to limit the release of radioactivity from primary containment during a LOCA, so the response time of the instrumentation is not critical or a factor in the accident analysis.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes the response time information for the drywell pressure - high instrumentation which was not originally included in the Technical Specifications. This change is being made to address the addition of the RCIC turbine exhaust vacuum breakers, which isolate on coincident RCIC steam line pressure - low and drywell pressure - high, to the Technical Specifications via addition of Valve Group 9. The steam line pressure - low signal indicates RCIC is isolated or no longer needed because the reactor is depressurized, and the drywell pressure - high signal indicates a LOCA exists and primary containment isolation is needed.

   This change adds response time requirements for existing instrumentation to provide additional assurance that primary containment is isolated when necessary. No new equipment is being added, and existing equipment will perform its safety function in the same manner as before. Existing requirements are being added to the Technical Specifications for clarity and completeness. Thus, there is no change in the probability of an accident, nor is there any change in the consequences of any accident.

2. The proposed change adds response time requirements to the Technical Specifications for existing instrumentation. This will provide a more complete and accurate document, thereby enhancing primary containment isolation assurance. It does not provide an opportunity for any new accidents because no equipment, operability requirements, or setpoints are being altered.

3. The proposed change adds requirements for existing instrumentation to the Technical Specifications. It does not reflect a physical change to the plant systems; it only provides a more complete and accurate description of existing equipment. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, there is no impact on the margin of safety of the plant.

Proposed Change Number 128

Replace Instrument Number B32-PS-N018A,B with Instrument Number B32-PS-N018A-1,B in Item 5.b, "Reactor Steam Dome Pressure - High," in Table 3.3.2-3 under Core Standby Cooling Systems Isolation on Page 3/4 3-29a.

Basis

The isolation instrumentation listed in Item 5.b isolates the shutdown cooling system should there be a malfunction or rupture of the residual heat removal system while operating in shutdown cooling mode. This minimizes reactor inventory loss.

Currently, the Technical Specifications list Instrument Number B32-PS-N018A,B under Item 5.b. The proposed change will revise that number to B32-PS-N018A-1,B. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference B32-PS-N018A,B with B32-PS-N018A-1,B. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 5.b of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Propos‹ l Change Number 129

Replace "...shown in Table 3.6.3-1..." with "...shown in plant procedure ________ ..." in Footnote (d) on Page 3/4 3-29b.

Basis

Technical Specification Section 3.6 addresses containment systems. Table 3.6.3-1 provides a partial listing of the automatic primary containment isolation valves (PCIVs). The Brunswick Plant Technical Specifications currently require the following surveillances:

> Specification 4.6.3.1 requires that primary containment isolation valves be demonstrated operable by cycling and isolation time testing following maintenance, repair, or replacement work.
>
> Specification 4.6.3.2 requires verification of isolation valve operability through the application of an isolation test signal to verify the valve actuates properly. This testing is required every 18 months.
>
> Specification 4.6.3.3 requires verification of the isolation time for power-operated and automatic valves.
>
> Specification 4.6.4.4 requires that reactor instrumentation system isolation valves be demonstrated operable by cycling the valve every 18 months.
>
> Specification 4.6.1.2.d requires Type B and C leak testing of primary containment isolation valves.

The Company has completed a review of the primary containment isolation system (PCIS) Technical Specifications and determined that certain changes to Table 3.6.3-1 are needed to accurately reflect the as-built design of the primary containment isolation system. However, rather than propose changes to the table, CP&L believes it is more appropriate to request removal of Table 3.6.3-1 from the Technical Specifications and replace references to the table with references to valve listings contained in plant procedures (the Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal). Our reasons for this determination are described below.

First, removal of the valve listings from the Technical Specifications is appropriate in the context of Technical Specification reform. The recent industry and NRC efforts to improve the Technical Specifications have identified component listings as items for removal from the Technical Specifications because they do not directly contribute to plant safety. In addition, the level of detail of such listings has resulted in frequent requests for license amendments. This process has resulted in inherent inaccuracies in the Technical Specifications due to the delays in receiving license amendments brought about by the current regulatory environment. The NRC has already recommended and approved removal of the analogous snubber listing from the Technical Specifications. In addition, tables relating to fire protection devices and containment penetration protection circuit

breakers have been removed from the Technical Specifications for other dockets. For example, listings relating to containment isolation valves, snubbers, containment penetration protection devices, motor operated valve thermal overload protection, the reactor vessel material surveillance program, engineered safety features actuation system instrumentation response times, and reactor trip system instrumentation response times have been removed from the Technical Specifications for the Shearon Harris Plant and references to plant procedures incorporated. The use of component listings through reference to plant procedures ensures that timely information is maintained that reflects recent plant modifications.

Second, changes to these plant procedures are strictly controlled under 10CFR50.59. In addition, the Technical Specifications requirement to maintain primary containment integrity (Technical Specification 3.6.1.1), as well as ASME Code Section XI testing requirements (Technical Specification 4.0.5) and 10CFR50, Appendix J requirements (Technical Specification 3.6.1.2) are maintained. Only the listing of valves would be subject to licensee control under 10CFR50.59. Plant procedures are auditable and subject to inspection by the NRC.

Third, appropriate mechanisms and controls exist to ensure correct testing for the valves through plant procedures. As discussed above, these referenced procedure(s) will incorporate valve list information developed to meet the requirements of Technical Specifications 4.6.3.1, 4.6.3.2, 4.6.3.3 and 4.6.3.4. In addition, Appendix J of 10CFR50 (implemented through Technical Specification 3/4.6.1.2) specifies leak rate test requirements for containment isolation valves, and thus maintenance of a list of such valves. Section XI of the ASME Code (implemented through Technical Specification 4.0.5) requires once per cycle testing of valve closure and timing for certain of the valves currently listed in Table 3.6.3-1. In addition, Technical Specification 4.3.2.2 requires once per cycle performance of a logic system functional test for most of the valves currently listed in Table 3.6.3-1. All the above requirements are appropriately integrated into plant procedure(s) which are subject to change only under requirements of 10CFR50.59. These plant procedures cover testing (to or beyond the present Technical Specification requirements) for the valves currently included in Table 3.6.3-1. A complete listing of containment isolation valves and required closure times is also provided as Table 6.2.4-2 to the Brunswick Updated Final Safety Analysis Report. References to plant procedures will replace references to Table 3.6.3-1 in the text of the Technical Specifications. Specific closure time testing requirements for the main steam isolation valves will be maintained in the Technical Specifications under existing Specification 3/4.4.7.

<u>10CFR50.92 Evaluation:</u>

The proposed change does not involve a significant hazards consideration for the following reasons:

1. Removal of the primary containment isolation valve listing from the Technical Specifications is appropriate in the context of Technical Specification reform. Recent industry and NRC efforts to improve the Technical Specifications have identified component listings as an item for removal because they do not directly contribute to plant safety. The use of component listings through reference to plant procedures will ensure that timely information is maintained that reflects recent plant modifications. Changes to these procedures will be controlled under the requirements of 10CFR50.59. Primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specifications 3.6.1.1 and 3.6.3 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3 (and thus the proper identification, control, and periodic testing of the appropriate containment isolation valves). Relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not change their design or operation, nor will it affect any of the accident analyses. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. As discussed in Item 1 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3 (and thus the proper identification, control, and periodic testing of the appropriate containment isolation valves). Thus, relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not change their design or operation. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. As discussed in Items 1 and 2 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3. Furthermore, the relocation of the primary containment isolation valve listing to plant procedures will clarify the Technical Specifications and ensure a more up to date and accurate primary containment isolation valve listing, thereby avoiding possible operator confusion. Therefore, since there is no change in containment design or pressure retaining capability, the proposed change does not involve a significant reduction in the margin of safety.

Proposed Change Number 130

Replace "...each valve group...and Table 3.6.5.2-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve" with "..each valve group/damper...for valves in each valve group and secondary containment isolation dampers to obtain ISOLATION SYSTEM RESPONSE TIME for each valve/damper." in Footnote (d) on Page 3/4 3-29b.

Basis

The Company is requesting the removal of Table 3.6.5.2-1 from the Technical Specifications, with the information being removed relocated to a plant procedure. The Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal. Our reasons for proposing this change are described below.

The Company has completed a review of the primary containment isolation system (PCIS) Technical Specifications and determined that certain changes to Table 3.6.3-1 are needed to accurately reflect the as-built design of the primary containment isolation system. However, rather than propose changes to the table, CP&L believes it is more appropriate to request removal of Table 3.6.3-1 from the Technical Specifications and replace references to the table with references to valve listings contained in plant procedures. The design function of the primary containment isolation valves is to preserve primary containment integrity. The design function of the secondary containment isolation dampers is similar to that for the primary containment isolation valves (i.e., preservation of secondary containment integrity). Since the Company is proposing to relocate the listing of primary containment isolation dampers from the Technical Specifications to plant procedures, we believe it is also appropriate to request removal of the Table 3.6.5.2-1 from the Technical Specifications.

Table 3.6.5.2-1 provides a listing of secondary containment isolation dampers. The Brunswick Plant Technical Specifications currently require the following surveillances:

> Specification 4.6.5.2.a requires that secondary containment isolation dampers be demonstrated operable by cycling the valve every 92 days.
>
> Specification 4.6.5.2.b requires that secondary containment isolation dampers be demonstrated operable by cycling and isolation time testing following maintenance, repair, or replacement work.
>
> Specification 4.6.5.2.c.1 requires the cycling of and verification of isolation time for automatic dampers.
>
> Specification 4.6.5.2.c.2 requires verification of isolation damper operability through the application of an isolation test signal to verify the damper actuates properly. This testing is required every 18 months.

First, removal of the secondary containment isolation damper listing from the Technical Specifications is appropriate in the context of Technical Specification reform. The recent industry and NRC efforts to improve the

Technical Specifications have identified component listings as items for removal from the Technical Specifications because they do not directly contribute to plant safety. In addition, the level of detail of such listings has resulted in frequent requests for license amendments. This process has resulted in inherent inaccuracies in the Technical Specifications due to the delays in receiving license amendments brought about by the current regulatory environment. The NRC has already recommended and approved removal of the analogous snubber listing from the Technical Specifications. In addition, tables relating to fire protection devices and containment penetration protection circuit breakers have been removed from the Technical Specifications for other dockets. For example, listings relating to containment isolation valves, snubbers, containment penetration protection devices, motor operated valve thermal overload protection, the reactor vessel material surveillance program, engineered safety features actuation system instrumentation response times, and reactor trip system instrumentation response times have been removed from the Technical Specifications for the Shearon Harris Plant and references to plant procedures incorporated. The use of component listings through reference to plant procedures ensures that timely information is maintained that reflects recent plant modifications.

Second, changes to these plant procedures are strictly controlled under 10CFR50.59. In addition, the Technical Specifications requirement to maintain secondary containment integrity (Technical Specification 3.6.5.1), are maintained. Only the listing of secondary containment isolation dampers would be subject to licensee control under 10CFR50.59. Plant procedures are auditable and subject to inspection by the NRC.

Third, appropriate mechanisms and controls exist to ensure correct testing for the secondary containment isolation dampers through plant procedures. As discussed above, these referenced procedure(s) will incorporate isolation damper list information developed to meet the requirements of Technical Specifications 4.6.5.2.a, 4.6.5.2.b, 4.6.5.2.c.1 and 4.6.5.2.c.2. The above requirements are appropriately integrated into plant procedure(s) which are subject to change only under requirements of 10CFR50.59. These plant procedures cover testing (to or beyond the present Technical Specification requirements) for the secondary containment isolation dampers currently included in Table 3.6.5.2-1. References to plant procedures will replace references to Table 3.6.5.2-1 in the text of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. Removal of the secondary containment isolation damper listing from the Technical Specifications is appropriate in the context of Technical Specification reform. Recent industry and NRC efforts to improve the Technical Specifications have identified component listings as an item for removal because they do not directly contribute to plant safety. The use of component listings through reference to plant procedures will ensure that timely information is maintained that reflects recent plant modifications. Changes to these procedures will be controlled under the requirements of 10CFR50.59. Secondary containment integrity is the function preserved by the secondary containment isolation dampers. Technical Specification 3.6.5.1 will continue to require secondary containment integrity while operating in Operational Conditions 1, 2, 3, 5 and when moving irradiated fuel assemblies in the secondary containment. Thus, the proper identification, control, and periodic testing of the appropriate secondary containment isolation dampers is necessary. Relocation of the secondary containment isolation damper listing from the Technical Specifications to plant procedures does not change their design or operation, nor does it affect any of the accident analyses; therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. As discussed in Item 1 above, secondary containment integrity is the function preserved by the secondary containment isolation dampers. Technical Specification 3.6.5.1 will continue to require secondary containment integrity while operating in Operational Conditions1, 2, 3, 5 and when moving irradiated fuel assemblies in the secondary containment (and thus the proper identification, control, and periodic testing of the appropriate secondary containment isolation dampers). Thus, relocation of the secondary containment isolation damper listing from the Technical Specifications to plant procedures does not change the design or operation of the system. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. As discussed in Items 1 and 2 above, secondary containment integrity is the function preserved by the secondary containment isolation dampers. Technical Specification 3.6.5.1 will continue to require secondary containment integrity while operating in Operational Conditions 1, 2, 3, 5 and when moving irradiated fuel assemblies in the secondary containment. Furthermore, the relocation of the secondary containment isolation damper listing to plant procedures will clarify the Technical Specifications and ensure a more up to date and accurate secondary containment isolation damper listing, thereby avoiding possible operator confusion. Since there is no change in containment design or pressure retaining capability, the proposed change does not involve a significant reduction in the margin of safety.

Proposed Change Number 131

Replace * with (c) and add "for MSIVs only. No diesel generator delays assumed" to the end of the footnote on Page 3/4 3-29b.

Basis

The isolation system response time is the sum of the instrumentation response time and the valve isolation time. For AC powered valves, the instrumentation response time must also include the emergency diesel generator start time since the power to these valves must be re-established after the loss of off-site power before the valves can stroke closed. The instrumentation response time is, therefore, the greater of the following: 1) the measured response time of the instrumentation by itself, including any time delay relays, or 2) the start time of the emergency diesel generators.

The current Technical Specifications show a Footnote * which states, "Isolation actuation instrumentation response time only." This footnote indicates that any time delay needed for diesel generator starting is not included in the response time requirement since the requirement applies only to the instrumentation.

Both AC powered valves and the main steam isolation valves (MSIVs) may be actuated by the trip functions that reference this footnote. The MSIVs are not dependent on AC power for closure. The response times currently specified are appropriate for the MSIVs, but not for the other valves which may be actuated by this instrumentation and are dependent upon the start of the diesel generators. The proposed change makes this note applicable only to the MSIVs. The proposed change revises Footnote * to (c), which makes it consistent with other nomenclature throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change replaces the current Footnote * with Footnote (c) which will provide consistency with other nomenclature throughout the table. It also revises the text of the footnote so it applies only the MSIVs, and not to the AC powered valves that are actuated by the same instrumentation. A new Footnote (h) has been created to deal with the AC powered valves and their response time.

