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10 CFR 50.90

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-45

Subject: License Amendment Request to Revise Technical Specifications (TS) 3.3.6, "Containment Ventilation Isolation Instrumentation"

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit or early site permit," Exelon Generation Company, LLC, (EGC) requests amendments to Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, and Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2. Specifically, this amendment request proposes to revise Footnote (b) of TS Table 3.3.6-1, "Containment Ventilation Isolation Instrumentation," which specifies the "Containment Radiation – High" trip setpoint for two containment area radiation monitors (i.e., 1(2)RE-AR011 and 1(2) RE-AR012). Upon sensing a high radiation condition, these area radiation monitors provide an isolation signal to the containment normal purge, minipurge and post-LOCA (Loss of Coolant Accident) systems' containment isolation valves.

The proposed changes would revise the "Containment Radiation – High" trip setpoint from the current, overly conservative value (i.e., a submersion dose rate of ≤ 10 mR/hr in the containment building), to ≤ 2 times the containment building background radiation reading at rated thermal power, consistent with NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." This change is needed to eliminate the unnecessary operator burden and distraction associated with frequent trip setpoint changes, frequent alert alarms and high alarms with associated unwarranted containment isolation signals. In support of the proposed setpoint change, EGC also proposes to make a change to TS 3.9.4, "Containment Penetrations." LCO 3.9.4, Item c.2, currently allows the option of moving RECENTLY IRRADIATED FUEL with the containment purge valves open but capable of being closed by an OPERABLE Containment Ventilation Isolation System; this LCO would be deleted. A number of additional TS changes, associated with the above proposed changes, are also proposed for consistency.

The attached request is subdivided as follows:

Attachment 1 provides an evaluation of the proposed change.

Attachments 2 and 3 include the marked-up TS pages with the proposed change indicated for the Braidwood Station and the Byron Station, respectively.

Attachments 4 and 5 include the marked-up TS Bases pages with the proposed change indicated for the Braidwood Station and the Byron Station, respectively. The TS Bases pages are provided for information only and do not require NRC approval.

The proposed amendment has been reviewed by the Braidwood Station and Byron Station Plant Operations Review Committees and approved by their respective Nuclear Safety Review Boards in accordance with the requirements of the EGC Quality Assurance Program.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the State of Illinois of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State of Illinois official.

EGC requests approval of the proposed license amendment by December 20, 2013. Once approved, the amendments will be implemented within 30 days.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this letter, please contact Joseph A. Bauer at (630) 657-2804.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 21st day of December 2012.

Respectfully,

David M. Gullott
Manager – Licensing
Exelon Generation Company, LLC

Attachments:
1. Evaluation of Proposed Change
2. Markup of Technical Specifications Pages for Braidwood Station, Units 1 and 2
3. Markup of Technical Specifications Pages for Byron Station, Units 1 and 2
4. Markup of Technical Specifications Bases Pages for Braidwood Station, Units 1 and 2
5. Markup of Technical Specifications Bases Pages for Byron Station, Units 1 and 2

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector, Braidwood Station
NRC Senior Resident Inspector, Byron Station
NRC Project Manager, NRR – Braidwood and Byron Stations
Illinois Emergency Management Agency - Division of Nuclear Safety

ATTACHMENT 1
Evaluation of Proposed Change

Subject: License Amendment Request to Revise Technical Specifications (TS) 3.3.6, "Containment Ventilation Isolation Instrumentation"

1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit or early site permit," Exelon Generation Company, LLC, (EGC) requests amendments to Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, and Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2. Specifically, this amendment request proposes to revise Footnote (b) of Technical Specification (TS) Table 3.3.6-1, "Containment Ventilation Isolation Instrumentation," which specifies the "Containment Radiation – High" trip setpoint for two containment area radiation monitors (i.e., 1(2)RE-AR011 and 1(2) RE-AR012). Upon sensing a high radiation condition, these area radiation monitors provide an isolation signal to the containment normal purge, minipurge and post- LOCA (Loss of Coolant Accident) systems containment isolation valves.

As detailed below in Section 2.0, "Detailed Description," and Section 3.0, "Technical Evaluation," the proposed change would revise the "Containment Radiation – High" trip setpoint from the current, overly conservative value (i.e., a submersion dose rate of ≤ 10 mR/hr in the containment building), to ≤ 2 times the containment building background radiation reading at rated thermal power, consistent with NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," (Reference 1) This change is needed to eliminate the unnecessary operator burden and distraction associated with frequent trip setpoint changes, frequent alert alarms and high alarms with associated unwarranted containment isolation signals. A nominal value for the containment background radiation level at 100% power (i.e., rated thermal power (RTP)) at all of the Braidwood Station or Byron Station units ranges from approximately 30 – 50 mR/hr. Therefore, the proposed "Containment Radiation – High" trip setpoint of 2x background at RTP will be in the range of approximately 60 – 100 mR/hr; however, may be higher or lower depending on actual containment conditions.

In support of the proposed setpoint change, EGC also proposes to make a change to TS 3.9.4, "Containment Penetrations." LCO 3.9.4, Item c.2 currently allows the option of moving RECENTLY IRRADIATED FUEL with the containment purge valves open but capable of being closed by an OPERABLE Containment Ventilation Isolation System. This LCO would be deleted to eliminate any potential release from the containment purge valves as all penetrations providing direct access from the containment atmosphere to the outside atmosphere will now be closed by a manual or automatic isolation valve, blind flange, or equivalent during movement of RECENTLY IRRADIATED FUEL. Four other additional TS changes associated with the deletion of LCO 3.9.4, Item c.2, are proposed for consistency (i.e., deleting a NOTE regarding MODE applicability in TS 3.3.6 CONDITION B, deleting TS 3.3.6 CONDITION C related only to LCO 3.9.4.c.2, deleting Footnote (a) of Table 3.3.6-1 regarding MODE applicability; and deleting two surveillances (i.e., SR 3.9.4.2 and SR 3.9.4.3) related to LCO 3.9.4.c.2).

Approval of this amendment application is requested by December 20, 2013. Once approved, the amendments will be implemented within 30 days.

2.0 DETAILED DESCRIPTION

This amendment request proposes to:

1. delete the NOTE in TS 3.3.6 Condition B regarding MODE applicability;

2. delete TS 3.3.6 Condition C; which address inoperable radiation monitoring channels;

3. delete Footnote (a) of TS Table 3.3.6-1, "Containment Ventilation Isolation Instrumentation," which identifies a MODE of APPLICABILITY for the associated FUNCTION;

4. revise Footnote (b) of TS Table 3.3.6-1, "Containment Ventilation Isolation Instrumentation," which specifies the "Containment Radiation – High" trip setpoint for two containment area radiation monitors (i.e., 1(2)RE-AR011 and 1(2)RE-AR012);

5. delete LCO 3.9.4.c.2 which partially defines the status of containment penetrations during movement of RECENTLY IRRADIATED FUEL within the containment; and

6. delete SR 3.9.4.2 and SR 3.9.4.3 which are surveillances for the containment purge valves.

TS 3.3.6 Condition B NOTE

The NOTE in TS 3.3.6 Condition B currently states:

> *"Only applicable in MODE 1, 2, 3, or 4."*

This NOTE will be deleted as the proposed changes eliminate the one other applicable MODE (i.e., *"When Item c.2 of LCO 3.9.4 is required")*; therefore, the NOTE is no longer needed.

TS 3.3.6 Condition C

TS 3.3.6 Condition C addresses inoperability of the radiation monitoring channels. Condition C is modified by a NOTE that states, *"Only applicable when Item c.2 of LCO 3.9.4 is required."* Since one of the proposed changes is to delete LCO 3.9.4.c.2, it follows that the entire Condition C is also deleted.

Footnote (a) of TS Table 3.3.6-1,

Footnote (a) of TS Table 3.3.6-1, "Containment Ventilation Isolation Instrumentation," identifies a MODE of APPLICABILITY for FUNCTION 4, "Containment Radiation-High." Footnote (a) currently states the following:

> *"When Item c.2 of LCO 3.9.4 is required."*

The proposed change would delete Footnote (a) in its entirety as Item c.2 of LCO 3.9.4 will be deleted as part of these proposed changes.

Footnote (b) of TS Table 3.3.6-1

Footnote (b) of TS Table 3.3.6-1, specifies the "Containment Radiation – High" trip setpoint for two containment area radiation monitors (i.e., 1(2)RE-AR011 and 1(2)RE-AR012). Upon sensing a high radiation condition, these area radiation monitors provide an isolation signal to the containment normal purge, minipurge and post- LOCA systems' containment isolation valves.

The setpoint for the "Containment Radiation – High" trip, given in Footnote (b) of TS Table 3.3.6-1, currently states the following:

> *"Trip setpoint shall be established such that actual submersion dose rate is ≤ 10 mR/hr in the Containment Building. The trip setpoint may be increased above this value in accordance with the methodology established in the Offsite Dose Calculation Manual."*

The proposed change would renumber Footnote (b) to (a) and revise it as follows:

> *"Trip setpoint shall be established at ≤ 2 x background in the Containment Building at RTP."*

<u>LCO 3.9.4.c.2</u>

LCO 3.9.4.c.2 partially defines the status of containment penetrations during movement of RECENTLY IRRADIATED FUEL. LCO 3.9.4.c, Items 1 and 2 currently states the following:

c. *Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:*

   1. *Closed by a manual or automatic isolation valve, blind flange, or equivalent, or*

   2. *Capable of being closed by an OPERABLE Containment Ventilation Isolation System.*

The proposed change would delete Item c.2 in its entirety as the option of moving RECENTLY IRRADIATED FUEL with the containment purge valves open but capable of being closed by an OPERABLE Containment Ventilation Isolation System is, conservatively, being eliminated as discussed below in Section 3.4, "Justification of Proposed Setpoint and Associated Changes." LCO 3.9.4.c.1 and 2 would be changed to:

c. *Each penetration providing direct access from the containment atmosphere to the outside atmosphere closed by a manual or automatic isolation valve, blind flange, or equivalent.*

<u>SR 3.9.4.2 and SR 3.9.4.3</u>

The proposed change would delete SR 3.9.4.2 and SR 3.9.4.3. SR 3.9.4.2 and SR 3.9.4.3 both address verifying operability of the containment purge valves in support of an OPERABLE Containment Ventilation Isolation System. Since LCO 3.9.4.c.2 is being deleted, operability of the Containment Ventilation Isolation System is not required; therefore, these SRs are no longer needed.

A detailed justification for the proposed setpoint and the associated changes is provided below in Section 3.0.

Attachments 2 and 3 include the marked-up TS pages with the proposed change indicated for the Braidwood Station and the Byron Station, respectively.

Attachments 4 and 5 include the marked-up TS Bases pages with the proposed change indicated for the Braidwood Station and the Byron Station, respectively. The TS Bases pages are provided for information only and do not require NRC approval.

3.0 TECHNICAL EVALUATION

Background

Currently, the setpoint for the "Containment Radiation – High" trip, given in Footnote (b) of TS Table 3.3.6-1, states the following:

> *"Trip setpoint shall be established such that actual submersion dose rate is ≤ 10 mR/hr in the Containment Building. The trip setpoint may be increased above this value in accordance with the methodology established in the Offsite Dose Calculation Manual."*

Note that the term "submersion dose" refers to the total effective dose equivalent (i.e., both internal and external) attributed to being immersed in a cloud of radioactive material, primarily noble gases. Therefore, the trip setpoint is referring to a radiation level attributable to a gaseous concentration in containment in addition to the normal background radiation level (i.e., primarily gamma radiation) in the containment building. Simplistically, the trip setpoint is referred to as "≤ 10 mR/hr above background."

