Duke Energy

W. R. McCollum, Jr.
Vice President

Duke Energy
Oconee Nuclear Station
7800 Rochester Highway
Seneca, SC 29672
(864) 885-3107 OFFICE
(864) 885-3564 FAX

October 16, 2001

U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ATTENTION: Document Control Desk

SUBJECT: Duke Energy Corporation
Oconee Nuclear Station Units 1, 2, and 3
Docket Nos. 50-269, 50-270, 50-287
License Nos. NPF-38, NPF-47, and NPF-55
Licensed Amendment Request for Full-Scope Implementation of the Alternative Source Term and Technical Specifications 3.3.5, Engineered Safeguards Protective System (ESPS) Analog Instrumentation; 3.3.6, Engineered Safeguards Protective System (ESPS) Manual Initiation; 3.3.7, Engineered Safeguards Protective System (ESPS) Digital Automatic Actuation Logic Channels; 3.7.10,Penetration Room Ventilation System; 3.7.17 Spent Fuel Pool Ventilation System; 3.9.3,Containment Penetrations; 5.5.2,Containment Leakage Rate Testing Program; and 5.5.12, Ventilation Filter Testing Program
Technical Specification Change Number 01-07

Reference: NRC meeting with Duke Energy Corporation on June 9, 2000, on Oconee 1, 2, and 3 Control Room Habitability Issues (TAC Nos. MA8942, MA8943, and MA8944)

Pursuant to 10CFR50.90, this letter submits a license amendment request (LAR) for the Oconee Nuclear Station Facility Operating License (FOL) and Technical Specifications (TS). This amendment is requested in accordance with the

A001

requirements of 10 CFR 50.67, which addresses the use of an Alternative Source Term (AST) at operating reactors, and relevant guidance provided in Regulatory Guide 1.183. This amendment requests approval of AST analysis methodology for the Oconee Nuclear Station that will support simplification of Ventilation System testing requirements and amend containment operability requirements during core alterations or movement of irradiated fuel. This amendment represents full-scope implementation of the AST for Oconee Nuclear Station as described in Regulatory Guide 1.183.

As a result of this AST implementation, coupled with plant modifications described later, several improvements to the FOL and TS have been developed. Specifically, this amendment revises the subject TS. The proposed changes include the following revisions:

- The Penetration Room Ventilation System (PRVS) is removed from TS because the PRVS will not be credited in licensing analyses that determine Control Room and off-site doses.
- The Spent Fuel Pool Ventilation System (SFPVS) is removed from TS because the SFPVS will not be credited in licensing analyses that determine Control Room and off-site doses.
- During certain refueling operations, the containment air locks and/or the equipment hatch and penetrations providing direct access from the containment atmosphere to the outside atmosphere will be permitted to be unisolated under administrative controls. Additionally, the requirement to maintain an operable automatic isolation capability for the Reactor Building Purge system during refueling is being removed from TS.
- The allowable value for the Reactor Building leakage rate is lowered from 0.25 w%/day to 0.20 w%/day.
- The requirement to measure Reactor Building leakage in excess of 50% of L_a to the penetration room is removed from TS.
- The Ventilation Filter Testing Program (VFTP) is revised to remove all references to the PRVS and SFPVS and their testing requirements.
- The VFTP acceptance criterion for the CRVS Booster Fan trains is revised to require ≥ 97.5% radioactive methyl iodide removal.

Discussions of each of the changes proposed are provided in Attachment 3.

The contents of this amendment package are as follows:

Attachment 1 provides a marked copy of the current Technical Specifications showing the proposed changes.

Attachment 2 provides the proposed new Technical Specifications.

Attachment 3 provides a Description of the Proposed Changes and Technical Justification.

Pursuant to 10 CFR 50.92, Attachment 4 documents the determination that the amendment contains No Significant Hazards Considerations.

Pursuant to 10 CFR 51.22(c)(9), Attachment 5 provides the basis for the categorical exclusion from performing an Environmental Assessment/Impact Statement.

Implementation of this amendment to the Oconee Facility Operating License and TS will impact the Oconee UFSAR. As a result of implementing this LAR, it will be necessary to revise various sections of the Oconee UFSAR. Necessary changes will be made in accordance with 10CFR50.71(e).

Plant Modifications are necessary to support the revised analysis methodology, the margin embedded in the analysis input values and assumptions and the overall resolution of issues related to Control Room Habitability. The existing Control Room outside air intake will be relocated from the roof of the Auxiliary Building to the roof of the Turbine Building. Dual intakes will be installed for each Control Room. This modification is necessary to support relaxations on containment closure during fuel movement. The High Pressure Injection/Low Pressure Injection (HPI/LPI) relief valve discharge will be re-routed back into the Reactor Building. This modification combined with the Control Room outside air intake modification is necessary to support elimination of credit for PRVS. Additionally, the existing active Caustic Addition System will

be replaced with a passive Caustic Addition System. The passive design of the system will remove the need for Operator action to control post-accident pH and the design will not be susceptible to single active failures.