The MSIV closure time is the most critical because it represents the worst case pathway for inventory, energy, and radiation loss. The response time for the MSIVs has not changed. The response time for other valves remains consistent with the current design and accident analysis assumptions. The design, operation, and reliability requirements of plant systems has not changed. The instrumentation will function in the same manner as it currently does; the response times for the referenced information will be more completely and accurately represented in the Technical Specifications. Thus, there is no increase in the probability of an accident, nor is there a change in the consequences of an accident.

2. The proposed change is a clarification of existing information. The MSIVs have a shorter response time than other AC valves actuated by the same instrumentation, because they do not need to include the diesel generator start times. The AC powered valves are dependent on the diesel generators, and, therefore, diesel generator delays need to be utilized in calculation of the response times for those valves. This information is now referenced in the Technical Specifications, making it a more complete and accurate document. No changes are being made to the design or operation requirements of the existing instrumentation; the proposed change only provides reference to existing information. The equipment will continue to function in the same manner as previously assumed. Therefore, it does not create the possibility of a new or different type of accident.

3. The proposed change does not change the design or operation of any existing instrumentation or equipment. It only provides reference to existing information which was not previously referenced in the Technical Specifications. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, there is no impact on the margin of safety.

Proposed Change Number 132

Replace # with (d) in the footnote listing on Page 3/4 3-29b.

Basis

This change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnote has not changed unless noticed elsewhere in this enclosure. The footnote has been changed from # to (d) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnotes. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Proposed Change Number 133

Replace ## with (e) in the footnote listing on Page 3/4 3-29b.

Basis

This change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnote has not changed unless noticed elsewhere in this enclosure. The footnote has been changed from ## to (e) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnotes. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Proposed Change Number 134

Replace ### with (f) in the footnote listing on Page 3/4 3-29b.

Basis

This change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnote has not changed unless noticed elsewhere in this enclosure. The footnote has been changed from ### to (f) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnotes. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Proposed Change Number 135

Add new Footnote (g), "Includes time delay added by the time delay relays (G31-R616C,D)," to the footnotes on Page 3/4 3-29b.

Basis

The reactor water cleanup (RWCU) system isolation actuation instrumentation isolates the RWCU supply isolation valves in the event of a rupture of the RWCU system to minimize the amount of radioactivity, energy, and inventory lost from the reactor vessel. A potential rupture is sensed by any of the following signals: high delta supply/return RWCU flow, high RWCU area temperature, high delta supply/return ventilation temperature, or reactor vessel water level - low, level 2. The system also isolates on standby liquid control system initiation to prevent dilution of the sodium pentaborate solution injected by standby liquid control. The high delta supply/return RWCU flow instrumentation measures the difference between the supply and return flow rates and assumes an excessive difference indicates a rupture. A 45 second time delay is provided for the Δ flow - high instrumentation to prevent spurious isolation signals which can result when starting an RWCU pump or changing the flow path.

Item 3.a of the current Technical Specifications specifies a response time of 13 seconds for the Δ flow - high trip function. This value does not reflect the 45 second time delay added by time delay relays G31-R616C,D, and therefore, the response time is being changed to 45 seconds. That change is addressed elsewhere in this submittal. These time delay relays currently exist; however, the associated time delays are not referenced in the response time for this instrumentation. The proposed change adds a new Footnote (g) which states "Includes time delay added by the time delay relay G31-R616C,D," which is referenced in Item 3.a.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The Δ flow - high instrumentation response time currently does not recognize the time delay resulting from the existing time delay relays in the logic. The original design of the plant includes a response time of 45 seconds for this instrumentation. The Technical Specifications are being revised to reflect this. There is no change to existing instrumentation or in the way it is assumed to operate, nor is there new instrumentation added. Thus, there is no increase in the probability of an accident, nor is there any change in the consequences of any accident.

2. The proposed change revises the Technical Specifications to reflect the correct, existing information. The current information does not include information relating to the time delay relays. By providing this information, no new accident situations are created because it merely represents the existing design of the plant. No physical changes are being made to the system, and there will be no change in how the equipment operates.

3. The proposed change does not impact the way in which the referenced instrumentation operates, nor does it change the design or testing requirements of the system. It merely references existing information in a more complete, correct manner. It does not change the design or capability of the primary containment system, nor does it change the demands placed upon this process barrier. Thus, it does not have any impact on the margin of safety.

Proposed Change Number 136

Add new Footnote (h), "Isolation system instrumentation response time for associated valves except MSIVs," to the footnotes on Page 3/4 3-29b.

Basis

The isolation system response time is the sum of the instrumentation response time and the valve isolation time. For AC powered valves, the instrumentation response time also includes the emergency diesel generator start time since the power to these valves must be re-established after the loss of off-site power before the valves can stroke closed. The instrumentation response time is, therefore, the greater of the following: 1) the measured response time of the instrumentation by itself, including any time delay relays, or 2) the start time of the emergency diesel generators.

The current Technical Specifications show a Footnote * which states, "Isolation actuation instrumentation response time only." This footnote indicates that any time delay needed for diesel generator starting is not included in the response time requirement since the requirement applies only to the instrumentation.

Both AC powered valves and the main steam isolation valves (MSIVs) may be actuated by the trip functions which reference this footnote. The MSIVs are not dependent on AC power for closure. The response times currently specified are appropriate for the MSIVs, but not for the other valves actuated by this instrumentation. The proposed change adds a new Footnote (h) which states, "Isolation system instrumentation response time for associated valves except MSIVs," which applies to valves other than the MSIVs. This change is being made in conjunction with the proposed changes to Footnote (c) which make that note applicable only to the MSIVs.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change adds a new Footnote (h) which states "Isolation system instrumentation response time for associated valves except MSIVs." This note applies only to AC powered valves that are dependent upon the diesel generators for actuation. The response times associated with these valves are longer than those associated with the MSIVs because the diesel generator startup times are included and a longer isolation time assumed in the accident analysis.

The MSIV closure time is the most critical because it represents the worst case pathway for inventory, energy, and radiation loss. The response time for the MSIVs has not changed. The response time for other valves remains consistent with the current design and accident analysis assumptions. The design, operation, and reliability of plant systems have not changed. The instrumentation will function in the same manner as it currently does; the response times for the referenced information will be more completely and accurately represented in the Technical Specifications. Thus, there is no increase in the probability of an accident, nor is there a change in the consequences of an accident.

2. The proposed change is a clarification of existing information. The MSIVs have a shorter response time than other AC valves actuated by the same instrumentation, because they do not need to include the diesel generator start times. The AC powered valves are dependent on the diesel generators, and therefore, diesel generator delays need to be utilized in calculation of the response times for those valves. This information is now referenced in the Technical Specifications, making it a more complete and accurate document. No changes are being made to the design or operation of the existing instrumentation; the proposed change only provides reference to existing information. The equipment will continue to function in the same manner as previously assumed. Therefore, the change does not create the possibility of a new or different type of accident.

3. The proposed change does not change the design or operation of any existing instrumentation or equipment. It does not change the design or capability of the primary containment, nor does it change the demands placed upon this process barrier. It only provides reference to existing information which was not previously referenced in the Technical Specifications. Thus, there is no impact on the margin of safety.

Proposed Change Number 137

Delete Item 1.a.2, "Reactor Vessel Water Level - Low, Level 2," under Primary Containment Isolation from Table 4.3.2-1 and re-label Item 1.a.3 as Item 1.a.2 on Page 3/4 3-29c (BSEP-2 only).

Basis

The Technical Specification change request submitted on September 29, 1987, as supplemented on October 14, 1987 and November 24, 1987, revised the reactor vessel water level trip function for the Valve Group 1 isolation valves from reactor vessel water level - low, level 2 to reactor vessel water level - low, level 3. This resulted in only Valve Group 3 being actuated by the reactor vessel water level - low, level 2 trip function. Valve Group 3 isolates the reactor water cleanup system and is addressed specifically in Item 3.e for the reactor vessel water level - low level 2 instrumentation. The proposed change deletes Item 1.a.2 because the instrumentation no longer actuates any valve groups that need to be addressed under Item 1. This change does not represent any physical change to the design or operation of any systems. It only more accurately describes the trip function associated with the Group 3 valves.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change deletes information which no longer belongs under Item 1. Via a previous submittal, the trip function for the Group 1 isolation valves was revised from reactor vessel water level - low, level 2 to reactor vessel water level - low, level 3. The information associated with the remaining valve group actuated by that trip function, Valve Group 3, is more appropriately referenced in Item 3.e, which describes the instrumentation that actuates reactor water cleanup system isolation. The proposed change does not reflect a change to the design or operation of the instrumentation and valve groups; it only clarifies existing information in the table. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change clarifies the table by placing the information associated with the reactor vessel water level - low, level 2 instrumentation in its appropriate place. It does not reflect a change in the design or operation of the system. Therefore, it does not create the possibility of a new or different accident.

3. The proposed change does not reflect a change to the design or operation of any equipment. It merely provides the applicable references to the reactor vessel water level - low, level 2 instrumentation in the appropriate place in the table. Therefore, there is no impact on the margin of safety.

Proposed Change Number 138

Replace "and" with ";" in the instrument number listings in Item 1.a.2, "Reactor Vessel Water Level - Low, Level 2," in Table 4.3.2-1 under Primary Containment Isolation on Page 3/4 3-29c (**BSEP-1 only**).

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table, and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability of the primary containment, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 139

Add related Instrument Number D12-RE-N006A,B,C,D to Item 1.c.1, "Main Steam Line Radiation - High," in Table 4.3.2-1 under Primary Containment Isolation on Page 3/4 3-29c.

Basis

The main steam line high radiation instrumentation detects radioactivity that may be released as a result of a control rod drop accident and provides isolation signals to the main steam line isolation valves and drain isolation valves. This limits the amount of radioactivity released from containment.

The current Technical Specifications reference only the instrument number for the radiation monitor drawer located in the control room. It does not list the associated radiation detectors which are located in the main steam isolation valve (MSIV) pit. The proposed change adds Instrument Number D12-RE-N006A,B,C,D, which represents the radiation detectors, to Item 1.c.1 in Table 4.3.2-1. These instruments are not being physically changed or added to the instrument loop; they currently exist, but are not referenced in the appropriate place in the table. By adding these instruments to the list, the instrument loop is more clearly and completely identified in the Technical Specifications. This may reduce the possibility of confusion concerning which instrument channel is subject to the requirements of Item 1.c.1.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change adds a reference to Instrument Number D12-RE-N006A,B,C,D, which represents the radiation detectors located in the MSIV pit, in Item 1.c.1 so the appropriate devices are clearly identified in the Technical Specifications in a format consistent with other plant documents and programs. It does not change the devices which perform the described function, nor does it change their ability to mitigate accidents; it simply references them more completely and correctly. This change will reduce the possibility of misunderstanding about which devices are associated with the isolation function, thereby potentially reducing errors. Therefore, it does not impact the probability or consequences of any accident previously evaluated.

2. The proposed change completes the instrument number references in Item 1.c.1 so the appropriate devices are clearly identified in the Technical Specifications in a format consistent with other plant documents and programs. The instrumentation performs the same function as before for mitigating and detecting accident conditions. It is merely being referenced more completely and correctly. This change will reduce the potential for misunderstanding about which devices are associated with the isolation function, thereby reducing the potential for errors. Therefore, no new accident possibilities are created.

3. The proposed change adds an instrument number reference so the appropriate devices are completely and clearly identified in the Technical Specifications in a format consistent with other plant documents and programs. It does not change the response, capability, reliability, or testing requirements of the instrumentation, nor does it change the ability of the instrumentation to mitigate or detect accident conditions. This change will reduce the possibility of misunderstanding about which devices are associated with the isolation function, thereby reducing potential errors. It does not change the design or capability of the primary containment, nor does it change the demands placed upon this process barrier. Therefore, there is no effect on the margin of safety.

Proposed Change Number 140

Replace Footnote # with (f) in Item 1.e, "Condenser Vacuum - Low," in Table 4.3.2-1 under Primary Containment Isolation on Page 3/4 3-29d.

Basis

This change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnote has not changed unless noticed elsewhere in this enclosure. The footnote has been changed from # to (f) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnotes. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Proposed Change Number 141

Add new Item 1.g, "Reactor Building Exhaust Radiation - High," and associated Instrument Numbers D12-RE-N010A,B and D12-RM-K609A,B to Table 4.3.2-1 under Primary Containment Isolation on Page 3/4 3-29d.

Basis

The reactor building exhaust radiation - high isolation signal causes the Group 6 isolation valves, which include the containment atmospheric control valves, and certain containment atmospheric monitoring and post-accident sampling system valves, to close during a loss of coolant accident (LOCA). This minimizes the amount of radiation released from primary containment under LOCA conditions. This signal is not of primary importance for Group 6 isolation; the reactor vessel water level - low, level 2 and drywell pressure - high signals provide Group 6 isolation signals much earlier in a LOCA scenario. These two signals directly detect a LOCA, while the reactor building exhaust radiation - high instrumentation detects radiation released from primary containment to the reactor building. It does not directly detect a LOCA and will isolate Group 6 much later than the other two signals will.

Item 1 provides information concerning isolation of the main steam lines, drywell drains, and the containment atmospheric control and monitoring systems. The main steam lines are isolated by Group 1 valves, the drywell drains and associated systems by the Group 2 valves, and the containment atmospheric control and monitoring systems by the Group 6 valves.

The new Item 1.g provides information relating to the reactor building exhaust radiation - high instrumentation surveillance requirements. This instrumentation consists of two channels; one per trip system, either of which can initiate a Group 6 isolation. The radiation monitors listed are located in the reactor building exhaust plenum and monitor normal HVAC effluent from secondary containment in the reactor building.

Item 1.g provides the appropriate operability and surveillance requirements for isolation of the Group 6 valves on a signal from the reactor building exhaust radiation - high instrumentation. These requirements are similar to those listed for the other Group 6 signals specified under Item 1 since they perform the same safety function.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes specific surveillance requirements for instrumentation which is designed to detect accident conditions, not prevent an accident. These requirements are consistent with the accident analysis assumptions and are similar to those for other instrumentation in this group. There is no change in how the equipment is operated. This change establishes operability and surveillance requirements for existing equipment and does not change the design or affect the operation of the instrumentation. Thus, there is no increase in the probability of an accident.

   The proposed change adds surveillance requirements which are appropriate for the Group 6 isolation valves and their related safety function. The existing requirements in the Technical Specifications are listed under the Secondary Containment section, and are not directly applicable to the Group 6 valves. The reactor building exhaust radiation - high signal is not depended upon for Group 6 isolation for any design basis accident; thus, there is no impact on the consequences of any accident.