Based on operating experience (further described in Section 3.3 below), the current setpoint has proven to be overly conservative, (i.e., too small of a value above the containment background radiation level). Maintaining the setpoint at ≤ 10 mR/hr above background during power ascension following a refueling outage and during routine power level changes requires frequent setpoint changes since containment background radiation level is proportional to power level. In addition, due to normal radiation monitor fluctuations of approximately 10 mR/hr, frequent alert alarms and high alarms with associated unwarranted containment isolation signals, have been received.

To eliminate this unnecessary operator burden and distraction, EGC proposes to change the "Containment Radiation – High" trip setpoint value to ≤ 2 times containment background at rated thermal power (RTP). This value is consistent with NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Note that the suggested (i.e., bracketed) value in NUREG-1431 is given as "≤ [2 x background]," however, the proposed change would modify the NUREG-1431 value to "≤ 2 x background at RTP" to avoid the need for setpoint changes during power ascension following a unit startup or during up and down power ramping maneuvers. A nominal value for the containment background radiation level at 100% power (i.e., RTP) at all of the Braidwood Station or Byron Station units ranges from approximately 30 – 50 mR/hr. Therefore, the proposed "Containment Radiation – High" trip setpoint of 2x background at RTP will be in the range of approximately 60 – 100 mR/hr; however, may be higher or lower depending on actual containment conditions.

As noted above, in addition to specifying the "Containment Radiation – High" numerical trip setpoint value of "≤ *10 mR/hr*," TS Table 3.3.6-1, Footnote (b) also states that,

> *"The trip setpoint may be increased above this value in accordance with the methodology established in the Offsite Dose Calculation Manual."*

This latitude in the setpoint value is intended to be used during actual containment purge and vent evolutions as noted in the Technical Requirements Manual (TRM); however, since the proposed setpoint of "≤ 2 x background at RTP" provides sufficient margin above the expected routine containment release activity, this latitude is no longer needed and will be deleted from the setpoint description in the TRM as appropriate.

The appropriate Radiation Protection Department procedures will define how to establish the "background in the Containment Building at rated thermal power" for all operational conditions, (e.g., startup of a new fuel cycle, startup after a maintenance outage, refueling outages, etc.) and conduct reviews throughout the cycle to ensure the "Containment Radiation – High" trip setpoint remains at an appropriate value.

The below discussion will show that the proposed change to the "Containment Radiation – High" trip setpoint remains consistent with the current licensing basis for the containment ventilation isolation function and bounded by the associated analysis. This discussion will include the following sections:

Section 3.1, "System Description," will present a brief description of the Primary Containment Purge system and the Containment Ventilation Isolation Instrumentation system to discuss the basic function of the systems. Note that complete system descriptions can be found in UFSAR Section 9.4.9, "Primary Containment Purge System," and UFSAR Section 6.2.4, "Containment Isolation System."

Section 3.2, "Licensing Basis," will review the current licensing basis for the containment ventilation isolation system.

Section 3.3, "Operating History – Basis for Requested Change," will discuss the past operational challenges the current containment ventilation isolation setpoint has presented, and thus prompting this request.

Section 3.4, "Justification of Proposed Setpoint and Associated TS Changes," will clearly show that the calculated offsite dose, given the proposed setpoint change, remains bounded by the current analysis. The other TS changes in support of the proposed setpoint change will also be discussed.

3.1 System Description

A schematic of the primary containment purge system is shown in Figure 1.

The primary containment purge system is subdivided into three subsystems: miniflow purge, normal purge, and post-LOCA (Loss of Coolant Accident) purge. These subsystems serve the containment during normal plant operating conditions, during planned reactor shutdowns, and during post-LOCA operating conditions. An independent purge system is provided for each unit's containment; however, the normal containment purge system is not normally used at Byron and Braidwood Stations; i.e., the normal purge valves are maintained sealed closed in MODES 1 – 4, as required by TS 3.6.3, "Containment Isolation Valves," and associated Bases Table 3.6.3-1, "Primary Containment Isolation Valves;" and are also normally maintained sealed closed during MODES 5, 6 and defueled as noted in Bases 3.6.3.

Miniflow Purge System

The miniflow purge system (see Figure 1) may be used (i.e., opened) during Modes 1, 2, 3, or 4 to reduce the concentration of noble gases within containment prior to and during personnel access and to equalize internal and external pressures as the valves used in the mini-purge system are designed to meet the requirements for automatic containment isolation valves. These functions are accomplished under administrative control as required by Technical Specifications. The miniflow purge system may also be used during shutdown conditions (i.e., Modes 5, 6, or defueled).

The following are the key attributes of the miniflow purge system:

a. The miniflow purge system consists of independent systems for Unit 1 and Unit 2 containments.

b. A supply fan induces air through outside air louvers, prefilters, medium efficiency filters, and distributes this air to the ductwork inside the containment via the normal purge.

c. Air is exhausted from the containment by the miniflow exhaust fan and is filtered through prefilters and HEPA filters prior to discharging to the atmosphere via the plant vent stack.

d. The containment penetrations are 8 inches maximum and each penetration contains two isolation valves, one inside and one outside the containment. The valves are designed to close in 5 seconds as a result of an ESF actuation signal.

e. Radiation monitors are provided to monitor the exhaust air prior to discharge to the plant vent stack.

f. Portions of the miniflow purge line can be used post-LOCA to provide a path from the containment to the post-LOCA purge unit. This portion of the system is Safety Category I. The portion of the system from the outside of the containment wall to the outside isolation valves is designed for the maximum containment pressure.

Normal Purge System

The Normal Purge System (see Figure 1) was originally designed to be used during shutdown conditions (i.e., MODES 5, 6, or defueled) to supply outside air into the containment for ventilation and cooling or heating, and to reduce the concentration of noble gases within containment prior to and during personnel access; however, the Normal Purge System is not normally used. The supply and exhaust lines each contain two isolation valves. Because of their large size, the 48 inch purge valves are not qualified for automatic closure from their open position under Design Basis Accident (DBA) conditions. Therefore, the 48 inch purge valves are required to be maintained sealed closed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained (see TS 3.6.3, "Containment Isolation Valves," and associated Bases Table 3.6.3-1, "Primary Containment Isolation Valves). The 48 inch purge valves are also normally maintained sealed closed during shutdown conditions (i.e., MODES 5, 6, or defueled) since the Normal Purge System is not normally used (see Bases 3.6.3, "Containment Isolation Valves").

The following are the key attributes of the normal purge system:

a. The normal containment purge system consists of independent systems for Unit 1 and Unit 2 containments.

b. Two 100% capacity, supply fans are provided. 100% outside air is induced through air louvers, prefilters, medium efficiency filters and distributed in the containment via ductwork around the periphery through linear grilles to create a fluid boundary, or air curtain, between access areas surrounding the pool and air with a potential tritium concentration. This purge rate provides approximately one air change per hour and will permit safe access to the containment 3 hours after a planned shutdown.

c. The exhaust air is induced from the containment by the purge exhaust fans. Suction is taken from the containment at the containment wall at elevation 463 feet 0 inches.

d. Two 100% capacity purge exhaust fans are provided. The purge exhaust is filtered through prefilters and HEPA filters prior to release to the atmosphere via the plant vent stack.

e. The normal purge supply and exhaust lines are 48-inches in diameter and are provided with two isolation valves each, one inside and one outside the containment.

f. Radiation monitors are provided to monitor exhaust air prior to discharge to the plant vent stack.

<u>Post-LOCA Purge System</u>

The combustible gas control system as defined by the requirements of 10 CFR 50.44, "Combustible gas control for nuclear power reactors," consists of a hydrogen monitoring system and a mixing system. Based on the revision to 10 CFR 50.44 which eliminated the design basis LOCA hydrogen release, the hydrogen recombiners and backup hydrogen vent and purge systems (i.e., post-LOCA purge system) are no longer required. Although the post-LOCA purge system is no longer required to meet the requirements of 10 CFR 50.44, the system remains in place and could be utilized following an accident.

<u>Containment Ventilation Isolation Instrumentation</u>

The containment ventilation isolation instrumentation closes the containment isolation valves in the minipurge, normal purge and post-LOCA systems. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident.

Containment ventilation isolation initiates on an automatic Safety Injection (SI) signal, by manual actuation of Phase A Isolation, by manual actuation of Phase B Isolation, or by a high radiation signal from containment area radiation monitors RE-AR011 or RE-AR012 (separate from the containment isolation signal).

These two area radiation monitoring channels (RE-AR011 and RE-AR012) provide input to the containment ventilation isolation. Each of the purge systems has inner and outer containment isolation valves in its supply and exhaust ducts. A high radiation signal from RE-AR011 initiates Train A containment ventilation isolation, which closes the inner containment isolation valves. A high radiation signal from RE-AR012 initiates Train B containment purge isolation, which closes the outer containment isolation valves.

Specifically, area radiation detectors RE-AR011 and RE-AR012 are interlocked with containment purge isolation valves VQ001A and B, and VQ002A and B, and containment mini-purge isolation valves VQ003, VQ004A and B, and VQ005A, B, and C. Upon detection of a high radiation level in containment, a containment ventilation isolation signal will be initiated and the above mentioned valves that are open will be closed. Again, it is noted that the containment ventilation isolation signal is separate from either the Phase A or Phase B containment isolation signal. The normal containment purge valves are locked closed by the administrative procedure of interrupting power to the valve at the circuit breaker (i.e., the circuit breaker is racked out (open)) and tagging the breaker "out of service". Valves VQ003, VQ004A/B, and VQ005A/B/C are equipped with an operator capable of closing the valves in 5 seconds for containment isolation.

If a routine containment purge is being conducted at the onset of an accident, the open containment mini-purge isolation valves will close upon receipt of an SI signal, backed-up by the containment high radiation signal from the containment area radiation monitors RE-AR011 and RE-AR012.

3.2 Licensing Basis

The containment ventilation isolation radiation monitors serve two primary functions, they:

a. act as backup to the Safety Injection (SI) signal to ensure closing of the purge valves; and

b. are the primary means for automatically isolating containment in the event of a fuel handling accident in containment.

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event (i.e., within approximately 60 seconds). The isolation of the purge valves has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed (i.e., the mini-purge system valves are capable of closing in 5 seconds as noted above, well within the assumed 60 seconds). The containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accident radiological doses are below 10 CFR 50.67, “Accident Source Term,” limits as noted in the current analysis of record (AOR).

In Modes 1 through 4, the Design Basis Accidents (DBAs) that result in a release of radioactive material within containment are a LOCA, Main Steam Line Break (MSLB), and Control Rod Ejection Accident (CREA). In the analyses for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analyses assume that the 48 inch purge valves are closed at event initiation as noted above.

In Mode 6 (i.e., Refueling), the containment ventilation isolation radiation monitors are also credited for containment isolation during a Fuel Handling Accident (FHA) involving RECENTLY IRRADIATED FUEL as described below in Section 3.4 below.

The requirement to close the containment purge and ventilation isolation valves upon receipt of a high radiation signal in containment comes from NUREG-0737, “Clarification of TMI [Three Mile Island] Action Plan Requirements," Item II.E.4.2, "Containment Isolation Dependability," Position 7, (Reference 2) which states:

> *"Containment purge and vent isolation valves must close on a high radiation signal."*

Additional guidance regarding the containment isolation system attributes is given in NUREG-0800, Section 6.2.4, "Containment Isolation System," (Reference 3) and NUREG-0800, Branch Technical Position (BTP) 6-4, "Containment Purging During Normal Plant Operations," (Reference 4). The 8-inch post-LOCA purge and miniflow purge valves meet the guidance of BTP 6-4.