Establishment of the appropriate attributes of a Control Room Habitability Program is a long standing industry issue. Duke is an active member of the Nuclear Energy Institute (NEI) Taskforce that has developed the revision to the NEI guidance document, NEI 99-03, Control Room Habitability Program. As such, Duke is continually aware of the industry efforts to resolve Control Room Habitability issues. Duke will consider the implications of the final resolution on this matter and evaluate any formulated guidance published either by the industry or the Commission as part of the resolution of the concern. Duke will continue to work closely with the industry efforts and staff on this issue.

Implementation of the AST into the Oconee design basis will be integrated with the completion of the associated plant modifications. In order to support the installation of the required modifications, approval of this LAR is requested by October 31, 2002.

In accordance with Duke administrative procedures and the Quality Assurance Program Topical Report, this proposed amendment has been previously reviewed and approved by the Oconee Plant Operations Review Committee and the Duke Corporate Nuclear Safety Review Board.

Pursuant to 10 CFR 50.91, a copy of this proposed amendment is being sent to the State of South Carolina.

Inquiries should be directed to L.E. Nicholson at (864) 885-3292.

Very truly yours,

W. R. McCollum, Jr.
Vice President, Oconee Nuclear Station

Attachments
xc w/attachments:

L. A. Reyes
U. S. Nuclear Regulatory Commission
Administrator, Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

L. N. Olshan
NRC Senior Project Manager
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

M. C. Shannon
Senior Resident Inspector
U. S. Nuclear Regulatory Commission
Oconee Nuclear Site

R. Wingard, Director
Division of Radioactive Waste Management
South Carolina Bureau of Land and Waste Management
2600 Bull Street
Columbia, SC 29207

AFFIDAVIT

William R. McCollum, Jr. states that he is Vice President, Oconee Nuclear Site, of Duke Energy Corporation; that he is authorized on the part of said corporation to sign and file with the Nuclear Regulatory Commission this amendment to the Oconee Nuclear Station(s) Facility Operating Licenses No(s). NPF-38, NPF-47, and NPF-55 and Technical Specifications; and that all statements and matters set forth herein are true and correct to the best of his knowledge.

W R McCollum, Vice President, Oconee Nuclear Site

Subscribed and sworn to me: October 16, 2001
Date

Connie M Breazeale, Notary Republic

My Commission Expires: 2/12/2003
Date

SEAL

ATTACHMENT 1

Marked copy of current Technical Specification pages showing the proposed changes

3.3 INSTRUMENTATION

3.3.6 Engineered Safeguards Protective System (ESPS) Manual Initiation

LCO 3.3.6 Two manual initiation channels of each one of the ESPS Functions below shall be OPERABLE:

a. High Pressure Injection, Reactor Building (RB) Non-Essential Isolation, Keowee Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input (ES Channels 1 and 2);

b. Low Pressure Injection and RB Essential Isolation (ES Channels 3 and 4);

c. RB Cooling~~,~~ and RB Essential Isolation ~~and Penetration Room Ventilation~~ (ES Channels 5 and 6)~~; and~~

d. RB Spray (ES Channels 7 and 8).

APPLICABILITY: MODES 1 and 2,
MODES 3 and 4 when associated engineered safeguard equipment is required to be OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ESPS Functions with one channel inoperable.	A.1 Restore channel to OPERABLE status.	72 hours

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ESPS manual initiation ensures that the control room operator can rapidly initiate ES Functions. The manual initiation trip Function is required as a backup to automatic trip functions and allows operators to initiate ESPS whenever any parameter is rapidly trending toward its trip setpoint.

The ESPS manual initiation functions satisfy Criterion 3 of 10 CFR 50.36 (Ref. 1).

LCO

Two ESPS manual initiation channels of each ESPS Function shall be OPERABLE whenever conditions exist that could require ES protection of the reactor or RB. Two OPERABLE channels ensure that no single random failure will prevent system level manual initiation of any ESPS Function. The ESPS manual initiation Function allows the operator to initiate protective action prior to automatic initiation or in the event the automatic initiation does not occur.