2. Specifying the correct surveillance requirements provides additional assurance that the instrumentation will perform its intended safety function. It does not change the design or operation of any equipment or cause it to operate in a manner not previously evaluated. It does not affect any equipment which could cause an accident. It only affects existing instrumentation designed to detect accident conditions. Therefore, it does not create the possibility of a new accident.

3. Specifying the correct surveillance requirements does not affect any equipment which could cause an accident. It only affects instrumentation designed to detect radioactivity in secondary containment and is not depended upon for mitigating any design basis accident. Therefore, it is not a factor in the margin of safety for any design basis accident. The instrumentation is designed to detect and mitigate the design basis accidents is not affected by this change and will have the same reliability and response characteristics.

Proposed Change Number 142

Replace * with (e) in Item 2.a, "Reactor Building Exhaust Radiation - High," in Table 4.3.2-1 under Secondary Containment Isolation on Page 3/4 3-29e.

Basis

This change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnote has not changed unless specified elsewhere in this enclosure. The footnote has been changed from * to (e) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnotes. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Proposed Change Number 143

Replace Instrument Number D12-RM-N010A,B with Instrument Number D12-RE-N010A,B in Item 2.a, "Reactor Building Exhaust Radiation - High," in Table 4.3.2-1 under Secondary Containment Isolation on Page 3/4 3-29e.

Basis

A signal from the reactor building exhaust radiation - high instrumentation closes the secondary containment isolation dampers, thereby establishing secondary containment and minimizing the release of radiation from the reactor building during an accident. The dampers work in conjunction with the standby gas treatment system to establish a negative pressure in the reactor building and a controlled vent path through the standby gas treatment system filters and the stack. Secondary containment isolation would occur during a loss of coolant accident (LOCA) due to reactor vessel water level - low, level 2 or drywell pressure - high. This would minimize the release of radioactivity from the reactor building. Secondary containment isolation is also assumed to occur during a refueling accident due to a signal from the reactor building exhaust radiation - high instrumentation to contain radiation released in a fuel handling accident.

Item 2.a currently incorrectly references Instrument Number D12-RM-N010A,B as the reactor building exhaust radiation - high instrumentation. This instrument should be identified as D12-RE-N010A,B. The proposed change does not result from a plant modification; the existing instrument number is not referenced correctly. It does not represent a change to the instrumentation. The proposed change will result in the instrument loop being more clearly and completely identified in the Technical Specifications, and reduce the possibility of confusion concerning which instrument channel is subject to those Technical Specification requirements.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will result in the reactor building exhaust radiation - high instrumentation loop being more completely and correctly identified. The nomenclature will be consistent with that used in other plant documents and programs. It does not represent any change to the devices which perform the specified function; it more clearly references existing instrumentation. This change will reduce misunderstanding about which devices are associated with the isolation function. Thus, there is no change in the probability of an accident, nor is there any change in the consequences of any accident.

2. The existing instrument number listed under Item 2.a is not consistent with other plant documents and programs. There is no change to the instrument it represents; this change will only revise the instrument number to match that listed in other documentation. this change may reduce misunderstanding about which devices are associated with the isolation function. Thus, this change does not create the possibility of a new accident.

3. The proposed change will result in the reactor building exhaust radiation - high instrumentation being more correctly identified in the Technical Specifications. The nomenclature will be consistent with that used in other plant documents and programs and therefore reduce confusion about which instrumentation is being referenced. There is no physical change to the instrumentation; only its number is being revised in the Technical Specifications. Therefore, this change has no impact on the margin of safety.

Proposed Change Number 144

Add related Instrument Number D12-RM-K609A,B to Item 2.a, Reactor Building Exhaust Radiation - High," in Table 4.3.2-1 under Secondary Containment Isolation on Page 3/4 3-29d.

Basis

A signal from the reactor building exhaust radiation - high instrumentation closes the secondary containment isolation dampers, thereby establishing secondary containment isolation and minimizing the release of radiation from the reactor building during an accident. The dampers work in conjunction with the standby gas treatment system to establish a negative pressure in the reactor building and a controlled vent path through the standby gas treatment system filters and the stack Secondary containment isolation would occur during a loss of coolant accident (LOCA) due to a reactor vessel water level - low, level 2 or drywell pressure - high signal. This would minimize the release of radioactivity from the reactor building. Secondary containment isolation is also assumed to occur during a refueling accident due to a signal from the reactor building exhaust radiation - high instrumentation to contain radiation released in a fuel handling accident. Item 2.a currently lists only the radiation monitor D12-RM-N010A,B. It should also list the radiation monitor drawer D12-RM-K609A,B which is located in the control room.

The proposed change does not result from a plant modification; it is an addition of a reference to existing instrumentation which is currently associated with this system, but not listed. The instrumentation will continue to perform its intended function just as before; however, it will now be listed in its appropriate place in the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will result in the reactor building exhaust radiation - high instrumentation loop being more completely and correctly identified. The list of instruments associated with the reactor building exhaust radiation - high instrumentation will be more complete and consistent with that used in other plant documents and programs. It does not represent any change to the devices which perform the specified function; it more clearly references existing instrumentation. This change will reduce misunderstanding about which devices are associated with the isolation function. Thus, there is no change in the probability of an accident, nor is there any change in the consequences of any accident.

2. The existing instrument number listed under Item 2.a does not completely represent the instrumentation associated with the isolation instrumentation. There is no change to the instrumentation represented; the proposed change will only add an additional instrument number to completely reference the instrumentation currently listed. This change may reduce misunderstanding about which devices are associated with the isolation function. Thus, this change does not create the possibility of a new accident.

3. The proposed change will result in the reactor building exhaust radiation - high instrumentation being more correctly identified in the Technical Specifications. The instrumentation will be referenced more completely and thereby reduce confusion about which instrumentation is associated with the isolation signal. There is no physical change to the instrumentation; an additional reference is being added to provide a more complete description of the isolation instrumentation. Therefore, this change has no impact on the margin of safety.

Proposed Change Number 145

Replace "and" with ";" in the instrument number listings in Item 2.c, "Reactor Vessel Water Level - Low, Level 2," in Table 4.3.2-1 under Secondary Containment Isolation on Page 3/4 3-29e.

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table, and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability of the primary containment, nor does it change the demands placed upon this process. Therefore, there is no impact on the margin of safety.

Proposed Change Number 146

Replace Instrument Number G31-dFS-N603-1A,1B with Instrument Number G31-FDS-N603-1A,1B in Item 3.a, "Δ Flow - High," in Table 4.3.2-1 under Reactor Water Cleanup System Isolation on Page 3/4 3-29e.

Basis

The isolation instrumentation listed in Item 3.a isolates the reactor water cleanup system supply valves in the event of a rupture of the reactor water cleanup system. This minimizes the amount of radioactivity, energy, and inventory lost from the reactor vessel.

Currently, the Technical Specifications list Instrument Number G31-dFS-N603-1A,1B under Item 3.a. The proposed change will revise that number to G31-FDS-N060-1A,1B. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference G31-dFS-N603-1A,1B with G31-FDS-N603-1A,1B. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device, and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 3.a of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment, nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 147

Replace Instrument Number G31-TS-N602A,B,C,D,E,F with Instrument Number G31-TDS-N602A,B,C,D,E,F in Item 3.c, "Area Ventilation Δ Temperature - High," in Table 4.3.2-1 under Reactor Water Cleanup System Isolation on Page 3/4 3-29e.

Basis

The isolation instrumentation listed in Item 3.c isolates the reactor water cleanup system supply valves in the event of a rupture of the reactor water cleanup system. This minimizes the amount of radioactivity, energy, and inventory lost from the reactor vessel.

Currently, the Technical Specifications list Instrument Number G31-TS-N602A,B,C,D,E,F under Item 3.c. The proposed change will revise that number to G31-TDS-N602A,B,C,D,E,F. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference G31-TS-N602A,B,C,D,E,F with G31-TDS-N602A,B,C,D,E,F. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device, and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new or different type of accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 3.c of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment, nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 148

Replace "and" with ";" in the instrument number listings in Item 3.e, "Reactor Vessel Water Level - Low, Level 2," in Table 4.3.2-1 under Reactor Water Cleanup System Isolation on Page 3/4 3-29e.

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table, and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability of the primary containment, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 149

Add new Item 3.f, "Δ Flow - High - Time Delay Relay," and associated Instrument Number G31-R616C,D to Table 4.3.2-1 under Reactor Water Cleanup System Isolation on Page 3.4 3-29e.

Basis

The reactor water cleanup (RWCU) system isolation instrumentation isolates the RWCU supply isolation valves in the event of a rupture of the RWCU system. This minimizes the amount of radioactivity, energy, and inventory lost from the reactor vessel. A potential rupture is sensed by any of the following signals: high delta supply/return RWCU flow, high RWCU area temperature, high delta supply/return ventilation temperature, or reactor vessel water level - low, level 2. The system also isolates on standby liquid control system initiation to prevent dilution of the poison injected due to cleanup from RWCU operation.

The current Technical Specifications do not specifically reference operability and surveillance requirements for the existing RWCU Δ flow - high time delay relays. The proposed change adds surveillance requirements for these relays. The proposed requirements are similar to those listed for Item 3.a since these relays are part of the Δ flow - high logic and have the same function. Initially, the time delay relays were considered to be a part of the isolation logic described in Item 3.a and were not listed separately in the table. These relays do not initiate any isolation signal; however, they are an important part of the instrumentation. It has been determined that they should be addressed separately in the table to avoid possible confusion.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes specific operability and surveillance requirements for instrumentation which is designed to detect an accident and not prevent one. These requirements are consistent with the accident analysis assumptions and are similar to those for other associated instrumentation. There is no change to the equipment or in how the equipment is operated; only references to existing instrumentation are being added to the table. Since the establishment of operability and surveillance requirements does not change the design or affect the operation of the instrumentation, there is no increase in the probability of an accident, nor is there any change in the consequences of any accident previously evaluated.

2. Specification of the correct operability and surveillance requirements provides additional assurance that the instrumentation will perform its intended safety function. It does not change the design or operation of any equipment or cause it to operate in a manner not previously assumed It does not affect any equipment which could cause an accident. It only affects instrumentation designed to detect the occurrence of an accident. Thus, this change does not create the possibility of a new accident.

3. Specification of specific operability and surveillance requirements that is similar to the other isolation instrumentation for this system will not affect or change the operation of the affected equipment. There will be no change in the reliability, response characteristics, or setpoints of the instrumentation. Therefore, there is no change in the margin of safety.

Proposed Change Number 150

Replace "HPCI Steam Line High Flow Time Delay Relay" with "HPCI Steam Line Flow - High Time Delay Relay" in the title of Item 4.a.2 in Table 4.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-29f.

Basis

The proposed change was made solely to provide consistency with other titles throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The instrumentation itself has not changed, nor has its function, design, or operation. Only its title has been reworded to be consistent with other titles in the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table. It does not involve a change in instrumentation, function, design, or operation. It does not change the design or capability of the primary containment, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 151

Replace Instrument Number E41-TS-3388 with Instrument Number E41-TS-3488 in Item 4.a.4, "HPCI Steam Line Tunnel Temperature - High," in Table 4.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-29f (**BSEP-2 only**).

Basis

This change is an administrative change to correct a typographical error. No instrumentation is being physically changed; only the existing instrument number is being revised to reflect the correct instrument number.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide the correct instrument number. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide a a corrected reference to the proper instrumentation associated with the HPCI steam line tunnel temperature - high instrumentation. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide the correct reference to the instrumentation associated with the HPCI steam line tunnel temperature - high function. No design, operation, surveillance, or testing requirements are being changed. It does not change the design or capability of the primary containment, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 152

Replace "and" with ";" in the instrument number listings in Item 4.a.5, "Bus Power Monitor," in Table 4.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-29f.

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table, and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability of the primary containment, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 153

Replace Instrument Number E51-dTS-N604C,D with Instrument Number E51-TDS-N604C,D in Item 4.a.8, "HPCI Steam Line Area Δ Temperature - High," in Table 4.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-29g.

Basis

The isolation instrumentation listed in Item 4.a.8 isolates the high pressure coolant injection (HPCI) system should there be a HPCI steam line break.

Currently, the Technical Specifications list Instrument Number E51-dTS-N604C,D under Item 4.a.8. The proposed change will revise that number to E51-TDS-N604C,D. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference E51-dTS-N604C,D with E51-TDS-N604C,D. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 4.a.8 of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment, nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 154

Replace "Emergency Area Cooler Temp - High" with "HPCI Equipment Area Temperature - High" in the title of Item 4.a.9 in Table 4.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-29g.

Basis

The emergency area cooler temperature - high isolation signal isolates the Group 4 isolation valves in the event of a high pressure coolant injection (HPCI) steam line break to mitigate the consequences of such a break. The Group 4 isolation valves include the HPCI inboard and outboard steam line isolation valves and the HPCI torus suction isolation valves.

The current Technical Specifications have an inaccurate description of the trip function. The proposed change revises the description of the trip function from "Emergency Area Cooler Temperature - High" to "HPCI Equipment Area Temperature - High."

This change is necessary so that the trip function description more clearly reflects the actual area the instrumentation monitors. The existing description is a generic BWR term which does not relate to the actual BSEP design. The area the instrumentation monitors at BSEP is referred to in other plant documents and programs as the "HPCI Equipment Area." This change will reduce the potential for confusion by more clearly describing the actual location of the instrumentation.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will change the description of a trip function to more clearly identify the area the instrumentation monitors. The current wording, "Emergency Area Cooler," does not reflect the commonly used description of the area. "HPCI Equipment Area" more clearly describes the area. This change will reduce confusion concerning which area is being referred to.

   No changes to the referenced instrumentation are being reflected by this change. The instrumentation will function exactly as before. Therefore, there is no change in the probability of an accident, nor is there any change in the consequences of an accident.

2. The proposed change revises the trip function description so the appropriate trip function is clearly identified in the Technical Specifications. It does not change the required function of the instrumentation; it merely clarifies the description of an existing trip function. Thus, no new accident possibilities are created.

3. The proposed change provides a clarification in the description of an existing isolation function. It does not change the design, reliability, or function of any plant equipment; it merely references them more clearly. Thus, there is no impact on the margin of safety as a result of this change.

Proposed Change Number 155

Add new Item 4.a.10, "Drywell Pressure - High," and associated Instrument Numbers E11-PT-N011C,D and E11-PTS-N011C-2,D-2 to Table 4.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-29g.

Basis

Valve Group 4 isolates the high pressure coolant injection (HPCI) inboard and outboard steam line isolation valves and the HPCI torus suction isolation valves in the event of a HPCI steam line break to mitigate the consequences of a break.