Position 7 was required to be implemented by July 1, 1981; therefore, this requirement was part of the initial licensing basis for both Braidwood and Byron Stations. The original Technical Specification for Braidwood and Byron Stations were based on NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," Revision 4, Fall 1981, (Reference 5).

The recommended setpoint value in NUREG-0452 for the containment purge and exhaust isolation signal (for the containment area radiation monitors) was "(≤ 2 x background)," applicable in MODE 6 only. The recommended setpoint value for the purge and exhaust isolation signal for RCS leakage detection, applicable in MODES 1-4, (from the containment process radiation monitors) was also "(≤ 2 x background)." Conservatively, both Braidwood Station and Byron Station chose a setpoint "such that the actual submersion dose rate would not exceed 10 mR/hr in the containment building." Note that the setpoint was applicable to the RE-AR011/12 area monitors during MODES 1-4 and 6 as the high radiation containment ventilation isolation signal comes solely off the area radiation monitors (i.e., RE-AR011/12) at Braidwood and Byron Stations.

3.3 Operating History – Basis for Requested Change

The value of the current "Containment Radiation – High" trip setpoint for the containment area radiation monitors (i.e., RE-AR011/12) has been in effect since initial startup at both Braidwood and Byron Stations. This setpoint (i.e., "…actual submersion dose rate is ≤ 10 mR/hr in the Containment Building") has proven to be overly conservative and has caused an unnecessary operator burden due to the distraction associated with frequent trip setpoint changes, frequent alert alarms and high alarms with associated unwarranted containment isolation signals.

During a normal power ascension coming out of a refueling outage, the RE-AR011/12 setpoints need to be revised approximately every 10% power change to remain within the setpoint limit, as containment background radiation is proportional to power level. During routine power changes for issues such as Turbine Stop Valve surveillances or for maintenance activities, the RE-AR011/12 setpoint must be adjusted to stay in compliance with the TS setpoint limit. There have been a number of occasions where RE-AR011 and/or RE-AR012 had to be declared inoperable due to normal detector drift or a gradual change in containment radiation background level over the course of a fuel cycle prompting entry into a TS REQUIRED ACTION statement. Containment background radiation also changes with core burnup causing a potential setpoint non-compliance if the change in background is not closely monitored and the setpoint revised accordingly. In addition, due to normal radiation monitor fluctuations of approximately 10 mR/hr, frequent alert alarms and high alarms with associated containment isolation signals, have been received.

A number of recent events are documented below:

- Braidwood LER 2012-003-00, dated August 2, 2012, discusses five different occurrences where one or both of the containment radiation monitors were inoperable due to normal detector drift or not appropriately revising the trip setpoints during power changes.

- Byron LER 2012-003-00, dated August 3, 2012, discusses both 1RE-AR011 and 1RE-AR012 being inoperable and exceeding TS REQUIRED ACTION times due to not adjusting the trip setpoint during power ascension following the Unit 1 Spring 2011 refueling outage.

- Braidwood Condition Report (CR) 01373856, dated June 3, 2012, discusses the output of 1RE-AR011 drifting up over a number of days causing the setpoint to be in non-compliance with the TS.

- Braidwood CR 01417791, dated September 24, 2012, documents 10 ALERT alarms on 2RE-AR011 and 14 ALERT alarms on 2RE-AR012 in a 24-hour time period.

- Braidwood CR 01430870, dated October 24, 2012 discusses a spike on 1RE-AR012 causing a containment ventilation isolation signal.

- Byron CR 01374318 discusses the need to adjust the setpoints for 1/2RE-AR011/12 during the power ascension after an outage in order to maintain the radiation monitors in an OPERABLE status.

As can be seen, the frequency of these operational challenges presents an unwarranted burden and distraction to the operators. These challenges have not been attributed to instrumentation deficiencies.

3.4 Justification of Proposed Setpoint and Associated TS Changes

As previously noted, a nominal value for the containment background radiation level at 100% power (i.e., RTP) at all of the Braidwood Station or Byron Station units ranges from approximately 30 – 50 mR/hr. Therefore, the proposed "Containment Radiation – High" trip setpoint of 2x background at RTP will be in the range of approximately 60 – 100 mR/hr; however, may be higher or lower depending on actual containment conditions.

<u>Impact on Routine Containment Venting Evolutions at Power</u>

TS 3.6.4, "Containment Pressure," requires that "Containment pressure shall be $\geq$ -0.1 psig and $\leq$ +1.0 psig." Due to variations in internal and external containment atmospheric conditions, the containment pressure slowly increases (or can decrease) with time which requires periodic venting to remain within the TS upper pressure limit. Routine containment venting is performed in accordance with station procedures, (Braidwood) RP-BR-980, "Containment Vent and Mini Purge Gaseous Effluents," and (Byron) BCP 400-TCNMT/ROUTINE, "Gaseous Effluent

Release Form, Type: Routine Containment Release." Prior to venting, a containment gas sample is drawn and analyzed for radionuclide concentrations. Based on the radionuclide concentrations, an assumed conservative vent flow rate and predetermined venting time duration, a resultant dose at the site boundary is calculated. This calculation is done in accordance with the ODCM methodology that ensures compliance with 10 CFR 20 limits. The "Containment Radiation – High" trip setpoint is not used in this calculation and has no impact on this evolution and resultant release.

Based on this information, the proposed change to the "Containment Radiation – High" trip setpoint has no impact on the dose associated with routine containment venting at power.

Impact on Accident Analysis

This section will discuss the effect a change to the "Containment Radiation – High" trip setpoint may have on the following accidents:

- Loss of Coolant Accident (LOCA)
- Fuel Handling Accidents (FHA) in the Containment
- Control Rod Ejection Accident (CREA)

The accident analysis discussed below was performed at 3658.3 MWth, which is 102% of RTP. Since containment background radiation is proportional to power level, it is also appropriate and conservative, from a dose standpoint, to assume a high radiation containment ventilation isolation setpoint at RTP.

As noted above, in Section 3.2, "Licensing Basis," the containment ventilation isolation radiation monitors serve two primary functions, they:

a. act as backup to the SI signal to ensure closing of the purge valves; and

b. are the primary means for automatically isolating containment in the event of a fuel handling accident in containment.

The containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accident radiological doses are below 10 CFR 50.67, "Accident Source Term," limits as noted in the current analysis of record.

Braidwood Station and Byron Station received approval to use the Alternate Source Term (AST) methodology discussed in 10 CFR 50.67 in a letter from R. F. Kuntz (NRC) to C. M. Crane (EGC), "Byron Station, Unit Nos. 1 and 2, and Braidwood Station, Unit Nos. 1 and 2 – Issuance of Amendments Re: Alternative Source Term," dated September 8, 2006 (Reference 6).

Implementation of the AST methodology required analysis of the following DBAs:

- Loss of Coolant Accident (LOCA)
- Fuel Handling Accidents (FHA) in the Fuel Handling Building and in Containment
- Control Rod Ejection Accident (CREA)
- Locked Rotor Accident (LRA)
- Main Steam Line Break (MSLB)

- Steam Generator Tube Rupture (SGTR)

The dose results for these accidents are listed in Table 3, "Licensee Calculated DBA Radiological Consequences," in the NRC AST Safety Evaluation. All dose results met the regulatory limits specified in 10 CFR 50.67 and Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," (Reference 7).

Of these accidents, only the LOCA, FHA in containment, and CREA could potentially generate a high radiation condition in containment and be impacted by the proposed revision to the "Containment Radiation – High" trip setpoint. Note that the radiation release for the LRA is assumed to come from the Steam Generator PORVs; the worst case MSLB, from a dose perspective, assumes a break outside of containment; and the SGTR radiation release is from the SG PORVs, therefore, these accidents would not be impacted by a change to the "Containment Radiation – High" trip setpoint for the containment area radiation monitors and are not considered here.

The LOCA and CREA would potentially generate an SI signal with an associated containment high radiation condition. The SI signal would subsequently generate a containment ventilation isolation signal. As previously noted, the SI signal is backed up by the "Containment Radiation – High" signal which provides an independent containment ventilation isolation signal. Therefore, if the containment mini-purge system or post-LOCA system isolation valves were open at the onset of one of the subject accidents; and the SI signal failed to generate a containment ventilation isolation signal, the backup signal, (i.e., the "Containment Radiation – High" trip setpoint), would generate an independent signal to close the containment ventilation isolation valves. For this scenario, the value of the "Containment Radiation – High" trip setpoint needs to be assessed to determine if the proposed change to the setpoint has an impact on the accident analysis dose results.

As noted in the safety analysis assumptions discussed above in Section 3.2, the purge valves are assume to rapidly close early in the event (i.e., within 60 seconds) and consequently, the isolation time of the purge valves has not been analyzed mechanistically in the dose calculations. Again note that the containment mini-purge isolation valves are designed to close in 5 seconds, well within the assumed 60 seconds. Any dose that could be specifically attributed to containment isolation valve closure time is embodied in the associated accident dose results.

Although not specifically addressed in the accident analysis, if the "Containment Radiation – High" backup signal to the SI signal was needed to initiate a containment ventilation isolation signal, the additional time delay to sense a containment high radiation condition, at the proposed higher setpoint, would be inconsequential and have an insignificant impact on the accident dose results. It is reasonable to assume that a LOCA or CREA, which is large enough to depressurize the reactor coolant system and generate an SI signal, would produce a prompt containment high radiation condition which would be promptly sensed by the containment area radiation monitors. A sensitivity calculation was performed for the AR011/12 radiation monitors during initial startup that showed their response time is a function of radiation level; i.e., the higher the radiation level, the faster the monitors will respond, however, the monitors have a minimum response time of 0.6 sec. The data indicated that the minimum response time was obtained at radiation levels ≥ 280 mR/hr. The radiation level in containment immediately after a LOCA is projected to be well in excess of the proposed setpoint of " *≤ 2 x background in the*

Containment Building at RTP," and greater than 280 mR/hr; therefore, it is concluded that the monitors would sense a high radiation condition due to a LOCA in the minimum response time of approximately 0.6 seconds and initiate the containment ventilation isolation signal.

The accident dose contribution specifically due to a containment purge in progress was also previously evaluated. During the review of the Braidwood and Byron AST submittal in 2006, the NRC specifically inquired about the accident dose contribution attributed to a containment vent at power in a request for additional information. Exelon responded to this request in a letter from J. A. Bauer (EGC) to the NRC, "Response to NRC Request for Additional Information With Respect to Request for License Amendment Related to Application of Alternative Radiological Source Term," dated February 13, 2006 (Reference 8). In this letter, EGC stated that following:

> *The current licensing basis analysis does not consider purge doses. A technical evaluation of the purge dose contribution was performed assuming an unfiltered release path via the containment miniflow purge system for a period of seven seconds until isolation occurs. This evaluation used modified Alternative Source Term (AST) LOCA analysis RADTRAD files to determine doses due to containment purge. These modifications include an updated nuclide inventory file to reflect the maximum coolant isotopic inventory permitted by Technical Specifications (TS), a release fraction and timing file that reflects iodine and noble gas isotopes, as well as a single containment volume (assuming no spray), which is released at a rate of 3000 cfm for a period of seven seconds (five seconds for valve closure and two seconds for instrument response).*
>
> *The results of this evaluation indicated a dose of approximately 5.24E-04 rem TEDE* [i.e., 0.524 mrem] *for the CR, 1 .69E-04* [i.e., 0.169 mrem] *for the EAB, and 1 .42E-05* [i.e., 0.0142 mrem] *for the LPZ using this conservative seven second valve closure time from initiation of the event. None of these doses results in a significant change to the doses reported in the original submittal. These doses are consistent with those reported for Seabrook Station, a pressurized water reactor site comparable in size to Byron and Braidwood Stations, in Reference [9] (i.e., 3.84E-04 rem TEDE for the CR, 4.24E-04 for the EAB and 2 .06E-04 for the LPZ).*

As can be seen, the dose contribution attributed to a containment vent in this scenario has an insignificant impact on the dose results for the subject accidents shown in Table 3 of Reference 6. The "Containment Radiation – High" trip setpoint value was not specifically considered in this evaluation but is included in the two second instrument response time; therefore, the impact of raising the "Containment Radiation – High" trip setpoint to "*≤ 2 x background in the Containment Building at RTP*" would have no impact on the already insignificant dose contribution attributed to a containment purge during a LOCA.

For the FHA in containment, two cases are evaluated:

1. The first case evaluates the FHA which occurs ≥48 hours after shutdown. No filtration of the release or automatic isolation of the containment is assumed, as the containment equipment hatch is assumed to be open to the Fuel Handling Building (FHB). Essentially all of the radioactivity released from the damaged fuel assembly is assumed to reach the environment within two hours with acceptable radiological results. Since no containment ventilation isolation is assumed to occur, a revision to the "Containment Radiation – High" trip setpoint has no impact on the consequences of this accident.

2. The second FHA case evaluates a FHA for RECENTLY IRRADIATED FUEL (i.e., fuel that has occupied part of a critical reactor core within the previous 48 hours). The FHA is assumed to occur at a minimum of six hours after shutdown. As noted in Reference 6, the current AOR shows that acceptable dose results are obtained assuming that a) containment closure is established; and/or b) the Fuel Handling Building (FHB) ventilation system and Control Room Filtration system are operable according to TS requirements. These two sub-cases are addressed below:

    a. The first sub-case assumes that containment closure has been established as specified in TS 3.9.4, "Containment Penetrations." Note that "containment closure," currently defined in the TS Bases, means that all potential escape paths are filtered, closed, or capable of being closed. Accordingly, the associated LCO 3.9.4 requires that each penetration providing direct access from the containment atmosphere to the outside atmosphere be either: 1) closed by a manual or automatic isolation valve, blind flange, or equivalent; or 2) capable of being closed by an OPERABLE Containment Ventilation Isolation System. Should the mini-purge supply and exhaust valves be open at the onset of a FHA in containment, an automatic closure signal would be initiated by the "Containment Radiation – High" trip setpoint and promptly isolate the containment, thus, ending any effluent release. The current analysis does not assign a specific dose to this case as the containment is assumed to be promptly isolated (i.e., "containment closure" is obtained).

       Exelon has conservatively elected to remove the option of allowing movement of RECENTLY IRRADIATED FUEL with the containment purge valves open with an OPERABLE Containment Ventilation Isolation System, by deleting LCO 3.9.4.c.2 (as noted in Section 2.0 above). This deletion eliminates any potential release from the containment purge valves even though the valves are assumed to promptly close due to a high radiation signal. Therefore, the proposed change in the "Containment Radiation – High" trip setpoint has no impact on the dose results for this accident scenario as a Containment Ventilation Isolation is not required since all penetrations providing direct access from the containment atmosphere to the outside atmosphere will be either closed by a manual or automatic isolation valve, blind flange, or equivalent.

    b. The second sub-case assumes that the containment equipment hatch is open to the FHB during the FHA. Note that the containment becomes part of the FHB ventilation system envelope with the equipment hatch or associated personnel airlock open. In this case, it is assumed that the FHB ventilation system and the Control Room filtration system are OPERABLE according to the TS requirements. In the event of a FHA in the containment, the FHB ventilation system creates a negative pressure in the containment and FHB relative to the auxiliary building and outside atmosphere. The current analysis states that the negative pressure ensures that any radioactivity released to the containment atmosphere will either remain in the containment or be filtered through a FHB Ventilation System train.

       Exelon has, again, conservatively elected to remove the option of allowing movement of RECENTLY IRRADIATED FUEL with the containment purge valves open with an OPERABLE Containment Ventilation Isolation System, by deleting LCO 3.9.4.c.2 (as noted in Section 2.0 above). This deletion eliminates any potential release from the containment purge valves even though the valves are assumed to promptly close due to a high radiation signal and the FHB ventilation system is OPERABLE. Therefore, the proposed change in the "Containment Radiation – High" trip setpoint has no impact on the dose results for this accident scenario as a Containment Ventilation Isolation is not required since all penetrations providing direct access from the containment atmosphere to the outside atmosphere will be either closed by a manual or automatic isolation valve, blind flange, or equivalent.

Based on the above discussion, revising the "Containment Radiation – High" trip setpoint to *"≤ 2 x background in the Containment Building at RTP,"* is acceptable as the accident dose consequences for all previously analyzed events are insignificantly impacted and remain within regulatory limits.

As noted above, the deletion of LCO 3.9.4.c.2 is a conservative change in support of the newly proposed "Containment Radiation – High" trip setpoint. This deletion removes the option of moving RECENTLY IRRADIATED FUEL in the containment with the containment purge valves open with an OPERABLE Containment Ventilation Isolation System. Note that the ability to move RECENTLY IRRADIATED FUEL in the containment is still maintained; however, all penetrations providing direct access from the containment atmosphere to the outside atmosphere will be either closed by a manual or automatic isolation valve, blind flange, or equivalent.

The four other additional proposed TS changes are all associated with the deletion of LCO 3.9.4, Item c.2, and are being made for consistency (i.e., deleting a NOTE regarding MODE applicability, deleting a CONDITION related only to LCO 3.9.4.c.2, deleting a footnote regarding MODE applicability; and deleting two surveillances related to LCO 3.9.4.c.2).

<u>Summary of TS Changes Justification</u>

1. It is appropriate to delete the NOTE in TS 3.3.6 Condition B stating, *"Only applicable in MODE 1, 2, 3, or 4,"* as the only other applicable MODE for this LCO listed in the current TS Table 3.3.6-1, i.e., "*(a) When Item c.2 of LCO 3.9.4 is required,"* is being deleted; therefore, the NOTE is no longer needed or appropriate.

2. It is appropriate to delete TS 3.3.6 Condition C, which addresses inoperable radiation monitoring channels, as Condition C was only applicable when Item c.2 of LCO 3.9.4 is required; and Item c.2 of LCO 3.9.4 is being conservatively deleted.

3. It is appropriate to delete Footnote (a) of TS Table 3.3.6-1, "Containment Ventilation Isolation Instrumentation," which identifies a MODE of APPLICABILITY for the Containment Radiation-High FUNCTION as *" When Item c.2 of LCO 3.9.4 is required."* As noted above, Item c.2 of LCO 3.9.4 is being deleted.

4. Footnote (b) of TS Table 3.3.6-1, "Containment Ventilation Isolation Instrumentation," which specifies the "Containment Radiation – High" trip setpoint for two containment area radiation monitors (i.e., 1(2)RE-AR011 and 1(2) RE-AR012), is renumbered to Footnote (a) and revised to state: *"Trip setpoint shall be established at ≤ 2 x background in the Containment Building at RTP."* This change has no impact on at-power containment releases or the associated accident analysis dose consequences as discussed above.

5. LCO 3.9.4.c.2, which allow moving RECENTLY IRRADIATED FUEL with the containment purge valves open but capable of being closed by an OPERABLE Containment Ventilation Isolation System, is, conservatively deleted in support of the proposed setpoint change. This deletion assures containment closure and no longer relies on an automatic containment ventilation isolation during a FHA when moving RECENTLY IRRADIATED FUEL in the containment

6. It is appropriate to delete SR 3.9.4.2 and SR 3.9.4.3. SR 3.9.4.2 and SR 3.9.4.3 both address verifying operability of the containment purge valves in support of an OPERABLE Containment Ventilation Isolation System. Since LCO 3.9.4.c.2 is being deleted, operability of the Containment Ventilation Isolation System is not required in LCO 3.9.4; therefore, these SRs are no longer needed.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR [light water reactor] Edition," Section 6.2.4, "Containment Isolation System," establishes design criteria for containment isolation systems.

NUREG-0800, Branch Technical Position (BTP) 6-4, "Containment Purging During Normal Plant Operations," Position 1.e states that instrumentation and control systems provided to isolate the purge system lines should be independently actuated by diverse parameters; e.g., containment pressure, safety injection actuation, and containment radiation level.

NUREG-0737, "Clarification of TMI [Three Mile Island] Action Plan Requirements," November 1980, Item II.E.4.2, Containment Isolation Dependability," Position 7 states that containment purge and vent isolation valves must close on a high radiation signal. This isolation signal functions to automatically isolate any open ventilation isolation valves in the event containment radiation levels rise above anticipated levels, terminating the containment release.

The Braidwood Station and Byron Station, Units 1 and 2 designs ensure containment isolation dependability by satisfying the requirements of NUREG-0737. A high radiation signal, separate from the containment isolation signal, will close the containment purge and vent isolation valves. Area radiation detectors 1/2RE-AR011 and 1/2RE-AR012 are interlocked with containment mini-purge isolation valves 1/2VQ003, 1/2VQ004A and B, and 1/2VQ005A, B, and C. Upon detection of high radiation levels, a containment ventilation isolation signal will be initiated and the above mentioned valves that are open will be closed. The containment ventilation isolation signal is separate from either the Phase A or Phase B containment isolation signal. Valves 1/2VQ003, 1/2VQ004A/B, and 1/2VQ005A/B/C are equipped with an operator capable of closing the valves in 5 seconds for containment isolation. These 8-inch post-LOCA purge and mini-flow purge valves meet the guidance of BTP 6-4.

4.2 Precedents

Other licensees and NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," have containment high-radiation setpoints, which generate a containment ventilation isolation signal, similar to the proposed setpoint for Braidwood and Byron Stations.

NUREG-1431: ≤ 2x background

Shearon Harris: ≤ 3x background at RTP (MODES 1-4)
≤ 150 mR/hr during fuel movement

St. Lucie: ≤ 90 mR/hr

4.3 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit or early site permit," Exelon Generation Company, LLC, (EGC) requests amendments to Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, and Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2. Specifically, this amendment request proposes to revise Footnote (b) of Technical Specification (TS) Table 3.3.6-1, "Containment Ventilation Isolation Instrumentation," which specifies the "Containment Radiation – High" trip setpoint for two containment area radiation monitors (i.e., 1(2)RE-AR011 and 1(2)RE-AR012). Upon sensing a high radiation condition, these area radiation monitors provide an isolation signal to the containment normal purge, mini purge and post- LOCA (Loss of Coolant Accident) systems containment isolation valves. The proposed change would revise the "Containment Radiation – High" trip setpoint from the current, overly conservative value (i.e., a submersion dose rate of ≤ 10 mR/hr in the containment building), to ≤ 2 times the containment building background radiation reading at rated thermal power, consistent with NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change is needed to eliminate the unnecessary operator burden and distraction associated with frequent trip setpoint changes, frequent alert alarms and high alarms with associated unwarranted containment isolation signals.