The HPCI system also has turbine exhaust vacuum breaker isolation valves (E41-F075 and E41-F079) on a vacuum relief line for the HPCI turbine exhaust. This line helps prevent the creation of a water column in the exhaust line. Preventing development of this column reduces the piping loads which could exist if the turbine is restarted. The valves isolate on coincident HPCI steam line pressure - low and drywell pressure - high signals to establish primary containment. The steam line pressure - low signal indicates HPCI is isolated or no longer needed because the reactor is depressurized. The drywell pressure - high signal indicates a loss of coolant accident (LOCA) exists and primary containment isolation is needed.

The current Technical Specifications specify operability and surveillance requirements for the Group 4 isolation valves but not for the vacuum breaker isolation valves. The proposed change reflects the creation of a new Valve Group 7, which includes the HPCI turbine exhaust vacuum breakers E41-F075 and E41-F079. Surveillance requirements for the actuation instrumentation are addressed under Item 4.a.3 for the HPCI steam supply pressure - low instrumentation portion of the logic. A new Item 4.a.10 is being added for the drywell pressure - high portion of the logic. The specified requirements are the same as for similar instrumentation for this application and are sufficient to ensure reliability.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes the surveillance requirements for the existing drywell pressure - high instrumentation which was not originally included in the Technical Specifications. The HPCI turbine exhaust vacuum breakers isolate on coincident HPCI steam line pressure - low and drywell pressure - high. The steam line pressure - low signal indicates HPCI is isolated or no longer needed because the reactor is depressurized, and the drywell pressure - high signal indicates a LOCA exists and primary containment isolation is needed.

This change adds requirements for existing instrumentation to provide additional assurance that primary containment is isolated when necessary. No new equipment is being added, and existing equipment will perform its safety function in the same manner as before. These requirements are being added to the Technical Specifications for clarity and completeness. Thus, there is no change in the probability of an accident; nor is there any change in the consequences of any accident.

2. The proposed change adds surveillance requirements to the Technical Specifications for existing instrumentation. This will provide a more complete and accurate document, thereby enhancing primary containment isolation assurance. It does not provide an opportunity for occurrence of any new accidents because no equipment, operability requirements, or setpoints are being altered.

3. The proposed change adds requirements for existing instrumentation to the Technical Specifications. It does not reflect a physical change to the plant systems; it only provides a more complete and accurate description of existing equipment. It does not change the design or capability of the primary containment, nor does it change the demands placed upon this process barrier. Thus, there is no impact on the margin of safety of the plant.

Proposed Change Number 156

Replace "RCIC Steam Line High Flow Time Delay Relay" with "RCIC Steam Line Flow - High Time Delay Relay" in the title of Item 4.b.2 in Table 4.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-29g.

Basis

The proposed change was made solely to provide consistency with other titles throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The instrumentation itself has not changed, nor has its function, design, or operation. Only its title has been reworded to be consistent with other titles in the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table. It does not involve a change in instrumentation, function, design, or operation. It does not change the design or capability of the primary containment, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 157

Replace "and" with ";" in the instrument number listings in Item 4.b.5, "Bus Power Monitor," in Table 4.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-29h.

Basis

The proposed changes were made solely to provide consistency with other notation throughout the section and to avoid confusion in interpretation of the table. The conjunction "and" was not intended to describe logic functions of the instruments. The word "and" was used in the tag number lists purely as a conjunction. The information provided is merely a list of instrument tag numbers associated with the described instrumentation and is provided solely to assist the operator in usage of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The logic associated with the instrumentation has not changed, nor has its function, design, or operation. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in logic, nor does it reflect a change in the instrument function, design, or operation. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency within the table, and eliminate any confusion there may be as to whether the logic associated with the instrumentation is represented. Logic is not represented and was never intended to be represented. It does not change the design or capability of the primary containment, nor does it change the demands placed upon this process barrier. Therefore, there is no impact on the margin of safety.

Proposed Change Number 158

Replace Instrument Number E51-dTS-N604A,B with Instrument Number E51-TDS-N604A,B in Item 4.b.8, "RCIC Steam Line Area Δ Temperature - High," in Table 4.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-29h.

Basis

The isolation instrumentation listed in Item 4.b.8 isolates the reactor core isolation cooling (RCIC) system should there be a RCIC steam line break.

Currently, the Technical Specifications list Instrument Number E51-dTS-N604A,B under Item 4.b.8. The proposed change will revise that number to E51-TDS-N604A,B. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference E51-dTS-N604A,B with E51-TDS-N604A,B. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 4.b.8 of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment, nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 159

Replace Instrument Number E51-dTS-N601A,B with Instrument Number E51-TDS-N601A,B in Item 4.b.10, "RCIC Equipment Room Δ Temperature - High," in Table 4.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-29h.

Basis

The isolation instrumentation listed in Item 4.b.10 isolates the reactor core isolation cooling (RCIC) system should there be a RCIC steam line break.

Currently, the Technical Specifications list Instrument Number E51-dTS-N601A,B under Item 4.b.10. The proposed change will revise that number to E51-TDS-N601A,B. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference E51-dTS-N601A,B with E51-TDS-N601A,B. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 4.b.10 of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment, nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 160

Add new Item 4.b.11, "RCIC Steam Line Tunnel Temperature - High - Time Delay Relay," and associated Instrument Number E51-KC-M602A,B to Table 4.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-29h.

Basis

The reactor core isolation cooling (RCIC) steam line tunnel temperature - high, time delay relay instrumentation is part of the instrumentation that isolates the Group 5 isolation valves in the event of a RCIC steam line break to mitigate the consequences of the break. The Group 5 valves include the RCIC inboard and outboard steam line isolation valves.

The current Technical Specifications do not specifically reference operability and surveillance requirements for the existing RCIC steam line tunnel temperature - high time delay relays. The proposed change adds surveillance requirements for these relays. The proposed requirements are similar to those listed for Item 4.b.7 since the relays are part of that logic and have the same function. Initially, the time delay relays were considered to be a part of the isolation logic described in Item 4.b.7 and were not listed separately in the table. These relays do not initiate any isolation signal; however, they are an important part of the instrumentation, and it has been determined that they should be addressed separately in the table to avoid possible confusion.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes specific surveillance requirements for instrumentation which is designed to detect an accident and not prevent one. These requirements are consistent with the accident analysis assumptions and are similar to those for other associated instrumentation. There is no change to the equipment or in how the equipment is operated; only references to existing instrumentation are being added to the table. Since the establishment of operability and surveillance requirements does not change the design or affect the operation of the instrumentation, there is no increase in the probability of an accident, nor is there any change in the consequences of any accident previously evaluated.

2. Specification of the correct operability and surveillance requirements provides additional assurance that the instrumentation will perform its intended safety function. It does not change the design or operation of any equipment or cause it to operate in a manner not previously assumed. It does not affect any equipment which could cause an accident. It only affects instrumentation designed to detect the occurrence of an accident. Thus, this change does not create the possibility of a new accident.

3. Specification of specific operability and surveillance requirements that are similar to the other isolation instrumentation for this system will not affect or change the operation of the affected equipment. There will be no change in the reliability, response characteristics, or setpoints of the instrumentation. It does not change the design or capability of the primary containment, nor does it change the demands placed upon this process barrier. Therefore, there is no change in the margin of safety.

Proposed Change Number 161

Add new Item 4.b.12, "Drywell Pressure - High," and associated Instrument Numbers E11-PT-N011A,B and E11-PTS-N011A-2,B-2 to Table 4.3.2-1 under Core Standby Cooling Systems Isolation on Page 3/4 3-29i.

Basis

Valve Group 5 isolates the reactor core isolation cooling (RCIC) steam line to mitigate the consequences of a break. The Group 5 isolation valves include the RCIC inboard and outboard steam line isolation valves.

The RCIC system also has turbine exhaust vacuum breaker isolation valves (E51-F062 and E51-F066) on a vacuum relief line for the RCIC turbine exhaust. The line helps prevent the development of a water column in the exhaust line. Preventing development of this column reduces the piping loads which could exist if the turbine is restarted. The valves isolate on coincident RCIC steam line pressure - low and drywell pressure - high signals to establish primary containment isolation. The steam line pressure - low signal indicates RCIC is isolated or no longer needed because the reactor is depressurized. The drywell pressure - high signal indicates a loss of coolant accident (LOCA) exists and primary containment isolation is needed.

The current Technical Specifications specify operability and maintenance requirements for the Group 5 isolation valves but not for the vacuum breaker isolation valves. The proposed change reflects the creation of the new Valve Group 9, which includes the RCIC turbine exhaust vacuum breakers E51-F062 and E51-F066.

The isolation logic associated with these vacuum breakers requires a signal from both the drywell pressure - high instrumentation and the RCIC steam supply pressure - low instrumentation. Item 4.b.3 addresses surveillance requirements for the RCIC steam supply pressure - low portion. The proposed change addressed here adds the surveillance requirement information associated with the drywell pressure - high instrumentation. The specified requirements are the same as similar instrumentation for this application and are sufficient to ensure reliability.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change establishes the surveillance requirement information for the drywell pressure - high instrumentation which was not originally included in the Technical Specifications. This change is being made to address the addition of the RCIC turbine exhaust vacuum breakers, which isolate on coincident RCIC steam line pressure - low and drywell pressure - high, to the Technical Specifications via addition of Valve Group 9. The steam line pressure - low signal indicates RCIC is isolated or no longer needed because the reactor is depressurized, and the drywell pressure - high signal indicates a LOCA exists and primary containment isolation is needed.

This change adds requirements for existing instrumentation to provide additional assurance that primary containment is isolated when necessary. No new equipment is being added, and existing equipment will perform its safety function in the same manner as before. These requirements are being added to the Technical Specifications for clarity and completeness. Thus, there is no change in the probability of an accident; nor is there any change in the consequences of any accident.

2. The proposed change adds operability requirements to the Technical Specifications for existing instrumentation. This will provide a more complete and accurate document, thereby enhancing primary containment isolation assurance. It does not provide an opportunity for any new accidents because no equipment, operability requirements, or setpoints are being altered.

3. The proposed change adds requirements for existing instrumentation to the Technical Specifications. It does not reflect a physical change to the plant systems; it only provides a more complete and accurate description of existing equipment. It does not change the design or capability of the primary containment, nor does it change the demands placed upon this process barrier. Thus, there is no impact on the margin of safety of the plant.

Proposed Change Number 162

Replace Instrument Number B32-PS-N018A,B with Instrument Number B32-PS-N018A-1,B in Item 5.b, "Reactor Steam Dome Pressure - High," in Table 4.3.2-1 under Shutdown Cooling System Isolation on Page 3/4 3-29i.

Basis

The isolation instrumentation listed in Item 5.b isolates the shutdown cooling system should there be a malfunction or rupture of the residual heat removal system while operating in shutdown cooling mode. This minimizes reactor inventory loss.

Currently, the Technical Specifications list Instrument Number B32-PS-N018A,B under Item 5.b. The proposed change will revise that number to B32-PS-N018A-1,B. The existing instrument number is not formatted consistently with that listed in other plant documents.

This change does not reflect a change to the instrumentation itself; it only provides a correct reference to existing information that is consistent with other plant documents.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change will replace instrument number reference B32-PS-N018A,B with B32-PS-N018A-1,B. It does not change the device which performs the specified function; it more clearly identifies the existing device. This may reduce confusion about which device is being referenced and provide consistency with other plant documents. Therefore, there is no impact on the consequences of an accident, nor is there any change in the probability of an accident.

2. The proposed change does not reflect any change in plant design. It more clearly identifies an existing device and make its nomenclature consistent with that used in other plant documents. Therefore, it does not create the possibility of a new accident.

3. The proposed change will more clearly identify an instrument that currently exists under Item 5.b of the table. The change does not represent any changes to the instrument, its function, or design. Only the instrument tag number is being revised to provide consistency in nomenclature with other plant documents. By making the nomenclature consistent, confusion related to identification of the instrument may be alleviated. It does not change the design or capability of the primary containment, nor does it change the demands placed upon this process barrier. Therefore, this change does not decrease the margin of safety.

Proposed Change Number 163

Replace * with (e) and move it to the end of the footnote table on Page 3/4 3-29j.

Basis

This change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnote has not changed unless noticed elsewhere in this enclosure. The footnote has been changed from * to (e) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnotes. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Proposed Change Number 164

Replace # with (f) and move it to the end of the footnote table on Page 3/4 3-29j.

Basis

This change was made solely to provide consistency throughout the table.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the footnote has not changed unless noticed elsewhere in this enclosure. The footnote has been changed from # to (f) to provide clarity and consistency to the table. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided in the table. It does not represent a change in the content of the footnotes. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Proposed Change Number 165

Delete the reference to Table 3.6.3-1 in Specification 3.6.1.1 on Page 3/4 6-1.

Basis

Technical Specification Section 3.6 addresses containment systems. Table 3.6.3-1 provides a partial listing of the automatic primary containment isolation valves (PCIVs). The Brunswick Plant Technical Specifications currently require the following surveillances:

Specification 4.6.3.1 requires that primary containment isolation valves be demonstrated operable by cycling and isolation time testing following maintenance, repair, or replacement work.

Specification 4.6.3.2 requires verification of isolation valve operability through the application of an isolation test signal to verify the valve actuates properly. This testing is required every 18 months.

Specification 4.6.3.3 requires verification of the isolation time for power-operated and automatic valves.

Specification 4.6.4.4 requires that reactor instrumentation system isolation valves be demonstrated operable by cycling the valve every 18 months.

Specification 4.6.1.2.d requires Type B and C leak testing of primary containment isolation valves.

The Company has completed a review of the primary containment isolation system (PCIS) Technical Specifications and determined that certain changes to Table 3.6.3-1 are needed to accurately reflect the as-built design of the primary containment isolation system. However, rather than propose changes to the table, CP&L believes it is more appropriate to request removal of Table 3.6.3-1 from the Technical Specifications and replace references to the table with references to valve listings contained in plant procedures (the Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal). Our reasons for this determination are described below.

First, removal of the valve listings from the Technical Specifications is appropriate in the context of Technical Specification reform. The recent industry and NRC efforts to improve the Technical Specifications have identified component listings as items for removal from the Technical Specifications because they do not directly contribute to plant safety. In addition, the level of detail of such listings has resulted in frequent requests for license amendments. This process has resulted in inherent inaccuracies in the Technical Specifications due to the delays in receiving license amendments brought about by the current regulatory environment. The NRC has already recommended and approved removal of the analogous snubber listing from the Technical Specifications. In addition, tables relating to fire protection devices and containment penetration protection circuit breakers have been removed from the Technical Specifications for other dockets. For example, listings relating to containment isolation valves, snubbers, containment penetration protection devices, motor operated valve thermal overload protection, the reactor vessel material surveillance program, engineered safety features actuation system instrumentation response times, and reactor trip system

instrumentation response times have been removed from the Technical Specifications for the Shearon Harris Plant and references to plant procedures incorporated. The use of component listings through reference to plant procedures ensures that timely information is maintained that reflects recent plant modifications.