In support of the proposed setpoint change, EGC also proposes to make a change to TS 3.9.4, "Containment Penetrations." LCO 3.9.4, Item c.2 currently allows the option of moving RECENTLY IRRADIATED FUEL with the containment purge valves open but capable of being closed by an OPERABLE Containment Ventilation Isolation System. This LCO would be deleted to eliminate any potential release from the containment purge valves as all penetrations providing direct access from the containment atmosphere to the outside atmosphere will now be closed by a manual or automatic isolation valve, blind flange, or equivalent during movement of RECENTLY IRRADIATED FUEL. Four other additional TS changes associated with the deletion of LCO 3.9.4, Item c.2, are proposed for consistency (i.e., deleting a NOTE regarding MODE applicability in TS 3.3.6 CONDITION B, deleting TS 3.3.6 CONDITION C related only to LCO 3.9.4.c.2, deleting Footnote (a) of Table 3.3.6-1 regarding MODE applicability; and deleting two surveillances (i.e., SR 3.9.4.2 and SR 3.9.4.3) related to LCO 3.9.4.c.2).

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

EGC has evaluated the proposed change for Braidwood Station and Byron Station, using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

Criteria

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The containment ventilation isolation radiation monitors serve two primary functions, they:

   a. act as backup to the SI signal to ensure closing of the purge valves; and

   b. are the primary means for automatically isolating containment in the event of a fuel handling accident in containment.

   Upon sensing a high radiation condition in containment, these area radiation monitors provide an isolation signal to the containment normal purge, mini purge and post- LOCA systems containment isolation valves (i.e., a containment ventilation isolation signal).

   The accidents that could potentially be impacted by the proposed change were evaluated; specifically the Loss of Coolant Accident (LOCA), Control Rod Ejection Accident (CREA) and Fuel Handling Accident (FHA) in Containment. The proposed change has no impact on probability of these accidents occurring as the subject containment radiation area monitors detect a high radiation condition resulting from these accidents. The radiation monitors do not initiate any accidents or transients. Changing the "Containment Radiation – High" trip setpoint from "≤ 10 mR/hr in the containment building," to "≤ 2 times the containment building background radiation reading at rated thermal power" only affects the point (i.e., the radiation level in containment) at which a containment ventilation isolation signal would be generated. The requested change does not involve any physical plant modifications or operational changes that could adversely affect system reliability or performance of the radiation monitors, or that could affect the probability of operator error. The requested change does not affect any postulated accident precursors and therefore, will not affect the probability of an accident previously evaluated.

The proposed change was evaluated to determine the impact on the dose consequences of the LOCA, CREA, or FHA in containment. The evaluation assumed a containment purge was in progress at the onset of the subject accidents and showed that the proposed change in the containment radiation monitors' setpoint had no effect on the purge valve isolation time. With regard to the LOCA and CREA, the safety analysis assumes a prompt purge valve isolation time (i.e., approximately 60 seconds) that significantly bounds the radiation monitor sensing and response time, and actual valve design closure time (i.e., a total of approximately 7 seconds). The radiation monitor setpoint is not considered in the safety analysis and any dose contribution associated with the containment purge, due to the proposed change in setpoint, was shown to be unaffected; therefore, the proposed change has no impact on the already insignificant dose contribution attributed to a containment purge during an accident of less than one mrem.

The dose consequences associated with the FHA in containment are also not impacted by the proposed change in containment radiation monitor setpoint. The existing dose consequences resulting from a FHA with moving non-RECENTLY IRRADIATED FUEL (i.e., fuel moved more than 48 hours after reactor shutdown) conservatively assume the containment purge valves remain open throughout the event; therefore, a change in the isolation setpoint does not impact the results of this analysis. With regard to movement of RECENTLY IRRADIATED FUEL (i.e., fuel moved less then 48 hours after reactor shutdown), EGC's proposal deletes TS LCO 3.9.4.c.2 which allowed the containment purge valves to be open provided the containment radiation isolation system is OPERABLE. Deletion of TS LCO 3.9.4.c.2 ensures that the containment purge valves are in the closed position when moving RECENTLY IRRADIATED FUEL, thus removing dependence on the containment radiation isolation system and associated radiation monitor setpoint from the FHA dose consequences.

The four other additional TS changes associated with the deletion of LCO 3.9.4, Item c.2, proposed for consistency (i.e., deleting a NOTE regarding MODE applicability, deleting a CONDITION related only to LCO 3.9.4.c.2, deleting a footnote regarding MODE applicability; and deleting two surveillances related to LCO 3.9.4.c.2), also have no affect on either the probability or consequences of an accident previously evaluated.

Based on the above discussion, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed changes do not result in a change to the design of the Containment Ventilation Isolation System or the manner in which the system operates or provides plant protection. The containment radiation monitors will sense radiation levels in the same way and will respond in the same manner when the setpoint is exceeded. The change in the "Containment Radiation – High" setpoint does not create a new failure mode for the associated containment radiation monitors or for any other plant equipment. The deletion of LCO 3.9.4, Item c.2, in support of the setpoint change during refueling operations, is more conservative than the current allowances and actually eliminates a potential failure mode for the assumed open containment ventilation isolation valves as the proposed deletion of LCO 3.9.4, Item c.2 would require the valves to be closed prior to moving RECENTLY IRRADIATED FUEL.

The changes do not result in the creation of any new accident precursors, the creation of any changes to the existing accident scenarios, nor do they create any new or different accident scenarios. Subsequently, the accidents defined in the UFSAR continue to represent the credible spectrum of events to be analyzed which determine safe plant operation.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The analysis methodologies used in the subject safety analyses are not modified as a result of the proposed TS changes to the "Containment Radiation – High" trip setpoint or the deletion of LCO 3.9.4, Item c.2, or any of the other four associated TS changes. Although the "Containment Radiation – High" trip setpoint is being increased, the increase in response time to a high radiation condition in containment, when compared to the current setpoint, is negligible due to the projected prompt rise in containment radiation level upon initiation of a LOCA. The dose consequences and resultant margin of safety to the regulatory acceptance limits, due to revising the "Containment Radiation – High" setpoint to ≤ 2 times the containment building background radiation reading at rated thermal power, was shown to be unaffected for normal at-power containment releases; have a negligible impact on the associated LOCA and CREA accident dose consequences; and have no impact on the FHA when moving RECENTLY IRRADIATED FUEL. Therefore, the proposed changes do not impact any analysis margins.

The proposed changes do not alter the manner in which the safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The current safety analyses remain bounding since their conclusions are not affected by the proposed changes. The safety systems credited in the safety analyses will continue to be available to perform their mitigation functions. All protection signals credited as the primary or secondary accident mitigating functions, and all operator actions credited in the accident analyses remain the same. The proposed changes will not result in plant operation in a configuration outside the design basis.

Based on the above information, the proposed change does not result in a significant reduction in the margin of safety.

Based on the above evaluation, EGC concludes that the proposed amendments do not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92, paragraph (c), and, accordingly, a finding of no significant hazards consideration is justified.

4.4 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by the operation of Units 1 and 2 at Byron Station and Braidwood Station in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

EGC has evaluated this proposed operating license amendment consistent with the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments." EGC has determined that these proposed changes meet the criteria for a categorical exclusion set forth in paragraph (c)(9) of 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," and as such, has determined that no irreversible consequences exist in accordance with paragraph (b) of 10 CFR 50.92, "Issuance of amendment." This determination is based on the fact that these changes are being proposed as an amendment to the license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or which changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

(i) The amendment involves no significant hazards consideration.

As demonstrated in Section 4.3, "No Significant Hazards Consideration," the proposed change does not involve any significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The proposed change does not result in an increase in power level, does not increase the production nor alter the flow path or method of disposal of radioactive waste or byproducts. It is expected that all plant equipment would operate as designed in the event of an accident to minimize the potential for any leakage of radioactive effluents. The effluent releases potentially affected by the subject changes, during normal at-power operations and during the appropriate accident scenarios, were specifically reviewed. These changes had an insignificant impact on all scenarios, and thus, there will be no significant change in the amounts of radiological effluents released offsite.

Based on the above evaluation, the proposed change will not result in a significant change in the types or significant increase in the amounts of any effluent released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

There is no change in individual or cumulative occupational radiation exposure due to the proposed change. Specifically, the change to the "Containment Radiation – High" trip setpoint has no impact on the containment radiation levels or the normal operational containment effluent releases controlled by procedure. The proposed action will not change the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposed action result in any change in the normal radiation levels within the plant.

Therefore, in accordance with 10 CFR 51.22, paragraph (b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. NUREG-1431, "Standard Technical Specifications, Westinghouse Plants"

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

3. NUREG-0800, Section 6.2.4, Containment Isolation System"

4. NUREG-0800, Branch Technical Position 6-4, "Containment Purging During Normal Plant Operations"

5. NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," Revision 4, Fall 1981

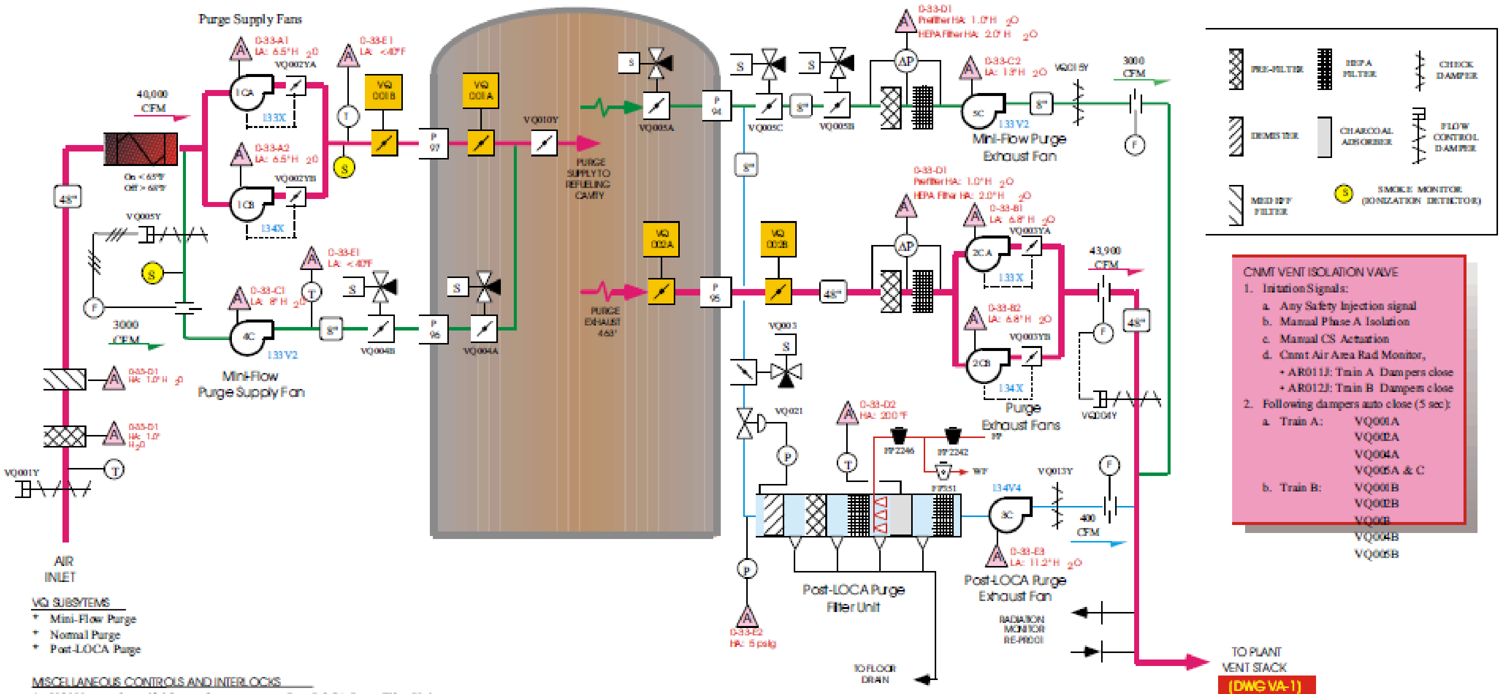
6. Letter from R. F. Kuntz (NRC) to C. M. Crane (EGC), "Byron Station, Unit Nos. 1 and 2, and Braidwood Station, Unit Nos. 1 and 2 – Issuance of Amendments Re: Alternative Source Term," dated September 8, 2006

7. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

8. Letter from J. A. Bauer (EGC) to the NRC, "Response to NRC Request for Additional Information With Respect to Request for License Amendment Related to Application of Alternative Radiological Source Term," dated February 13, 2006

9. Letter from M. E. Warner (FPL Energy) to NRC, "Seabrook Station License Amendment Request 03-02, 'Implementation of Alternate Source Term,'" dated October 6, 2003

Figure 1
Containment Purge System



VQ SUBSYTEMS

* Mini-Flow Purge
* Normal Purge
* Post-LOCA Purge

MISCELLANEOUS CONTROLS AND INTERLOCKS

1. VQ003 auto closes if deluge valves are open to Post-LOCA Purge Filter Unit
2. Post-LOCA Purge Filter Unit flow control modulates VQ021 to maintain pressure of 14.9 psia at filter inlet

MINI-FLOW PURGE

1. Functions:
   a. Cnmt pressure control during normal reactor operations
   b. Cnmt fission product activity control
2. Cnmt isolation dampers - 8 inches diameter
3. Per ITS SR 3.6.3.2, Cnmt purge (supply or exhaust) valves may be opened to purge or vent Cnmt under administrative controls.

NORMAL PURGE

1. Function: Purge Cnmt with outside air to ensure continuous access to Cnmt during planned reactor shutdown periods.
2. Cnmt isolation dampers - 48 inches diameter
3. Per ITS SR 3.6.3.1, dampers must be sealed closed in Modes 1-4
4. One complete air change per hour (Cnmt 2.8 E6 ft^3)
5. Safe access to Cnmt within 3 hours after planned reactor S/D (Modes 5/6)

POST-LOCA PURGE

1. Function: Backup to hydrogen recombiners to purge the Cnmt Atmosphere of hydrogen to maintain H_2 < 4% by volume following LOCA.
2. Does not operate during normal plant operating conditions.

EPN	FAN	STARTING INTERLOCKS	AUTOMATIC TRIPS
1VQ004C	Mini-Flow Purge Supply Fan	1. Cnmt Isolation Dampers (VQ004A & B) full open 2. Suction Damper (VQ001Y) full open	1. Starting interlocks not met 2. Overcurrent
1VQ005C	Mini-Flow Purge Exhaust Fan	1. Cnmt Isolation Dampers (VQ005A, B, & C) full open	1. Starting interlocks not met 2. Overcurrent
1VQ001CA/B	Normal Purge Supply Fan	1. Suction Damper (VQ001Y) full open 2. Cnmt Isolation Dampers (VQ001A & B) full open 3. Parallel fan's discharge damper (VQ002YA or B) full close	1. Starting interlocks not met 2. Inlet air temperature < 40° F 3. Smoke detected in air supply 4. Overcurrent
1VQ002CA/B	Normal Purge Exhaust Fan	1. Supply dampers full open (VQ002A & VQ002B) 2. Parallel fan's discharge damper (VQ003YA or B) full close 3. Either supply fan running	1. Starting interlocks not met 2. Overcurrent
1VQ003C	Post-LOCA Purge Fan	1. Cnmt Isolation Damper (VQ005A) full open 2. Cnmt Isolation Damper (VQ003) full open	1. Starting interlocks not met 2. Overcurrent

SOURCES OF HYDROGEN

1. Radiolytic Decomposition
   $2H_2O \rightarrow 2H_2 + O_2$ (Major Source)
2. Zinc Water Reaction
   $Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 + HEAT$
3. Hydrogen Release From RCS
4. Plant Material Corrosion
   Aluminum: $2Al + 3H_2O \rightarrow Al_2O_3 + 3H_2$
   Zinc: $Zn + 2H_2O \rightarrow Zn(OH)_2 + H_2$

UFSAR: 9.4.9 (Cnmt Purge System)

VP-2, CONTAINMENT PURGE
JUL 28, 1999, REV. 5
FOR TRAINING USE ONLY

ATTACHMENT 2
Markup of Technical Specifications Pages for Braidwood Station, Units 1 and 2

Braidwood Station, Units 1 and 2

Facility Operating License Nos. NPF-72 and NPF-77

AFFECTED TECHNICAL SPECIFICATIONS PAGES

3.3.6-2
3.3.6-5
3.9.4-1
3.9.4-2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. ~~------NOTE------~~ ~~Only applicable in MODE 1, 2, 3, or 4.~~ ~~--------------------~~ One or more automatic actuation trains inoperable. OR Two radiation monitoring channels inoperable. OR Required Action and associated Completion Time of Condition A not met.	B.1 Enter applicable Conditions and Required Actions of LCO 3.6.3, "Containment Isolation Valves," for containment purge valves made inoperable by isolation instrumentation.	Immediately
~~C. ------NOTE------~~ ~~Only applicable when Item c.2 of LCO 3.9.4 is required.~~ ~~--------------------~~ ~~Two radiation monitoring channels inoperable.~~ ~~OR~~ ~~Required Action and associated Completion Time of Condition A not met.~~	~~C.1 Place and maintain containment purge valves in the closed position.~~ ~~OR~~ ~~C.2 Enter applicable Conditions and Required Actions of LCO 3.9.4, "Containment Penetrations," for containment purge valves made inoperable by isolation instrumentation.~~	~~Immediately~~ ~~Immediately~~

Table 3.3.6-1 (page 1 of 1)
Containment Ventilation Isolation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Manual Initiation - Phase A	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a.1, for all initiation functions and requirements.			
2.	Manual Initiation - Phase B	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.b.1, for all initiation functions and requirements.			
3.	Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5	NA
4.	Containment Radiation-High	1,2,3,4,~~(a)~~	2	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.6	~~(b)~~ (a)
5.	Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.			

~~(a) When Item c.2 of LCO 3.9.4 is required.~~

~~(b) Trip setpoint shall be established such that actual submersion dose rate is ≤ 10 mR/hr in the Containment Building. The trip setpoint may be increased above this value in accordance with the methodology established in the Offsite Dose Calculation Manual.~~

(a) Trip setpoint shall be established at ≤ 2 x background in the Containment Building at RTP.

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. One door in the personnel air lock closed and the equipment hatch held in place by ≥ 4 bolts;

b. One door in the emergency air lock closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere ~~either:~~

~~1.~~ ~~c~~Closed by a manual or automatic isolation valve, blind flange, or equivalent~~, or~~ .

~~2. Capable of being closed by an OPERABLE Containment Ventilation Isolation System.~~

---NOTE---

LCO 3.9.4.a is not required to be met when in compliance with LCO 3.7.13, "Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System," or its associated Conditions and Required Actions.

APPLICABILITY: During movement of RECENTLY IRRADIATED FUEL assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend movement of RECENTLY IRRADIATED FUEL assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	In accordance with the Surveillance Frequency Control Program
~~SR 3.9.4.2~~	~~Verify each required containment purge valve actuates to the isolation position on an actual or simulated actuation signal.~~	~~In accordance with the Surveillance Frequency Control Program~~
~~SR 3.9.4.3~~	~~Verify the isolation time of each required containment purge valve is within limits.~~	~~In accordance with the Inservice Testing Program~~

ATTACHMENT 3
Markup of Technical Specifications Pages for Byron Station, Units 1 and 2

Byron Station, Units 1 and 2

Facility Operating License Nos. NPF-37 and NPF-66

<u>AFFECTED TECHNICAL SPECIFICATIONS PAGES</u>

3.3.6-2
3.3.6-5
3.9.4-1
3.9.4-2

ACTIONS (continued)

CONDITION		REQUIRED ACTION		COMPLETION TIME
B.	~~-----------NOTE-----------~~ ~~Only applicable in MODE 1, 2, 3, or 4.~~ ~~---------------------------~~ One or more automatic actuation trains inoperable. OR Two radiation monitoring channels inoperable. OR Required Action and associated Completion Time of Condition A not met.	B.1	Enter applicable Conditions and Required Actions of LCO 3.6.3, "Containment Isolation Valves," for containment purge valves made inoperable by isolation instrumentation.	Immediately
~~C.~~	~~-----------NOTE-----------~~ ~~Only applicable when Item c.2 of LCO 3.9.4 is required.~~ ~~---------------------------~~ ~~Two radiation monitoring channels inoperable.~~ ~~OR~~ ~~Required Action and associated Completion Time of Condition A not met.~~	~~C.1~~ ~~OR~~ ~~C.2~~	~~Place and maintain containment purge valves in the closed position.~~ ~~Enter applicable Conditions and Required Actions of LCO 3.9.4, "Containment Penetrations," for containment purge valves made inoperable by isolation instrumentation.~~	~~Immediately~~ ~~Immediately~~

Table 3.3.6-1 (page 1 of 1)
Containment Ventilation Isolation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Manual Initiation - Phase A	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a.1, for all initiation functions and requirements.			
2.	Manual Initiation - Phase B	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.b.1, for all initiation functions and requirements.			
3.	Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5	NA
4.	Containment Radiation-High	1,2,3,4,~~(a)~~	2	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.6	~~(b)~~ (a)
5.	Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.			

~~(a) When Item c.2 of LCO 3.9.4 is required.~~

~~(b) Trip setpoint shall be established such that actual submersion dose rate is ≤ 10 mR/hr in the Containment Building. The trip setpoint may be increased above this value in accordance with the methodology established in the Offsite Dose Calculation Manual.~~

(a) Trip setpoint shall be established at ≤ 2 x background in the Containment Building at RTP.

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. One door in the personnel air lock closed and the equipment hatch held in place by ≥ 4 bolts;

b. One door in the emergency air lock closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere ~~either:~~

~~1.~~ Closed by a manual or automatic isolation valve, blind flange, or equivalent~~, or~~ .

~~2. Capable of being closed by an OPERABLE Containment Ventilation Isolation System.~~

----------------------------NOTE----------------------------

LCO 3.9.4.a is not required to be met when in compliance with LCO 3.7.13, "Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System," or its associated Conditions and Required Actions.

APPLICABILITY: During movement of RECENTLY IRRADIATED FUEL assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend movement of RECENTLY IRRADIATED FUEL assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	In accordance with the Surveillance Frequency Control Program
~~SR 3.9.4.2~~	~~Verify each required containment purge valve actuates to the isolation position on an actual or simulated actuation signal.~~	~~In accordance with the Surveillance Frequency Control Program~~
~~SR 3.9.4.3~~	~~Verify the isolation time of each required containment purge valve is within limits.~~	~~In accordance with the Surveillance Frequency Control Program~~

ATTACHMENT 4
Markup of Technical Specifications Bases Pages for Braidwood Station, Units 1 and 2

Braidwood Station, Units 1 and 2

Facility Operating License Nos. NPF-72 and NPF-77

AFFECTED TECHNICAL SPECIFICATIONS BASES PAGES
(NOTE: TS Bases pages are provided for information only.)

B 3.3.6-1
B 3.3.6-2
B 3.3.6-4
B 3.3.6-5
B 3.3.6-6
B 3.9.4-3
B 3.9.4-4
B 3.9.4-5
B 3.9.4-6
B 3.9.4-7

B 3.3 INSTRUMENTATION

B 3.3.6 Containment Ventilation Isolation Instrumentation

BASES

BACKGROUND

Containment ventilation isolation instrumentation closes the containment isolation valves in the Minipurge System and the Normal Purge System. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. A discussion of the containment ventilation system is provided in the Bases for LCO 3.6.3, "Containment Isolation Valves."