Second, changes to these plant procedures are strictly controlled under 10CFR50.59. In addition, the Technical Specifications requirement to maintain primary containment integrity (Technical Specification 3.6.1.1), as well as ASME Code, Section XI testing requirements (Technical Specification 4.0.5) and 10CFR50, Appendix J requirements (Technical Specification 3.6.1.2) are maintained. Only the listing of valves would be subject to licensee control under 10CFR50.59. Plant procedures are auditable and subject to inspection by the NRC.

Third, appropriate mechanisms and controls exist to ensure correct testing for the valves through plant procedures. As discussed above, these referenced procedure(s) will incorporate valve list information developed to meet the requirements of Technical Specifications 4.6.3.1, 4.6.3.2, 4.6.3.3 and 4.6.3.4. In addition, Appendix J of 10CFR50 (implemented through Technical Specification 3/4.6.1.2) specifies leak rate test requirements for containment isolation valves, and thus maintenance of a list of such valves. Section XI of the ASME Code (implemented through Technical Specification 4.0.5) requires once per cycle testing of valve closure and timing for certain of the valves currently listed in Table 3.6.3-1. In addition, Technical Specification 4.3.2.2 requires once per cycle performance of a logic system functional test for most of the valves currently listed in Table 3.6.3-1. All the above requirements are appropriately integrated into plant procedure(s) which are subject to change only under requirements of 10CFR50.59. These plant procedures cover testing (to or beyond the present Technical Specification requirements) for the valves currently included in Table 3.6.3-1. A complete listing of containment isolation valves and required closure times is also provided as Table 6.2.4-2 to the Brunswick Updated Final Safety Analysis Report. References to plant procedures will replace references to Table 3.6.3-1 in the text of the Technical Specifications. Specific closure time testing requirements for the main steam isolation valves will be maintained in the Technical Specifications under Specification 3/4.4.7.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. Removal of the primary containment isolation valve listing from the Technical Specifications is appropriate in the context of Technical Specification reform. Recent industry and NRC efforts to improve the Technical Specifications have identified component listings as an item for removal because they do not directly contribute to plant safety. The use of component listings through reference to plant procedures will ensure that timely information is maintained that reflects recent plant modifications. Changes to these procedures will be controlled under the requirements of 10CFR50.59. Primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specifications 3.6.1.1 and 3.6.3 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3 (and thus the proper identification, control, and periodic testing of the appropriate containment

isolation valves). Relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not change their design or operation, nor does it affect any of the accident analyses; therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. As discussed in Item 1 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3 (and thus the proper identification, control, and periodic testing of the appropriate containment isolation valves). Thus, relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not change their design or operation so the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. As discussed in Items 1 and 2 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3. Furthermore, the relocation of the primary containment isolation valve listing to plant procedures will clarify the Technical Specifications and ensure a more up to date and accurate primary containment isolation valve listing, thereby avoiding possible operator confusion. Since there is no change in containment design or pressure retaining capability, the proposed change does not involve a significant reduction in the margin of safety.

Proposed Change Number 166

Delete references to Table 3.6.3-1 in Technical Specification 3.6.1.2 in Item b under Limiting Condition for Operation, in Item b under "with" in the Action statement, and in Item b under "restore" in the Action statement on Page 3/4 6-2.

Basis

Technical Specification Section 3.6 addresses containment systems. Table 3.6.3-1 provides a partial listing of the automatic primary containment isolation valves (PCIVs). The Brunswick Plant Technical Specifications currently require the following surveillances:

> Specification 4.6.3.1 requires that primary containment isolation valves be demonstrated operable by cycling and isolation time testing following maintenance, repair, or replacement work.
>
> Specification 4.6.3.2 requires verification of isolation valve operability through the application of an isolation test signal to verify the valve actuates properly. This testing is required every 18 months.
>
> Specification 4.6.3.3 requires verification of the isolation time for power-operated and automatic valves.
>
> Specification 4.6.4.4 requires that reactor instrumentation system isolation valves be demonstrated operable by cycling the valve every 18 months.
>
> Specification 4.6.1.2.d requires Type B and C leak testing of primary containment isolation valves.

The Company has completed a review of the primary containment isolation system (PCIS) Technical Specifications and determined that certain changes to Table 3.6.3-1 are needed to accurately reflect the as-built design of the primary containment isolation system. However, rather than propose changes to the table, CP&L believes it is more appropriate to request removal of Table 3.6.3-1 from the Technical Specifications and replace references to the table with references to valve listings contained in plant procedures (the Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal). Our reasons for this determination are described below.

First, removal of the valve listings from the Technical Specifications is appropriate in the context of Technical Specification reform. The recent industry and NRC efforts to improve the Technical Specifications have identified component listings as items for removal from the Technical Specifications because they do not directly contribute to plant safety. In addition, the level of detail of such listings has resulted in frequent requests for license amendments. This process has resulted in inherent inaccuracies in the Technical Specifications due to the delays in receiving license amendments brought about by the current regulatory environment. The NRC has already recommended and approved removal of the analogous snubber listing from the Technical Specifications. In addition, tables relating to fire protection devices and containment penetration protection circuit breakers have been removed from the Technical Specifications for other dockets. For example, listings relating to containment isolation valves, snubbers, containment penetration protection devices, motor operated valve thermal overload

protection, the reactor vessel material surveillance program, engineered safety features actuation system instrumentation response times, and reactor trip system instrumentation response times have been removed from the Technical Specifications for the Shearon Harris Plant and references to plant procedures incorporated. The use of component listings through reference to plant procedures ensures that timely information is maintained that reflects recent plant modifications.

Second, changes to these plant procedures are strictly controlled under 10CFR50.59. In addition, the Technical Specifications requirement to maintain primary containment integrity (Technical Specification 3.6.1.1), as well as ASME Code, Section XI testing requirements (Technical Specification 4.0.5) and 10CFR50, Appendix J requirements (Technical Specification 3.6.1.2) are maintained. Only the listing of valves would be subject to licensee control under 10CFR50.59. Plant procedures are auditable and subject to inspection by the NRC.

Third, appropriate mechanisms and controls exist to ensure correct testing for the valves through plant procedures. As discussed above, these referenced procedure(s) will incorporate valve list information developed to meet the requirements of Technical Specifications 4.6.3.1, 4.6.3.2, 4.6.3.3 and 4.6.3.4. In addition, Appendix J of 10CFR50 (implemented through Technical Specification 3/4.6.1.2) specifies leak rate test requirements for containment isolation valves, and thus maintenance of a list of such valves. Section XI of the ASME Code (implemented through Technical Specification 4.0.5) requires once per cycle testing of valve closure and timing for certain of the valves currently listed in Table 3.6.3-1. In addition, Technical Specification 4.3.2.2 requires once per cycle performance of a logic system functional test for most of the valves currently listed in Table 3.6.3-1. All the above requirements are appropriately integrated into plant procedure(s) which are subject to change only under requirements of 10CFR50.59. These plant procedures cover testing (to or beyond the present Technical Specification requirements) for the valves currently included in Table 3.6.3-1. A complete listing of containment isolation valves and required closure times is also provided as Table 6.2.4-2 to the Brunswick Updated Final Safety Analysis Report. References to plant procedures will replace references to Table 3.6.3-1 in the text of the Technical Specifications. Specific closure time testing requirements for the main steam isolation valves will be maintained in the Technical Specifications under Specification 3/4.4.7.

<u>10CFR50.92 Evaluation</u>

The proposed change does not involve a significant hazards consideration for the following reasons:

1. Removal of the primary containment isolation valve listing from the Technical Specifications is appropriate in the context of Technical Specification reform. Recent industry and NRC efforts to improve the Technical Specifications have identified component listings as an item for removal because they do not directly contribute to plant safety. The use of component listings through reference to plant procedures will ensure that timely information is maintained that reflects recent plant modifications. Changes to these procedures will be controlled under the requirements of 10CFR50.59. Primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specifications 3.6.1.1 and 3.6.3 will continue to require primary containment integrity while

operating in Operational Conditions 1, 2, and 3 (and thus the proper identification, control, and periodic testing of the appropriate containment isolation valves). Relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not change their design or operation, nor does it affect any of the accident analyses; therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. As discussed in Item 1 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3 (and thus the proper identification, control, and periodic testing of the appropriate containment isolation valves). Thus, relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not change their design or operation so the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. As discussed in Items 1 and 2 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3. Furthermore, the relocation of the primary containment isolation valve listing to plant procedures will clarify the Technical Specifications and ensure a more up to date and accurate primary containment isolation valve listing, thereby avoiding possible operator confusion. Since there is no change in containment design or pressure retaining capability, the proposed change does not involve a significant reduction in the margin of safety.

Proposed Change Number 167

Delete the reference to Table 3.6.3-1 in Specification 3.6.3 on Page 3/4 6-12 and reference plant procedure ___________.

Basis

Technical Specification Section 3.6 addresses containment systems. Table 3.6.3-1 provides a partial listing of the automatic primary containment isolation valves (PCIVs). The Brunswick Plant Technical Specifications currently require the following surveillances:

> Specification 4.6.3.1 requires that primary containment isolation valves be demonstrated operable by cycling and isolation time testing following maintenance, repair, or replacement work.
>
> Specification 4.6.3.2 requires verification of isolation valve operability through the application of an isolation test signal to verify the valve actuates properly. This testing is required every 18 months.
>
> Specification 4.6.3.3 requires verification of the isolation time for power-operated and automatic valves.
>
> Specification 4.6.4.4 requires that reactor instrumentation system isolation valves be demonstrated operable by cycling the valve every 18 months.
>
> Specification 4.6.1.2.d requires Type B and C leak testing of primary containment isolation valves.

The Company has completed a review of the primary containment isolation system (PCIS) Technical Specifications and determined that certain changes to Table 3.6.3-1 are needed to accurately reflect the as-built design of the primary containment isolation system. However, rather than propose changes to the table, CP&L believes it is more appropriate to request removal of Table 3.6.3-1 from the Technical Specifications and replace references to the table with references to valve listings contained in plant procedures (the Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal). Our reasons for this determination are described below.

First, removal of the valve listings from the Technical Specifications is appropriate in the context of Technical Specification reform. The recent industry and NRC efforts to improve the Technical Specifications have identified component listings as items for removal from the Technical Specifications because they do not directly contribute to plant safety. In addition, the level of detail of such listings has resulted in frequent requests for license amendments. This process has resulted in inherent inaccuracies in the Technical Specifications due to the delays in receiving license amendments brought about by the current regulatory environment. The NRC has already recommended and approved removal of the analogous snubber listing from the Technical Specifications. In addition, tables relating to fire protection devices and containment penetration protection circuit breakers have been removed from the Technical Specifications for other dockets. For example, listings relating to containment isolation valves, snubbers, containment penetration protection devices, motor operated valve thermal overload protection, the reactor vessel material surveillance program, engineered safety

features actuation system instrumentation response times, and reactor trip system instrumentation response times have been removed from the Technical Specifications for the Shearon Harris Plant and references to plant procedures incorporated. The use of component listings through reference to plant procedures ensures that timely information is maintained that reflects recent plant modifications.

Second, changes to these plant procedures are strictly controlled under 10CFR50.59. In addition, the Technical Specifications requirement to maintain primary containment integrity (Technical Specification 3.6.1.1), as well as ASME Code, Section XI testing requirements (Technical Specification 4.0.5) and 10CFR50, Appendix J requirements (Technical Specification 3.6.1.2) are maintained. Only the listing of valves would be subject to licensee control under 10CFR50.59. Plant procedures are auditable and subject to inspection by the NRC.

Third, appropriate mechanisms and controls exist to ensure correct testing for the valves through plant procedures. As discussed above, these referenced procedure(s) will incorporate valve list information developed to meet the requirements of Technical Specifications 4.6.3.1, 4.6.3.2, 4.6.3.3 and 4.6.3.4. In addition, Appendix J of 10CFR50 (implemented through Technical Specification 3/4.6.1.2) specifies leak rate test requirements for containment isolation valves, and thus maintenance of a list of such valves. Section XI of the ASME Code (implemented through Technical Specification 4.0.5) requires once per cycle testing of valve closure and timing for certain of the valves currently listed in Table 3.6.3-1. In addition, Technical Specification 4.3.2.2 requires once per cycle performance of a logic system functional test for most of the valves currently listed in Table 3.6.3-1. All the above requirements are appropriately integrated into plant procedure(s) which are subject to change only under requirements of 10CFR50.59. These plant procedures cover testing (to or beyond the present Technical Specification requirements) for the valves currently included in Table 3.6.3-1. A complete listing of containment isolation valves and required closure times is also provided as Table 6.2.4-2 to the Brunswick Updated Final Safety Analysis Report. References to plant procedures will replace references to Table 3.6.3-1 in the text of the Technical Specifications. Specific closure time testing requirements for the main steam isolation valves will be maintained in the Technical Specifications under Specification 3/4.4.7.

<u>10CFR50.92 Evaluation</u>

The proposed change does not involve a significant hazards consideration for the following reasons:

1. Removal of the primary containment isolation valve listing from the Technical Specifications is appropriate in the context of Technical Specification reform. Recent industry and NRC efforts to improve the Technical Specifications have identified component listings as an item for removal because they do not directly contribute to plant safety. The use of component listings through reference to plant procedures will ensure that timely information is maintained that reflects recent plant modifications. Changes to these procedures will be controlled under the requirements of 10CFR50.59. Primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specifications 3.6.1.1 and 3.6.3 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3 (and thus the proper

identification, control, and periodic testing of the appropriate containment isolation valves). Relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not change their design or operation, nor does it affect any of the accident analyses; therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. As discussed in Item 1 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3 (and thus the proper identification, control, and periodic testing of the appropriate containment isolation valves). Thus, relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not change their design or operation so the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. As discussed in Items 1 and 2 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3. Furthermore, the relocation of the primary containment isolation valve listing to plant procedures will clarify the Technical Specifications and ensure a more up to date and accurate primary containment isolation valve listing, thereby avoiding possible operator confusion. Since there is no change in containment design or pressure retaining capability, the proposed change does not involve a significant reduction in the margin of safety.

Proposed Change Number 168

Replace the reference to Table 3.6.3-1 with a reference to plant procedure ______ in Action a of Technical Specification 3.6.3 on Page 3/4 6-12.

Basis

Technical Specification Section 3.6 addresses containment systems. Table 3.6.3-1 provides a partial listing of the automatic primary containment isolation valves (PCIVs). The Brunswick Plant Technical Specifications currently require the following surveillances:

> Specification 4.6.3.1 requires that primary containment isolation valves be demonstrated operable by cycling and isolation time testing following maintenance, repair, or replacement work.
>
> Specification 4.6.3.2 requires verification of isolation valve operability through the application of an isolation test signal to verify the valve actuates properly. This testing is required every 18 months.
>
> Specification 4.6.3.3 requires verification of the isolation time for power-operated and automatic valves.
>
> Specification 4.6.4.4 requires that reactor instrumentation system isolation valves be demonstrated operable by cycling the valve every 18 months.
>
> Specification 4.6.1.2.d requires Type B and C leak testing of primary containment isolation valves.