Containment ventilation isolation initiates on an automatic Safety Injection (SI) signal, by manual actuation of Phase A Isolation, by manual actuation of Phase B Isolation, or by a high radiation signal from RE-AR011 or RE-AR012. The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss the ESFAS modes of initiation.

Two radiation monitoring channels (RE-AR011 and RE-AR012) provide input to the containment ventilation isolation. Each of the purge systems has inner and outer containment isolation valves in its supply and exhaust ducts. A high radiation signal from RE-AR011 initiates Train A containment ventilation isolation, which closes the inner containment isolation valves. A high radiation signal from RE-AR012 initiates Train B containment purge isolation, which closes the outer containment isolation valves.

The trip setpoint is established such that the actual ~~submersion~~ dose rate would not exceed ~~10 mR/hr~~ two times background at Rated Thermal Power (RTP) in the containment building. ~~The setpoint value may be increased up to twice the maximum concentration activity in containment determined by the~~ A sample analysis is performed prior to each release ~~provided~~ to ensure the ~~value~~ resultant dose does not exceed ~~10% of~~ the limits determined by the Offsite Dose Calculation Manual.

BASES

APPLICABLE SAFETY ANALYSES	The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event (i.e., within approximately 60 seconds). The isolation of the purge valves has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The containment ventilation isolation radiation monitors act as backup to the SI signal to ensure closing of the purge valves. ~~They are also the primary means for automatically isolating containment in the event of a fuel handling accident.~~ Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental radiological doses are below 10 CFR 50.67, “Accident Source Term,” (Ref. 1) limits. The containment ventilation isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The LCO requirements ensure that the instrumentation necessary to initiate Containment Ventilation Isolation, listed in Table 3.3.6-1, is OPERABLE. 1. Manual Initiation - Phase A Refer to LCO 3.3.2, Function 3.a.1, for all initiating Functions and requirements. 2. Manual Initiation - Phase B Refer to LCO 3.3.2, Function 3.b.1, for all initiating Functions and requirements.

BASES

APPLICABILITY

The Containment Ventilation Isolation Functions must be OPERABLE in the MODES ~~or other specified conditions~~ identified in Table 3.3.6-1. Under these conditions, the potential exists for an accident that could release fission product radioactivity into containment. Therefore, the containment ventilation isolation instrumentation must be OPERABLE in these MODES.

While in MODES 5 and 6 without fuel handling in progress, ~~or with a penetration closed by a manual or automatic isolation valve, blind flange, or equivalent,~~ the containment ventilation isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1. If fuel handling is in progress for RECENTLY IRRADIATED FUEL, the containment ventilation isolation instrumentation need not be OPERABLE as each penetration providing access from the containment atmosphere to the outside atmosphere must be closed by a manual or automatic isolation valve, blind flange, or equivalent in accordance with LCO 3.9.4.c.

The Applicability for the containment ventilation isolation on the ESFAS Manual Initiation-Phase A, Manual Initiation-Phase B, and Safety Injection Functions are specified in LCO 3.3.2. Refer to the Bases for LCO 3.3.2 for discussion of the Manual Initiation-Phase A, Manual Initiation-Phase B, and Safety Injection Functions Applicabilities. The Applicability for the containment ventilation isolation Automatic Actuation Logic and Actuation Relays is MODES 1, 2, 3, and 4. The Applicability for the containment ventilation isolation on Containment Radiation - High is MODES 1, 2, 3, 4. ~~, and when Item C.2 of LCO 3.9.4 is required.~~

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by plant specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

BASES

ACTIONS (continued)

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for the Automatic Actuation Logic and Actuation Relays Function and the Containment Radiation - High Function listed in Table 3.3.6-1. The Completion Time(s) of the inoperable channel(s)/train(s) of the given Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the failure of one containment ventilation isolation radiation monitor channel. Condition A requires the inoperable channel to be restored to OPERABLE status within 4 hours. The Completion Time is justified by the low likelihood of events occurring during this interval, and recognition that the remaining channel will respond to most events.

B.1

Condition B addresses the train orientation of the Solid State Protection System (SSPS) and the master and slave relays for the Containment Ventilation Isolation Function. It also addresses the failure of both radiation monitoring channels or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

If one or both automatic actuation trains are inoperable, both radiation monitoring channels are inoperable, or the Required Action and associated Completion Time of Condition A is not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation.

~~Condition B is modified by a Note stating that the Condition is only applicable in MODE 1, 2, 3, or 4.~~

BASES

ACTIONS (continued)

~~C.1 and C.2~~

~~Condition C addresses the failure of both radiation monitoring channels or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1. If both channels are inoperable or the Required Action and associated Completion Time of Condition A is not met, operation may continue as long as the Required Action to place and maintain containment purge valves in their closed position is met or the applicable Conditions of LCO 3.9.4, "Containment Penetrations," are met for each valve made inoperable by failure of isolation instrumentation. The Completion Time for these Required Actions is immediately.~~

~~A Note states that Condition C is only applicable when Item C.2 of LCO 3.9.4 is required.~~

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which Containment Ventilation Isolation Functions.

SR 3.3.6.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

BASES

BACKGROUND (continued)

The Containment Ventilation Isolation System consists of the normal purge subsystem, the mini purge subsystem, and the post Loss Of Coolant Accident purge subsystem. These three subsystems contain penetrations which provide direct access from the containment to the outside atmosphere. In MODE 6, the minipurge subsystem is normally used to exchange large volumes of containment air to support refueling operations. Each penetration contains inside and outside containment isolation valves which close automatically on an actuation signal~~.~~; however, d~~D~~uring movement of RECENTLY IRRADIATED FUEL within containment, each penetration providing direct access from the containment atmosphere to the outside atmosphere will be closed by a manual or automatic isolation valve, blind flange, or equivalent in accordance with LCO 3.9.4.c. The Containment Ventilation Isolation System is not relied on to provide automatic containment closure and; therefore, need not be OPERABLE.~~all required valves within a subsystem must be capable of being closed by a containment ventilation isolation signal whenever the associated subsystem is in operation.~~ A list of the instrumentation which functions to isolate the valves in these penetrations is provided in LCO 3.3.6, "Containment Ventilation Isolation Instrumentation."

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by a closed automatic isolation valve, a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods allowed under the provisions of 10 CFR 50.59 may include use of a material that can provide a temporary atmospheric pressure ventilation barrier during movement of RECENTLY IRRADIATED FUEL within the containment.

BASES

APPLICABLE SAFETY ANALYSES

During movement of RECENTLY IRRADIATED FUEL assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents, analyzed in Reference 2, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," ensure that the release of fission product radioactivity, subsequent to a fuel handling accident in containment, results in doses that are within the 10 CFR 50.67 (Ref. 5) limits. The radiological dose assessments for the Design Basis Fuel Handling Accident in containment were performed in accordance with the guidance of Regulatory Guide 1.183 (Ref 4).

When moving RECENTLY IRRADIATED FUEL in containment, the requirements of the LCO must be met with the exception that LCO 3.9.4.a need not be met if at least one train of the FHB Ventilation System is OPERABLE as specified in the Note. This exception permits movement of RECENTLY IRRADIATED FUEL with both personnel air lock doors open or the equipment hatch not intact.

When moving fuel in the containment that is not RECENTLY IRRADIATED FUEL, the LCO is not applicable. Due to radioactive decay, neither containment closure nor an OPERABLE FHB Ventilation System train are required to meet the dose limits of 10 CFR 50.67 during a fuel handling accident.

Another consideration, which may result in a limiting decay time prior to fuel handling, is the impact of decay heat on the spent fuel pool cooling requirements described in Reference 3.

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO limits the consequences of a fuel handling accident involving handling RECENTLY IRRADIATED FUEL in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed. ~~except for the OPERABLE containment purge (supply and exhaust) penetrations. For the OPERABLE containment purge~~

BASES

LCO (continued)

~~penetrations, this LCO ensures that unisolated penetrations are isolable by the Containment Ventilation Isolation System.~~ The ~~OPERABILITY~~ requirements for this LCO ensure that containment closure is ~~the automatic purge valve closure times specified in the UFSAR can be~~ achieved and, therefore, meet the assumptions used in the safety analysis to ensure ~~that releases through the valves are terminated, such~~ that radiological doses are within the acceptance limits.

The LCO is modified by a Note which allows both personnel air lock doors to be open or the equipment hatch not intact when the FHB Ventilation System is in compliance with LCO 3.7.13 or its associated Conditions and Required Actions. When the equipment hatch is installed it serves to contain fission product radioactivity that may be released following a fuel handling accident in the containment. When the equipment hatch is not intact, or when both doors of the personnel air lock are simultaneously opened, the internal containment pressure is essentially equal to the internal pressure of the fuel handling building. In the event of a fuel handling accident in the containment, realigning of the fuel handling building ventilation system creates a negative pressure in the containment and fuel handling building relative to the auxiliary building and outside atmosphere. The negative pressure ensures that any radioactivity released to the containment atmosphere will either remain in the containment or be filtered through a FHB Ventilation System train. As such, with the equipment hatch not intact, or with both personnel air lock doors open, the consequences of a fuel handling accident involving RECENTLY IRRADIATED FUEL in containment would not exceed those calculated for a fuel handling accident involving RECENTLY IRRADIATED FUEL in the fuel handling building.

In addition, a commitment has been made to implement compensatory measures during movement of irradiated fuel as described in UFSAR Section 15.7.4, “Fuel Handling Accidents.” These compensatory measures support the Alternate Source Term methodology and reduce doses even further below that provided by natural decay and avoid unmonitored releases in the event of a postulated fuel handling accident.

BASES

APPLICABILITY

The containment penetration requirements are applicable during movement of RECENTLY IRRADIATED FUEL assemblies within containment because this is when there is a potential for a limiting fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODE 5, and in MODE 6 when movement of RECENTLY IRRADIATED FUEL assemblies within containment are not being conducted, the potential for a limiting fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1

If the containment equipment hatch, air lock doors, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, the unit must be placed in a condition where containment closure is not needed. This is accomplished by immediately suspending movement of RECENTLY IRRADIATED FUEL assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that each of the containment penetrations required to be isolated is isolated. ~~This Surveillance for the open purge valves demonstrates that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power which will ensure that each valve is capable of being closed by an OPERABLE automatic Containment Ventilation Isolation signal.~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~SR 3.9.4.2~~

~~This Surveillance demonstrates that each required containment purge valve actuates to its isolation position on an actual or simulated high radiation signal. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. SR 3.9.4.3 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.~~

~~SR 3.9.4.3~~

~~This Surveillance demonstrates that the isolation time of each required containment purge valve providing direct access from the containment atmosphere to the outside atmosphere is in accordance with the Inservice Testing Program requirements. This SR, along with SR 3.9.4.2, ensures the containment purge valves in penetrations which provide direct access from the containment atmosphere to the outside atmosphere are capable of closing after a postulated fuel handling accident to limit the release of fission product radioactivity from the containment.~~

REFERENCES

1. UFSAR, Section 15.7.4.
2. NUREG-0800, Section 15.0.1, Revision 0, July 2000.
3. NUREG-0800, Section 9.1.3.
4. Regulatory Guide 1.183, July 2000.
5. 10 CFR 50.67.

ATTACHMENT 5
Markup of Technical Specifications Bases Pages for Byron Station, Units 1 and 2

Byron Station, Units 1 and 2

Facility Operating License Nos. NPF-37 and NPF-66

AFFECTED TECHNICAL SPECIFICATIONS BASES PAGES
(NOTE: TS Bases pages are provided for information only.)

B 3.3.6-1
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B 3.3 INSTRUMENTATION

B 3.3.6 Containment Ventilation Isolation Instrumentation

BASES

BACKGROUND

Containment ventilation isolation instrumentation closes the containment isolation valves in the Minipurge System and the Normal Purge System. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. A discussion of the containment ventilation system is provided in the Bases for LCO 3.6.3, "Containment Isolation Valves."

Containment ventilation isolation initiates on an automatic Safety Injection (SI) signal, by manual actuation of Phase A Isolation, by manual actuation of Phase B Isolation, or by a high radiation signal from RE-AR011 or RE-AR012. The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss the ESFAS modes of initiation.

Two radiation monitoring channels (RE-AR011 and RE-AR012) provide input to the containment ventilation isolation. Each of the purge systems has inner and outer containment isolation valves in its supply and exhaust ducts. A high radiation signal from RE-AR011 initiates Train A containment ventilation isolation, which closes the inner containment isolation valves. A high radiation signal from RE-AR012 initiates Train B containment purge isolation, which closes the outer containment isolation valves.

The trip setpoint is established such that the actual ~~submersion~~ dose rate would not exceed ~~10 mR/hr~~ two times background at Rated Thermal Power (RTP) in the containment building. ~~The setpoint value may be increased up to twice the maximum concentration activity in containment determined by the~~ A sample analysis is performed prior to each release ~~provided~~ to ensure the ~~value~~ resultant dose does not exceed ~~10% of~~ the limits determined by the Offsite Dose Calculation Manual.

BASES

APPLICABLE SAFETY ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event (i.e., within approximately 60 seconds). The isolation of the purge valves has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The containment ventilation isolation radiation monitors act as backup to the SI signal to ensure closing of the purge valves. ~~They are also the primary means for automatically isolating containment in the event of a fuel handling accident.~~ Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental radiological doses are below 10 CFR 50.67, “Accident Source Term,” (Ref. 1) limits.

The containment ventilation isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Ventilation Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. Manual Initiation - Phase A

   Refer to LCO 3.3.2, Function 3.a.1, for all initiating Functions and requirements.

2. Manual Initiation - Phase B

   Refer to LCO 3.3.2, Function 3.b.1, for all initiating Functions and requirements.

BASES

APPLICABILITY

The Containment Ventilation Isolation Functions must be OPERABLE in the MODES ~~or other specified conditions~~ identified in Table 3.3.6-1. Under these conditions, the potential exists for an accident that could release fission product radioactivity into containment. Therefore, the containment ventilation isolation instrumentation must be OPERABLE in these MODES.

While in MODES 5 and 6 without fuel handling in progress, ~~or with a penetration closed by a manual or automatic isolation valve, blind flange, or equivalent,~~ the containment ventilation isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1. If fuel handling is in progress for RECENTLY IRRADIATED FUEL, the containment ventilation isolation instrumentation need not be OPERABLE as each penetration providing access from the containment atmosphere to the outside atmosphere must be closed by a manual or automatic isolation valve, blind flange, or equivalent in accordance with LCO 3.9.4.c.

The Applicability for the containment ventilation isolation on the ESFAS Manual Initiation-Phase A, Manual Initiation-Phase B, and Safety Injection Functions are specified in LCO 3.3.2. Refer to the Bases for LCO 3.3.2 for discussion of the Manual Initiation-Phase A, Manual Initiation-Phase B, and Safety Injection Functions Applicabilities. The Applicability for the containment ventilation isolation Automatic Actuation Logic and Actuation Relays is MODES 1, 2, 3, and 4. The Applicability for the containment ventilation isolation on Containment Radiation - High is MODES 1, 2, 3, 4. ~~, and when Item C.2 of LCO 3.9.4 is required.~~

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by plant specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

BASES

ACTIONS (continued)

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for the Automatic Actuation Logic and Actuation Relays Function and the Containment Radiation - High Function listed in Table 3.3.6-1. The Completion Time(s) of the inoperable channel(s)/train(s) of the given Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the failure of one containment ventilation isolation radiation monitor channel. Condition A requires the inoperable channel to be restored to OPERABLE status within 4 hours. The Completion Time is justified by the low likelihood of events occurring during this interval, and recognition that the remaining channel will respond to most events.

B.1

Condition B addresses the train orientation of the Solid State Protection System (SSPS) and the master and slave relays for the Containment Ventilation Isolation Function. It also addresses the failure of both radiation monitoring channels or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

If one or both automatic actuation trains are inoperable, both radiation monitoring channels are inoperable, or the Required Action and associated Completion Time of Condition A is not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation.

~~Condition B is modified by a Note stating that the Condition is only applicable in MODE 1, 2, 3, or 4.~~

BASES

ACTIONS (continued)

~~C.1 and C.2~~

~~Condition C addresses the failure of both radiation monitoring channels or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1. If both channels are inoperable or the Required Action and associated Completion Time of Condition A is not met, operation may continue as long as the Required Action to place and maintain containment purge valves in their closed position is met or the applicable Conditions of LCO 3.9.4, "Containment Penetrations," are met for each valve made inoperable by failure of isolation instrumentation. The Completion Time for these Required Actions is immediately.~~

~~A Note states that Condition C is only applicable when Item C.2 of LCO 3.9.4 is required.~~

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which Containment Ventilation Isolation Functions.

SR 3.3.6.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

BASES

BACKGROUND (continued)

The Containment Ventilation Isolation System consists of the normal purge subsystem, the mini purge subsystem, and the post Loss Of Coolant Accident purge subsystem. These three subsystems contain penetrations which provide direct access from the containment to the outside atmosphere. In MODE 6, the minipurge subsystem is normally used to exchange large volumes of containment air to support refueling operations. Each penetration contains inside and outside containment isolation valves which close automatically on an actuation signal~~.~~; however, d~~D~~uring movement of RECENTLY IRRADIATED FUEL within containment, each penetration providing direct access from the containment atmosphere to the outside atmosphere will be closed by a manual or automatic isolation valve, blind flange, or equivalent in accordance with LCO 3.9.4.c. The Containment Ventilation Isolation System is not relied on to provide automatic containment closure and; therefore, need not be OPERABLE.~~all required valves within a subsystem must be capable of being closed by a containment ventilation isolation signal whenever the associated subsystem is in operation.~~ A list of the instrumentation which functions to isolate the valves in these penetrations is provided in LCO 3.3.6, "Containment Ventilation Isolation Instrumentation."

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by a closed automatic isolation valve, a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods allowed under the provisions of 10 CFR 50.59 may include use of a material that can provide a temporary atmospheric pressure ventilation barrier during movement of RECENTLY IRRADIATED FUEL within the containment.

BASES

APPLICABLE SAFETY ANALYSES

During movement of RECENTLY IRRADIATED FUEL assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents, analyzed in Reference 2, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," ensure that the release of fission product radioactivity, subsequent to a fuel handling accident in containment, results in doses that are within the 10 CFR 50.67 (Ref. 5) limits. The radiological dose assessments for the Design Basis Fuel Handling Accident in containment were performed in accordance with the guidance of Regulatory Guide 1.183 (Ref 4).

When moving RECENTLY IRRADIATED FUEL in containment, the requirements of the LCO must be met with the exception that LCO 3.9.4.a need not be met if at least one train of the FHB Ventilation System is OPERABLE as specified in the Note. This exception permits movement of RECENTLY IRRADIATED FUEL with both personnel air lock doors open or the equipment hatch not intact.

When moving fuel in the containment that is not RECENTLY IRRADIATED FUEL, the LCO is not applicable. Due to radioactive decay, neither containment closure nor an OPERABLE FHB Ventilation System train are required to meet the dose limits of 10 CFR 50.67 during a fuel handling accident.

Another consideration, which may result in a limiting decay time prior to fuel handling, is the impact of decay heat on the spent fuel pool cooling requirements described in Reference 3.

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO limits the consequences of a fuel handling accident involving handling RECENTLY IRRADIATED FUEL in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed. ~~except for the OPERABLE containment purge (supply and exhaust) penetrations. For the OPERABLE containment purge~~

BASES

LCO (continued)

~~penetrations, this LCO ensures that unisolated penetrations are isolable by the Containment Ventilation Isolation System.~~ The ~~OPERABILITY~~ requirements for this LCO ensure that containment closure is ~~the automatic purge valve closure times specified in the UFSAR can be~~ achieved and, therefore, meet the assumptions used in the safety analysis to ensure ~~that releases through the valves are terminated, such~~ that radiological doses are within the acceptance limits.

The LCO is modified by a Note which allows both personnel air lock doors to be open or the equipment hatch not intact when the FHB Ventilation System is in compliance with TS 3.7.13. When the equipment hatch is installed it serves to contain fission product radioactivity that may be released following a fuel handling accident in the containment. When the equipment hatch is not intact, or when both doors of the personnel air lock are simultaneously opened, the internal containment pressure is essentially equal to the internal pressure of the fuel handling building. In the event of a fuel handling accident in the containment, realigning of the fuel handling building ventilation system creates a negative pressure in the containment and fuel handling building relative to the auxiliary building and outside atmosphere. The negative pressure ensures that any radioactivity released to the containment atmosphere will either remain in the containment or be filtered through a FHB Ventilation System train. As such, with the equipment hatch not intact, or with both personnel air lock doors open, the consequences of a fuel handling accident involving RECENTLY IRRADIATED FUEL in containment would not exceed those calculated for a fuel handling accident involving RECENTLY IRRADIATED FUEL in the fuel handling building.

In addition, a commitment has been made to implement compensatory measures during movement of irradiated fuel as described in UFSAR Section 15.7.4, “Fuel Handling Accidents.” These compensatory measures support the Alternate Source Term methodology and reduce doses even further below that provided by natural decay and avoid unmonitored releases in the event of a postulated fuel handling accident.

BASES

APPLICABILITY

The containment penetration requirements are applicable during movement of RECENTLY IRRADIATED FUEL assemblies within containment because this is when there is a potential for a limiting fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODE 5, and in MODE 6 when movement of RECENTLY IRRADIATED FUEL assemblies within containment are not being conducted, the potential for a limiting fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1

If the containment equipment hatch, air lock doors, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, the unit must be placed in a condition where containment closure is not needed. This is accomplished by immediately suspending movement of RECENTLY IRRADIATED FUEL assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that each of the containment penetrations required to be isolated is isolated. ~~This Surveillance for the open purge valves demonstrates that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power which will ensure that each valve is capable of being closed by an OPERABLE automatic Containment Ventilation Isolation signal.~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~SR 3.9.4.2~~

~~This Surveillance demonstrates that each required containment purge valve actuates to its isolation position on an actual or simulated high radiation signal. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. SR 3.9.4.3 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.~~

~~SR 3.9.4.3~~

~~This Surveillance demonstrates that the isolation time of each required containment purge valve providing direct access from the containment atmosphere to the outside atmosphere is in accordance with the Inservice Testing Program requirements. This SR, along with SR 3.9.4.2, ensures the containment purge valves in penetrations which provide direct access from the containment atmosphere to the outside atmosphere are capable of closing after a postulated fuel handling accident to limit the release of fission product radioactivity from the containment.~~

REFERENCES

1. UFSAR, Section 15.7.4.

2. NUREG-0800, Section 15.0.1, Rev. 0, July 2000.

3. NUREG-0800, Section 9.1.3.

4. Regulatory Guide 1.183, July 2000.

5. 10 CFR 50.67.