The Company has completed a review of the primary containment isolation system (PCIS) Technical Specifications and determined that certain changes to Table 3.6.3-1 are needed to accurately reflect the as-built design of the primary containment isolation system. However, rather than propose changes to the table, CP&L believes it is more appropriate to request removal of Table 3.6.3-1 from the Technical Specifications and replace references to the table with references to valve listings contained in plant procedures (the Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal). Our reasons for this determination are described below.

First, removal of the valve listings from the Technical Specifications is appropriate in the context of Technical Specification reform. The recent industry and NRC efforts to improve the Technical Specifications have identified component listings as items for removal from the Technical Specifications because they do not directly contribute to plant safety. In addition, the level of detail of such listings has resulted in frequent requests for license amendments. This process has resulted in inherent inaccuracies in the Technical Specifications due to the delays in receiving license amendments brought about by the current regulatory environment. The NRC has already recommended and approved removal of the analogous snubber listing from the Technical Specifications. In addition, tables relating to fire protection devices and containment penetration protection circuit breakers have been removed from the Technical Specifications for other dockets. For example, listings relating to containment isolation valves, snubbers, containment penetration protection devices, motor operated valve thermal overload protection, the reactor vessel material surveillance program, engineered safety

features actuation system instrumentation response times, and reactor trip system instrumentation response times have been removed from the Technical Specifications for the Shearon Harris Plant and references to plant procedures incorporated. The use of component listings through reference to plant procedures ensures that timely information is maintained that reflects recent plant modifications.

Second, changes to these plant procedures are strictly controlled under 10CFR50.59. In addition, the Technical Specifications requirement to maintain primary containment integrity (Technical Specification 3.6.1.1), as well as ASME Code, Section XI testing requirements (Technical Specification 4.0.5) and 10CFR50, Appendix J requirements (Technical Specification 3.6.1.2) are maintained. Only the listing of valves would be subject to licensee control under 10CFR50.59. Plant procedures are auditable and subject to inspection by the NRC.

Third, appropriate mechanisms and controls exist to ensure correct testing for the valves through plant procedures. As discussed above, these referenced procedure(s) will incorporate valve list information developed to meet the requirements of Technical Specifications 4.6.3.1, 4.6.3.2, 4.6.3.3 and 4.6.3.4. In addition, Appendix J of 10CFR50 (implemented through Technical Specification 3/4.6.1.2) specifies leak rate test requirements for containment isolation valves, and thus maintenance of a list of such valves. Section XI of the ASME Code (implemented through Technical Specification 4.0.5) requires once per cycle testing of valve closure and timing for certain of the valves currently listed in Table 3.6.3-1. In addition, Technical Specification 4.3.2.2 requires once per cycle performance of a logic system functional test for most of the valves currently listed in Table 3.6.3-1. All the above requirements are appropriately integrated into plant procedure(s) which are subject to change only under requirements of 10CFR50.59. These plant procedures cover testing (to or beyond the present Technical Specification requirements) for the valves currently included in Table 3.6.3-1. A complete listing of containment isolation valves and required closure times is also provided as Table 6.2.4-2 to the Brunswick Updated Final Safety Analysis Report. References to plant procedures will replace references to Table 3.6.3-1 in the text of the Technical Specifications. Specific closure time testing requirements for the main steam isolation valves will be maintained in the Technical Specifications under Specification 3/4.4.7.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. Removal of the primary containment isolation valve listing from the Technical Specifications is appropriate in the context of Technical Specification reform. Recent industry and NRC efforts to improve the Technical Specifications have identified component listings as an item for removal because they do not directly contribute to plant safety. The use of component listings through reference to plant procedures will ensure that timely information is maintained that reflects recent plant modifications. Changes to these procedures will be controlled under the requirements of 10CFR50.59. Primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specifications 3.6.1.1 and 3.6.3 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3 (and thus the proper

identification, control, and periodic testing of the appropriate containment isolation valves). Relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not change their design or operation, nor does it affect any of the accident analyses; therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. As discussed in Item 1 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3 (and thus the proper identification, control, and periodic testing of the appropriate containment isolation valves). Thus, relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not change their design or operation so the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. As discussed in Items 1 and 2 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3. Furthermore, the relocation of the primary containment isolation valve listing to plant procedures will clarify the Technical Specifications and ensure a more up to date and accurate primary containment isolation valve listing, thereby avoiding possible operator confusion. Since there is no change in containment design or pressure retaining capability, the proposed change does not involve a significant reduction in the margin of safety.

Proposed Change Number 169

Add a reference to plant procedure ________ in Action b of Technical Specification 3.6.3 on Page 3/4 6-12.

Basis

Technical Specification Section 3.6 addresses containment systems. Table 3.6.3-1 provides a partial listing of the automatic primary containment isolation valves (PCIVs). The Brunswick Plant Technical Specifications currently require the following surveillances:

> Specification 4.6.3.1 requires that primary containment isolation valves be demonstrated operable by cycling and isolation time testing following maintenance, repair, or replacement work.
>
> Specification 4.6.3.2 requires verification of isolation valve operability through the application of an isolation test signal to verify the valve actuates properly. This testing is required every 18 months.
>
> Specification 4.6.3.3 requires verification of the isolation time for power-operated and automatic valves.
>
> Specification 4.6.4.4 requires that reactor instrumentation system isolation valves be demonstrated operable by cycling the valve every 18 months.
>
> Specification 4.6.1.2.d requires Type B and C leak testing of primary containment isolation valves.

The Company has completed a review of the primary containment isolation system (PCIS) Technical Specifications and determined that certain changes to Table 3.6.3-1 are needed to accurately reflect the as-built design of the primary containment isolation system. However, rather than propose changes to the table, CP&L believes it is more appropriate to request removal of Table 3.6.3-1 from the Technical Specifications and replace references to the table with references to valve listings contained in plant procedures (the Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal). Our reasons for this determination are described below.

First, removal of the valve listings from the Technical Specifications is appropriate in the context of Technical Specification reform. The recent industry and NRC efforts to improve the Technical Specifications have identified component listings as items for removal from the Technical Specifications because they do not directly contribute to plant safety. In addition, the level of detail of such listings has resulted in frequent requests for license amendments. This process has resulted in inherent inaccuracies in the Technical Specifications due to the delays in receiving license amendments brought about by the current regulatory environment. The NRC has already recommended and approved removal of the analogous snubber listing from the Technical Specifications. In addition, tables relating to fire protection devices and containment penetration protection circuit breakers have been removed from the Technical Specifications for other dockets. For example, listings relating to containment isolation valves, snubbers, containment penetration protection devices, motor operated valve thermal overload protection, the reactor vessel material surveillance program, engineered safety

features actuation system instrumentation response times, and reactor trip system instrumentation response times have been removed from the Technical Specifications for the Shearon Harris Plant and references to plant procedures incorporated. The use of component listings through reference to plant procedures ensures that timely information is maintained that reflects recent plant modifications.

Second, changes to these plant procedures are strictly controlled under 10CFR50.59. In addition, the Technical Specifications requirement to maintain primary containment integrity (Technical Specification 3.6.1.1), as well as ASME Code, Section XI testing requirements (Technical Specification 4.0.5) and 10CFR50, Appendix J requirements (Technical Specification 3.6.1.2) are maintained. Only the listing of valves would be subject to licensee control under 10CFR50.59. Plant procedures are auditable and subject to inspection by the NRC.

Third, appropriate mechanisms and controls exist to ensure correct testing for the valves through plant procedures. As discussed above, these referenced procedure(s) will incorporate valve list information developed to meet the requirements of Technical Specifications 4.6.3.1, 4.6.3.2, 4.6.3.3 and 4.6.3.4. In addition, Appendix J of 10CFR50 (implemented through Technical Specification 3/4.6.1.2) specifies leak rate test requirements for containment isolation valves, and thus maintenance of a list of such valves. Section XI of the ASME Code (implemented through Technical Specification 4.0.5) requires once per cycle testing of valve closure and timing for certain of the valves currently listed in Table 3.6.3-1. In addition, Technical Specification 4.3.2.2 requires once per cycle performance of a logic system functional test for most of the valves currently listed in Table 3.6.3-1. All the above requirements are appropriately integrated into plant procedure(s) which are subject to change only under requirements of 10CFR50.59. These plant procedures cover testing (to or beyond the present Technical Specification requirements) for the valves currently included in Table 3.6.3-1. A complete listing of containment isolation valves and required closure times is also provided as Table 6.2.4-2 to the Brunswick Updated Final Safety Analysis Report. References to plant procedures will replace references to Table 3.6.3-1 in the text of the Technical Specifications. Specific closure time testing requirements for the main steam isolation valves will be maintained in the Technical Specifications under Specification 3/4.4.7.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. Removal of the primary containment isolation valve listing from the Technical Specifications is appropriate in the context of Technical Specification reform. Recent industry and NRC efforts to improve the Technical Specifications have identified component listings as an item for removal because they do not directly contribute to plant safety. The use of component listings through reference to plant procedures will ensure that timely information is maintained that reflects recent plant modifications. Changes to these procedures will be controlled under the requirements of 10CFR50.59. Primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specifications 3.6.1.1 and 3.6.3 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3 (and thus the proper

identification, control, and periodic testing of the appropriate containment isolation valves). Relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not change their design or operation, nor does it affect any of the accident analyses; therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. As discussed in Item 1 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3 (and thus the proper identification, control, and periodic testing of the appropriate containment isolation valves). Thus, relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not change their design or operation so the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. As discussed in Items 1 and 2 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3. Furthermore, the relocation of the primary containment isolation valve listing to plant procedures will clarify the Technical Specifications and ensure a more up to date and accurate primary containment isolation valve listing, thereby avoiding possible operator confusion. Since there is no change in containment design or pressure retaining capability, the proposed change does not involve a significant reduction in the margin of safety.

Proposed Change Number 170

Replace the references to Table 3.6.3-1 with references to plant procedure ____________ in Surveillance Requirements 4.6.3.1, 4.6.3.2, 4.6.3.3, and 4.6.3.4 on Page 3/4 6-13.

Basis

Technical Specification Section 3.6 addresses containment systems. Table 3.6.3-1 provides a partial listing of the automatic primary containment isolation valves (PCIVs). The Brunswick Plant Technical Specifications currently require the following surveillances:

> Specification 4.6.3.1 requires that primary containment isolation valves be demonstrated operable by cycling and isolation time testing following maintenance, repair, or replacement work.
>
> Specification 4.6.3.2 requires verification of isolation valve operability through the application of an isolation test signal to verify the valve actuates properly. This testing is required every 18 months.
>
> Specification 4.6.3.3 requires verification of the isolation time for power-operated and automatic valves.
>
> Specification 4.6.4.4 requires that reactor instrumentation system isolation valves be demonstrated operable by cycling the valve every 18 months.
>
> Specification 4.6.1.2.d requires Type B and C leak testing of primary containment isolation valves.

The Company has completed a review of the primary containment isolation system (PCIS) Technical Specifications and determined that certain changes to Table 3.6.3-1 are needed to accurately reflect the as-built design of the primary containment isolation system. However, rather than propose changes to the table, CP&L believes it is more appropriate to request removal of Table 3.6.3-1 from the Technical Specifications and replace references to the table with references to valve listings contained in plant procedures (the Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal). Our reasons for this determination are described below.

First, removal of the valve listings from the Technical Specifications is appropriate in the context of Technical Specification reform. The recent industry and NRC efforts to improve the Technical Specifications have identified component listings as items for removal from the Technical Specifications because they do not directly contribute to plant safety. In addition, the level of detail of such listings has resulted in frequent requests for license amendments. This process has resulted in inherent inaccuracies in the Technical Specifications due to the delays in receiving license amendments brought about by the current regulatory environment. The NRC has already recommended and approved removal of the analogous snubber listing from the Technical Specifications. In addition, tables relating to fire protection devices and containment penetration protection circuit breakers have been removed from the Technical Specifications for other dockets. For example, listings relating to containment isolation valves, snubbers, containment penetration protection devices, motor operated valve thermal overload

protection, the reactor vessel material surveillance program, engineered safety features actuation system instrumentation response times, and reactor trip system instrumentation response times have been removed from the Technical Specifications for the Shearon Harris Plant and references to plant procedures incorporated. The use of component listings through reference to plant procedures ensures that timely information is maintained that reflects recent plant modifications.

Second, changes to these plant procedures are strictly controlled under 10CFR50.59. In addition, the Technical Specifications requirement to maintain primary containment integrity (Technical Specification 3.6.1.1), as well as ASME Code, Section XI testing requirements (Technical Specification 4.0.5) and 10CFR50, Appendix J requirements (Technical Specification 3.6.1.2) are maintained. Only the listing of valves would be subject to licensee control under 10CFR50.59. Plant procedures are auditable and subject to inspection by the NRC.

Third, appropriate mechanisms and controls exist to ensure correct testing for the valves through plant procedures. As discussed above, these referenced procedure(s) will incorporate valve list information developed to meet the requirements of Technical Specifications 4.6.3.1, 4.6.3.2, 4.6.3.3 and 4.6.3.4. In addition, Appendix J of 10CFR50 (implemented through Technical Specification 3/4.6.1.2) specifies leak rate test requirements for containment isolation valves, and thus maintenance of a list of such valves. Section XI of the ASME Code (implemented through Technical Specification 4.0.5) requires once per cycle testing of valve closure and timing for certain of the valves currently listed in Table 3.6.3-1. In addition, Technical Specification 4.3.2.2 requires once per cycle performance of a logic system functional test for most of the valves currently listed in Table 3.6.3-1. All the above requirements are appropriately integrated into plant procedure(s) which are subject to change only under requirements of 10CFR50.59. These plant procedures cover testing (to or beyond the present Technical Specification requirements) for the valves currently included in Table 3.6.3-1. A complete listing of containment isolation valves and required closure times is also provided as Table 6.2.4-2 to the Brunswick Updated Final Safety Analysis Report. References to plant procedures will replace references to Table 3.6.3-1 in the text of the Technical Specifications. Specific closure time testing requirements for the main steam isolation valves will be maintained in the Technical Specifications under Specification 3/4.4.7.

<u>10CFR50.92 Evaluation</u>

The proposed change does not involve a significant hazards consideration for the following reasons:

1. Removal of the primary containment isolation valve listing from the Technical Specifications is appropriate in the context of Technical Specification reform. Recent industry and NRC efforts to improve the Technical Specifications have identified component listings as an item for removal because they do not directly contribute to plant safety. The use of component listings through reference to plant procedures will ensure that timely information is maintained that reflects recent plant modifications. Changes to these procedures will be controlled under the requirements of 10CFR50.59. Primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specifications 3.6.1.1 and 3.6.3 will continue to require primary containment integrity while

operating in Operational Conditions 1, 2, and 3 (and thus the proper identification, control, and periodic testing of the appropriate containment isolation valves). Relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not change their design or operation, nor does it affect any of the accident analyses; therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. As discussed in Item 1 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3 (and thus the proper identification, control, and periodic testing of the appropriate containment isolation valves). Thus, relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not change their design or operation so the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. As discussed in Items 1 and 2 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3. Furthermore, the relocation of the primary containment isolation valve listing to plant procedures will clarify the Technical Specifications and ensure a more up to date and accurate primary containment isolation valve listing, thereby avoiding possible operator confusion. Since there is no change in containment design or pressure retaining capability, the proposed change does not involve a significant reduction in the margin of safety.

Proposed Change Number 171

Delete Footnote * under Specification 4.6.3.4 (BSEP-1 only) on Page 3/4 6-13, which states the following:

> The end of the current surveillance period for the surveillance requirements below may be extended beyond the time limit specified by Technical Specification 4.0.2a. After November 2, 1984, the plant shall not be operated in Operational Conditions 1, 2, or 3 until the surveillance requirements listed below have been completed. Upon accomplishment of the surveillances, the provisions of Technical Specification 4.0.2a shall apply.
>
> Technical Specification 4.6.3.4; as applicable to excess flow check valves B21-F047C, B21-F047D, B21-F049C, and B21-F049D.

Basis

Footnote * was added to the Technical Specifications via Amendment 72 to provide a one-time extension of the 18 month surveillance interval for four reactor instrumentation system isolation valves. This extension was applicable until November 2, 1984; therefore, the footnote is no longer applicable or necessary for normal operation.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. The proposed change deletes a footnote that no longer applies. The footnote was added to provide a one-time extension of the surveillance interval associated with the 18 month cycling of reactor instrumentation system isolation valves. This extension was applicable until November 2, 1984; thus, the proposed change has no effect on the probability of an accident, nor does it affect the consequences of any accidents.

2. The referenced footnote no longer applies to BSEP-1. The unit was shutdown prior to expiration of the footnote, the required surveillances were performed, and unit operation continued. The restrictions and requirements of the footnote are no longer necessary, and deletion of the footnote will not create the possibility of a new or different type of accident.

3. Footnote * was added as a one-time extension of a surveillance interval. The requirements of the footnote are no longer applicable since the required surveillance testing has been performed and unit operation recommenced. Therefore, this deletion has no impact on the margin of safety.

Proposed Change Number 172

Delete Table 3.6.3-1, "Primary Containment Isolation Valves," which is currently located on Pages 3/4 6-14 through 3/4 6-17, and repaginate the existing Pages 3/4 6-18 through 3/4 6-30 to accommodate its deletion.

Basis

Technical Specification Section 3.6 addresses containment systems. Table 3.6.3-1 provides a partial listing of the automatic primary containment isolation valves (PCIVs). The Brunswick Plant Technical Specifications currently require the following surveillances:

> Specification 4.6.3.1 requires that primary containment isolation valves be demonstrated operable by cycling and isolation time testing following maintenance, repair, or replacement work.
>
> Specification 4.6.3.2 requires verification of isolation valve operability through the application of an isolation test signal to verify the valve actuates properly. This testing is required every 18 months.
>
> Specification 4.6.3.3 requires verification of the isolation time for power-operated and automatic valves.
>
> Specification 4.6.4.4 requires that reactor instrumentation system isolation valves be demonstrated operable by cycling the valve every 18 months.
>
> Specification 4.6.1.2.d requires Type B and C leak testing of primary containment isolation valves.

The Company has completed a review of the primary containment isolation system (PCIS) Technical Specifications and determined that certain changes to Table 3.6.3-1 are needed to accurately reflect the as-built design of the primary containment isolation system. However, rather than propose changes to the table, CP&L believes it is more appropriate to request removal of Table 3.6.3-1 from the Technical Specifications and replace references to the table with references to valve listings contained in plant procedures (the Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal). Our reasons for this determination are described below.

First, removal of the valve listings from the Technical Specifications is appropriate in the context of Technical Specification reform. The recent industry and NRC efforts to improve the Technical Specifications have identified component listings as items for removal from the Technical Specifications because they do not directly contribute to plant safety. In addition, the level of detail of such listings has resulted in frequent requests for license amendments. This process has resulted in inherent inaccuracies in the Technical Specifications due to the delays in receiving license amendments brought about by the current regulatory environment. The NRC has already recommended and approved removal of the analogous snubber listing from the Technical Specifications. In addition, tables relating to fire protection devices and containment penetration protection circuit breakers have been removed from the Technical Specifications for other dockets. For example, listings relating to containment isolation valves, snubbers, containment penetration protection devices, motor operated valve thermal overload

protection, the reactor vessel material surveillance program, engineered safety features actuation system instrumentation response times, and reactor trip system instrumentation response times have been removed from the Technical Specifications for the Shearon Harris Plant and references to plant procedures incorporated. The use of component listings through reference to plant procedures ensures that timely information is maintained that reflects recent plant modifications.

Second, changes to these plant procedures are strictly controlled under 10CFR50.59. In addition, the Technical Specifications requirement to maintain primary containment integrity (Technical Specification 3.6.1.1), as well as ASME Code, Section XI testing requirements (Technical Specification 4.0.5) and 10CFR50, Appendix J requirements (Technical Specification 3.6.1.2) are maintained. Only the listing of valves would be subject to licensee control under 10CFR50.59. Plant procedures are auditable and subject to inspection by the NRC.

Third, appropriate mechanisms and controls exist to ensure correct testing for the valves through plant procedures. As discussed above, these referenced procedure(s) will incorporate valve list information developed to meet the requirements of Technical Specifications 4.6.3.1, 4.6.3.2, 4.6.3.3 and 4.6.3.4. In addition, Appendix J of 10CFR50 (implemented through Technical Specification 3/4.6.1.2) specifies leak rate test requirements for containment isolation valves, and thus maintenance of a list of such valves. Section XI of the ASME Code (implemented through Technical Specification 4.0.5) requires once per cycle testing of valve closure and timing for certain of the valves currently listed in Table 3.6.3-1. In addition, Technical Specification 4.3.2.2 requires once per cycle performance of a logic system functional test for most of the valves currently listed in Table 3.6.3-1. All the above requirements are appropriately integrated into plant procedure(s) which are subject to change only under requirements of 10CFR50.59. These plant procedures cover testing (to or beyond the present Technical Specification requirements) for the valves currently included in Table 3.6.3-1. A complete listing of containment isolation valves and required closure times is also provided as Table 6.2.4-2 to the Brunswick Updated Final Safety Analysis Report. References to plant procedures will replace references to Table 3.6.3-1 in the text of the Technical Specifications. Specific closure time testing requirements for the main steam isolation valves will be maintained in the Technical Specifications under Specification 3/4.4.7.

<u>10CFR50.92 Evaluation</u>

The proposed change does not involve a significant hazards consideration for the following reasons:

1. Removal of the primary containment isolation valve listing from the Technical Specifications is appropriate in the context of Technical Specification reform. Recent industry and NRC efforts to improve the Technical Specifications have identified component listings as an item for removal because they do not directly contribute to plant safety. The use of component listings through reference to plant procedures will ensure that timely information is maintained that reflects recent plant modifications. Changes to these procedures will be controlled under the requirements of 10CFR50.59. Primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specifications 3.6.1.1 and 3.6.3 will continue to require primary containment integrity while

operating in Operational Conditions 1, 2, and 3 (and thus the proper identification, control, and periodic testing of the appropriate containment isolation valves). Relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not change their design or operation, nor does it affect any of the accident analyses; therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. As discussed in Item 1 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3 (and thus the proper identification, control, and periodic testing of the appropriate containment isolation valves). Thus, relocation of the primary containment isolation valve listing from the Technical Specifications to plant procedures does not change their design or operation so the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. As discussed in Items 1 and 2 above, primary containment integrity is the function preserved by the primary containment isolation valves. Technical Specification 3.6.1.1 will continue to require primary containment integrity while operating in Operational Conditions 1, 2, and 3. Furthermore, the relocation of the primary containment isolation valve listing to plant procedures will clarify the Technical Specifications and ensure a more up to date and accurate primary containment isolation valve listing, thereby avoiding possible operator confusion. Since there is no change in containment design or pressure retaining capability, the proposed change does not involve a significant reduction in the margin of safety.

Proposed Change Number 173

Delete references to Table 3.6.5.2-1 in Technical Specification 3.6.5.2 on Page 3/4 6-19.

Basis

The Company requests the removal of Table 3.6.5.2-1 from the Technical Specifications and proposes to replace references to the table with references to valve listings contained in plant procedures. The Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal. Our reasons for proposing this change are described below.

The Company has completed a review of the primary containment isolation system (PCIS) Technical Specifications and determined that certain changes to Table 3.6.3-1 are needed to accurately reflect the as-built design of the primary containment isolation system. However, rather than propose changes to the table, CP&L believes it is more appropriate to request removal of Table 3.6.3-1 from the Technical Specifications and replace references to the table with references to valve listings contained in plant procedures. The design function of the primary containment isolation valves is to preserve primary containment integrity. The design function of the secondary containment isolation dampers is similar to that for the primary containment isolation valves (i.e., preservation of secondary containment integrity). Since the Company is proposing to relocate the listing of primary containment isolation dampers from the Technical Specifications to plant procedures, we believe it is also appropriate to request removal of the Table 3.6.5.2-1 from the Technical Specifications.

Changes to these plant procedures are strictly controlled under 10CFR50.59. In addition, the Technical Specifications requirement to maintain secondary containment integrity (Technical Specification 3.6.5.1) are maintained. Only the listing of valves would be subject to licensee control under 10CFR50.59. Plant procedures are auditable and subject to inspection by the NRC.

Removal of the secondary containment isolation damper listing from the Technical Specifications is appropriate in the context of Technical Specification reform. The recent industry and NRC efforts to improve the Technical Specifications have identified component listings as items for removal from the Technical Specifications because they do not directly contribute to plant safety. In addition, the level of detail of such listings has resulted in frequent requests for license amendments. This process has resulted in inherent inaccuracies in the Technical Specifications due to the delays in receiving license amendments brought about by the current regulatory environment. The use of component listings through reference to plant procedures will ensure that timely information is maintained that reflects recent plant modifications.

Appropriate mechanisms and controls exist to ensure correct testing for the secondary containment isolation dampers through plant procedures. As discussed above, these referenced procedure(s) will incorporate isolation damper list information developed to meet the requirements of Technical Specifications 4.6.5.2.a, 4.6.5.2.b, 4.6.5.2.c.1 and 4.6.5.2.c.2. In addition, Technical Specification 4.3.2.2 requires once per cycle performance of a logic system functional test for the dampers currently listed in Table 3.6.5.2-1. All the above requirements are appropriately integrated into plant procedure(s) which

are subject to change only under requirements of 10CFR50.59. These plant procedures cover testing (to or beyond the present Technical Specification requirements) for the secondary containment isolation dampers currently included in Table 3.6.5.2-1. References to plant procedures will replace references to Table 3.6.5.2-1 in the text of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. Removal of the secondary containment isolation damper listing from the Technical Specifications is appropriate in the context of Technical Specification reform. Recent industry and NRC efforts to improve the Technical Specifications have identified component listings as an item for removal because they do not directly contribute to plant safety. The use of component listings through reference to plant procedures will ensure that timely information is maintained that reflects recent plant modifications. Changes to these procedures will be controlled under the requirements of 10CFR50.59. Secondary containment integrity is the function preserved by the secondary containment isolation dampers. Technical Specification 3.6.5.1 will continue to require secondary containment integrity while operating in Operational Conditions 1, 2, 3, 5 and when moving irradiated fuel assemblies in the secondary containment. Thus, the proper identification, control, and periodic testing of the appropriate secondary containment isolation dampers is necessary. Relocation of the secondary containment isolation damper listing from the Technical Specifications to plant procedures does change their design or operation, nor does it affect any of the accident analyses; therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. As discussed in Item 1 above, secondary containment integrity is the function preserved by the secondary containment isolation dampers. Technical Specification 3.6.5.1 will continue to require secondary containment integrity while operating in Operational Conditions 1, 2, 3, 5 and when moving irradiated fuel assemblies in the secondary containment (and thus the proper identification, control, and periodic testing of the appropriate secondary containment isolation dampers). Thus, relocation of the secondary containment isolation damper listing from the Technical Specifications to plant procedures does not change their design or operation so the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. As discussed in Items 1 and 2 above, secondary containment integrity is the function preserved by the secondary containment isolation dampers. Technical Specification 3.6.5.1 will continue to require secondary containment integrity while operating in Operational Conditions 1, 2, 3, 5 and when moving irradiated fuel assemblies in the secondary containment. Furthermore, the relocation of the secondary containment isolation damper listing to plant procedures will clarify the Technical Specifications and ensure a more up to date and accurate secondary containment isolation damper listing, thereby avoiding possible operator confusion. Since there is no

change in containment design or pressure retaining capability, the proposed change does not involve a significant reduction in the margin of safety.

Proposed Change Number 174

Delete reference to Table 3.6.5.2-1 in Surveillance Requirement 4.6.5.2 on Page 3/4 6-20.

Basis

The Company requests the removal of Table 3.6.5.2-1 from the Technical Specifications and proposes to replace references to the table with references to valve listings contained in plant procedures. The Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal. Our reasons for proposing this change are described below.

The Company has completed a review of the primary containment isolation system (PCIS) Technical Specifications and determined that certain changes to Table 3.6.3-1 are needed to accurately reflect the as-built design of the primary containment isolation system. However, rather than propose changes to the table, CP&L believes it is more appropriate to request removal of Table 3.6.3-1 from the Technical Specifications and replace references to the table with references to valve listings contained in plant procedures. The design function of the primary containment isolation valves is to preserve primary containment integrity. The design function of the secondary containment isolation dampers is similar to that for the primary containment isolation valves (i.e., preservation of secondary containment integrity). Since the Company is proposing to relocate the listing of primary containment isolation dampers from the Technical Specifications to plant procedures, we believe it is also appropriate to request removal of the Table 3.6.5.2-1 from the Technical Specifications.

Changes to these plant procedures are strictly controlled under 10CFR50.59. In addition, the Technical Specifications requirement to maintain secondary containment integrity (Technical Specification 3.6.5.1), as well as ASME code testing requirements (Technical Specification 4.0.5) are maintained. Only the listing of valves would be subject to licensee control under 10CFR50.59. Plant procedures are auditable and subject to inspection by the NRC.

Removal of the secondary containment isolation damper listing from the Technical Specifications is appropriate in the context of Technical Specification reform. The recent industry and NRC efforts to improve the Technical Specifications have identified component listings as items for removal from the Technical Specifications because they do not directly contribute to plant safety. In addition, the level of detail of such listings has resulted in frequent requests for license amendments. This process has resulted in inherent inaccuracies in the Technical Specifications due to the delays in receiving license amendments brought about by the current regulatory environment. The use of component listings through reference to plant procedures will ensure that timely information is maintained that reflects recent plant modifications.

Appropriate mechanisms and controls exist to ensure correct testing for the secondary containment isolation dampers through plant procedures. As discussed above, these referenced procedure(s) will incorporate isolation damper list information developed to meet the requirements of Technical Specifications 4.6.5.2.a, 4.6.5.2.b, 4.6.5.2.c.1 and 4.6.5.2.c.2. Section XI of the ASME Code (implemented through Technical Specification 4.0.5) requires once per cycle testing of valve closure and timing for the secondary containment

isolation dampers currently listed in Table 3.6.5.2-1. In addition, Technical Specification 4.3.2.1 requires once per cycle performance of a logic system functional test for the dampers currently listed in Table 3.6.5.2-1. All the above requirements are appropriately integrated into plant procedure(s) which are subject to change only under requirements of 10CFR50.59. These plant procedures cover testing (to or beyond the present Technical Specification requirements) for the secondary containment isolation dampers currently included in Table 3.6.5.2-1. References to plant procedures will replace references to Table 3.6.5.2-1 in the text of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. Removal of the secondary containment isolation damper listing from the Technical Specifications is appropriate in the context of Technical Specification reform. Recent industry and NRC efforts to improve the Technical Specifications have identified component listings as an item for removal because they do not directly contribute to plant safety. The use of component listings through reference to plant procedures will ensure that timely information is maintained that reflects recent plant modifications. Changes to these procedures will be controlled under the requirements of 10CFR50.59. Secondary containment integrity is the function preserved by the secondary containment isolation dampers. Technical Specification 3.6.5.1 will continue to require secondary containment integrity while operating in Operational Conditions 1, 2, 3, 5 and when moving irradiated fuel assemblies in the secondary containment. Thus, the proper identification, control, and periodic testing of the appropriate secondary containment isolation dampers is necessary. Relocation of the secondary containment isolation damper listing from the Technical Specifications to plant procedures does affect any of the accident analyses; therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. As discussed in Item 1 above, secondary containment integrity is the function preserved by the secondary containment isolation dampers. Technical Specification 3.6.5.1 will continue to require secondary containment integrity while operating in Operational Conditions 1, 2, 3, 5 and when moving irradiated fuel assemblies in the secondary containment (and thus the proper identification, control, and periodic testing of the appropriate secondary containment isolation dampers). Thus, relocation of the secondary containment isolation damper listing from the Technical Specifications to plant procedures does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. As discussed in Items 1 and 2 above, secondary containment integrity is the function preserved by the secondary containment isolation dampers. Technical Specification 3.6.5.1 will continue to require secondary containment integrity while operating in Operational Conditions 1, 2, 3, 5 and when moving irradiated fuel assemblies in the secondary containment. Furthermore, the relocation of the secondary containment isolation damper listing to plant procedures will clarify the Technical Specifications and ensure a more up to date and accurate secondary containment isolation damper

listing, thereby avoiding possible operator confusion. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Proposed Change Number 175

Delete Technical Specification Table 3.6.5.2-1, "Secondary Containment Automatic Isolation Dampers," from Page 3/4 6-24.

Basis

The Company requests the removal of Table 3.6.5.2-1 from the Technical Specifications and proposes to replace references to the table with references to valve listings contained in plant procedures. The Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal. Our reasons for proposing this change are described below.

The Company has completed a review of the primary containment isolation system (PCIS) Technical Specifications and determined that certain changes to Table 3.6.3-1 are needed to accurately reflect the as-built design of the primary containment isolation system. However, rather than propose changes to the table, CP&L believes it is more appropriate to request removal of Table 3.6.3-1 from the Technical Specifications and replace references to the table with references to valve listings contained in plant procedures. The design function of the primary containment isolation valves is to preserve primary containment integrity. The design function of the secondary containment isolation dampers is similar to that for the primary containment isolation valves (i.e., preservation of secondary containment integrity). Since the Company is proposing to relocate the listing of primary containment isolation dampers from the Technical Specifications to plant procedures, we believe it is also appropriate to request removal of the Table 3.6.5.2-1 from the Technical Specifications.

Changes to these plant procedures are strictly controlled under 10CFR50.59. In addition, the Technical Specifications requirement to maintain secondary containment integrity (Technical Specification 3.6.5.1) are maintained. Only the listing of valves would be subject to licensee control under 10CFR50.59. Plant procedures are auditable and subject to inspection by the NRC.

Removal of the secondary containment isolation damper listing from the Technical Specifications is appropriate in the context of Technical Specification reform. The recent industry and NRC efforts to improve the Technical Specifications have identified component listings as items for removal from the Technical Specifications because they do not directly contribute to plant safety. In addition, the level of detail of such listings has resulted in frequent requests for license amendments. This process has resulted in inherent inaccuracies in the Technical Specifications due to the delays in receiving license amendments brought about by the current regulatory environment. The use of component listings through reference to plant procedures will ensure that timely information is maintained that reflects recent plant modifications.

Appropriate mechanisms and controls exist to ensure correct testing for the secondary containment isolation dampers through plant procedures. As discussed above, these referenced procedure(s) will incorporate isolation damper list information developed to meet the requirements of Technical Specifications 4.6.5.2.a, 4.6.5.2.b, 4.6.5.2.c.1 and 4.6.5.2.c.2. In addition, Technical Specification 4.3.2.2 requires once per cycle performance of a logic system functional test for the dampers currently listed in Table 3.6.5.2-1. All the above requirements are appropriately integrated into plant procedure(s) which

are subject to change only under requirements of 10CFR50.59. These plant procedures cover testing (to or beyond the present Technical Specification requirements) for the secondary containment isolation dampers currently included in Table 3.6.5.2-1. References to plant procedures will replace references to Table 3.6.5.2-1 in the text of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change does not involve a significant hazards consideration for the following reasons:

1. Removal of the secondary containment isolation damper listing from the Technical Specifications is appropriate in the context of Technical Specification reform. Recent industry and NRC efforts to improve the Technical Specifications have identified component listings as an item for removal because they do not directly contribute to plant safety. The use of component listings through reference to plant procedures will ensure that timely information is maintained that reflects recent plant modifications. Changes to these procedures will be controlled under the requirements of 10CFR50.59. Secondary containment integrity is the function preserved by the secondary containment isolation dampers. Technical Specification 3.6.5.1 will continue to require secondary containment integrity while operating in Operational Conditions 1, 2, 3, 5 and when moving irradiated fuel assemblies in the secondary containment. Thus, the proper identification, control, and periodic testing of the appropriate secondary containment isolation dampers is necessary. Relocation of the secondary containment isolation damper listing from the Technical Specifications to plant procedures does change their design or operation, nor does it affect any of the accident analyses; therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. As discussed in Item 1 above, secondary containment integrity is the function preserved by the secondary containment isolation dampers. Technical Specification 3.6.5.1 will continue to require secondary containment integrity while operating in Operational Conditions 1, 2, 3, 5 and when moving irradiated fuel assemblies in the secondary containment (and thus the proper identification, control, and periodic testing of the appropriate secondary containment isolation dampers). Thus, relocation of the secondary containment isolation damper listing from the Technical Specifications to plant procedures does not change their design or operation so the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. As discussed in Items 1 and 2 above, secondary containment integrity is the function preserved by the secondary containment isolation dampers. Technical Specification 3.6.5.1 will continue to require secondary containment integrity while operating in Operational Conditions 1, 2, 3, 5 and when moving irradiated fuel assemblies in the secondary containment. Furthermore, the relocation of the secondary containment isolation damper listing to plant procedures will clarify the Technical Specifications and ensure a more up to date and accurate secondary containment isolation damper listing, thereby avoiding possible operator confusion. Since there is no

change in containment design or pressure retaining capability, the proposed change does not involve a significant reduction in the margin of safety.

Proposed Change Number 176

Add paragraph to Bases Section 3/4.6.3, Primary Containment Isolation Valves, on Page B 3/4 6-5 to reflect the requirement to maintain a listing at the plant of primary containment isolation valves and to assure that changes to the list are made in accordance with the requirements of 10CFR50.59.

Basis

The Company has completed a review of the primary containment isolation system (PCIS) Technical Specifications and determined that certain changes to Table 3.6.3-1 are needed to accurately reflect the as-built design of the primary containment isolation system. However, rather than propose changes to the table, CP&L believes it is more appropriate to request removal of Table 3.6.3-1 from the Technical Specifications and replace references to the table with references to valve listings contained in plant procedures (the Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal).

Changes to these plant procedures are strictly controlled under 10CFR50.59. In addition, the Technical Specifications requirement to maintain primary containment integrity (Technical Specification 3.6.1.1), as well as ASME Code, Section XI testing requirements (Technical Specification 4.0.5) and 10CFR50, Appendix J requirements (Technical Specification 3.6.1.2) are maintained. Only the listing of valves would be subject to licensee control under 10CFR50.59. Plant procedures are auditable and subject to inspection by the NRC.

Removal of the valve listings from the Technical Specifications is appropriate in the context of Technical Specification reform. The recent industry and NRC efforts to improve the Technical Specifications have identified component listings as items for removal from the Technical Specifications because they do not directly contribute to plant safety. In addition, the level of detail of such listings has resulted in frequent requests for license amendments. This process has resulted in inherent inaccuracies in the Technical Specifications due to the delays in receiving license amendments brought about by the current regulatory environment. The use of component listings through reference to plant procedures will ensure that timely information is maintained that reflects recent plant modifications.

Appropriate mechanisms and controls exist to ensure correct testing for the valves through plant procedures. As discussed above, these referenced procedure(s) will incorporate valve list information developed to meet the requirements of Technical Specifications 4.6.3.1, 4.6.3.2, 4.6.3.3 and 4.6.3.4. In addition, Appendix J of 10CFR50 (implemented through Technical Specification 3/4.6.1.2) specifies leak rate test requirements for containment isolation valves, and thus maintenance of a list of such valves. Section XI of the ASME Code (implemented through Technical Specification 4.0.5) requires once per cycle testing of valve closure and timing for certain of the valves currently listed in Table 3.6.3-1. In addition, Technical Specification 4.3.2.2 requires once per cycle performance of a logic system functional test for most of the valves currently listed in Table 3.6.3-1. All the above requirements are appropriately integrated into plant procedure(s) which are subject to change only under requirements of 10CFR50.59. These plant procedures cover testing (to or beyond the present Technical Specification requirements) for the valves currently included in Table 3.6.3-1. A complete

listing of containment isolation valves and required closure times is also provided as Table 6.2.4-2 to the Brunswick Updated Final Safety Analysis Report. References to plant procedures will replace references to Table 3.6.3-1 in the text of the Technical Specifications. Specific closure time testing requirements for the main steam isolation valves will be maintained in the Technical Specifications under Specification 3/4.4.7.

<u>10CFR50.92 Evaluation</u>

The proposed change revises the BASES section consistent with the relocation of the listing of primary containment isolation valves from Technical Specification Table 3.6.3-1 to a plant procedure.

A 10CFR50.92 significant hazards evaluation is not provided for this change since the Bases are only summary statements in support of the Technical Specifications and are not considered part of the actual Technical Specifications as provided in 10CFR50.36.

Proposed Change Number 177

Add paragraph to Bases Section 3/4.6.5, Secondary Containment Isolation Dampers, on Page B 3/4 6-6 to reflect the requirement to maintain a listing at the plant of secondary containment isolation dampers and to assure that changes to the list are made in accordance with the requirements of 10CFR50.59.

Basis

The Company requests the removal of Table 3.6.5.2-1 from the Technical Specifications and proposes to replace references to the table with references to valve listings contained in plant procedures. The Company will provide a specific procedure reference for inclusion in the Technical Specifications via a subsequent submittal. Our reasons for proposing this change are described below.

The Company has completed a review of the primary containment isolation system (PCIS) Technical Specifications and determined that certain changes to Table 3.6.3-1 are needed to accurately reflect the as-built design of the primary containment isolation system. However, rather than propose changes to the table, CP&L believes it is more appropriate to request removal of Table 3.6.3-1 from the Technical Specifications and replace references to the table with references to valve listings contained in plant procedures. The design function of the primary containment isolation valves is to preserve primary containment integrity. The design function of the secondary containment isolation dampers is similar to that for the primary containment isolation valves (i.e., preservation of secondary containment integrity). Since the Company is proposing to relocate the listing of primary containment isolation dampers from the Technical Specifications to plant procedures, we believe it is also appropriate to request removal of the Table 3.6.5.2-1 from the Technical Specifications.

Changes to these plant procedures are strictly controlled under 10CFR50.59. In addition, the Technical Specifications requirement to maintain primary containment integrity (Technical Specification 3.6.1.1) are maintained. Only the listing of valves would be subject to licensee control under 10CFR50.59. Plant procedures are auditable and subject to inspection by the NRC.

Removal of the secondary containment isolation damper listing from the Technical Specifications is appropriate in the context of Technical Specification reform. The recent industry and NRC efforts to improve the Technical Specifications have identified component listings as items for removal from the Technical Specifications because they do not directly contribute to plant safety. In addition, the level of detail of such listings has resulted in frequent requests for license amendments. This process has resulted in inherent inaccuracies in the Technical Specifications due to the delays in receiving license amendments brought about by the current regulatory environment. The use of component listings through reference to plant procedures will ensure that timely information is maintained that reflects recent plant modifications.

Appropriate mechanisms and controls exist to ensure correct testing for the secondary containment isolation dampers through plant procedures. As discussed above, these referenced procedure(s) will incorporate isolation damper list information developed to meet the requirements of Technical Specifications 4.6.5.2.a, 4.6.5.2.b, 4.6.5.2.c.1 and 4.6.5.2.c.2. In addition, Technical Specification 4.3.2.1 requires once per cycle performance of a logic

system functional test for the dampers currently listed in Table 3.6.5.2-1. All the above requirements are appropriately integrated into plant procedure(s) which are subject to change only under requirements of 10CFR50.59. These plant procedures cover testing (to or beyond the present Technical Specification requirements) for the secondary containment isolation dampers currently included in Table 3.6.5.2-1. References to plant procedures will replace references to Table 3.6.5.2-1 in the text of the Technical Specifications.

10CFR50.92 Evaluation

The proposed change revises the BASES section consistent with the relocation of the listing of secondary containment isolation valves from Technical Specification Table 3.6.3-1 to a plant procedure.

A 10CFR50.92 significant hazards evaluation is not provided for this change since the Bases are only summary statements in support of the Technical Specifications and are not considered part of the actual Technical Specifications as provided in 10CFR50.36.