The required Function is provided by two associated channels as indicated in the following table:

Function	Associated Channels
HPI and RB Non-Essential Isolation, Keowee Emergency Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input	1 & 2
LPI and RB Essential isolation	3 & 4
RB Cooling and ~~,~~ RB Essential isolation ~~and Penetration Room Vent.~~	5 & 6
RB Spray	7 & 8

APPLICABILITY

The ESPS manual initiation Functions shall be OPERABLE in MODES 1 and 2, and in MODES 3 and 4 when the associated engineered safeguard equipment is required to be OPERABLE. The manual initiation channels are required because ES Functions are designed to provide protection in these MODES. ESPS initiates systems that are either reconfigured for decay heat removal operation or disabled while in MODES 5 and 6. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components. Adequate time is available to evaluate unit conditions and to respond by manually operating the ES components, if required.

B 3.3 INSTRUMENTATION

B 3.3.5 Engineered Safeguards Protective System (ESPS) Analog Instrumentation

BASES

BACKGROUND

The ESPS initiates necessary safety systems, based on the values of selected unit Parameters, to protect against violating core design limits and to mitigate accidents.

ESPS actuates the following systems:

- High pressure injection (HPI);
- Low pressure injection (LPI);
- Reactor building (RB) cooling;
- ~~Penetration room ventilation;~~
- RB Spray;
- RB Isolation; and
- Keowee Hydro Unit Emergency Start.

The ESPS operates in a distributed manner to initiate the appropriate systems. The ESPS does this by determining the need for actuation in each of three analog channels monitoring each actuation Parameter. Once the need for actuation is determined, the condition is transmitted to digital automatic actuation logic channels, which perform the two-out-of-three logic to determine the actuation of each end device. Each end device has its own automatic actuation logic, although all digital automatic actuation logic channels take their signals from the same bistable in each channel for each Parameter.

Four Parameters are used for actuation:

- Low Reactor Coolant System (RCS) Pressure;
- Low Low RCS Pressure;
- High RB Pressure; and
- High High RB Pressure.

BASES

BACKGROUND (continued)

LCO 3.3.5 covers only the analog instrumentation channels that measure these Parameters. These channels include all intervening equipment necessary to produce actuation before the measured process Parameter exceeds the limits assumed by the accident analysis. This includes sensors, bistable devices, operational bypass circuitry, and output relays. LCO 3.3.6, "Engineered Safeguards Protective System (ESPS) Manual Initiation," and LCO 3.3.7, "Engineered Safeguards Protective System (ESPS) Digital Automatic Actuation Logic Channels," provide requirements on the manual initiation and digital automatic actuation logic Functions.

The ESPS contains three analog channels. Each analog channel provides input to digital logic channels that initiate equipment with a two-out-of-three logic on each digital logic channel. Each analog channel includes inputs from one analog instrumentation channel of Low RCS Pressure, Low Low RCS Pressure, High RB Pressure, and High High RB Pressure. Digital automatic actuation logic channels combine the three analog channel trips to actuate the individual Engineered Safeguards (ES) components needed to initiate each ES System. Figure 7.5, UFSAR, Chapter 7 (Ref. 1), illustrates how analog instrumentation channel trips combine to cause digital logic channel trips.

The following matrix identifies the analog instrumentation (measurement) channels and the Digital Automatic Actuation Logic Channels actuated by each.

Digital Logic Channels	Actuated Systems/ Functions	RCS PRESS LOW	RCS PRESS LOW LOW	RB PRESS HIGH	RB PRESS HIGH HIGH
1 and 2	HPI and RB Non-Essential Isolation, Keowee Emergency Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input	X		X	
3 and 4	LPI and RB Essential isolation		X	X	
5 and 6	RB Cooling ~~,~~ and RB Essential isolation ~~, and Penetration Room Vent.~~			X	
7 and 8	RB Spray				X

The ES equipment is generally divided between the two redundant digital actuation logic channels. The division of the equipment between the two digital actuation logic channels is based on the equipment redundancy and

BASES

APPLICABLE SAFETY ANALYSES (continued)

Reactor Building Spray, Reactor Building Cooling, and Reactor Building Isolation

The ESPS actuation of the RB coolers and RB Spray have been credited in RB analysis for LOCAs, both for RB performance and equipment environmental qualification pressure and temperature envelope definition. Accident dose calculations have credited RB Isolation and RB Spray.

~~Penetration Room Ventilation Actuation~~

~~The ESPS actuation of the penetration room ventilation system has been assumed for LOCAs. Accident dose calculations have credited penetration room ventilation.~~

Keowee Hydro Unit Emergency Start

The ESPS initiated Keowee Hydro Unit Emergency Start has been included in the design to ensure that emergency power is available throughout the limiting LOCA scenarios.

The small break LOCA analyses assume a conservative 48 second delay time for the actuation of HPI and LPI in UFSAR, Chapter 15 (Ref. 4). The large break LOCA analyses assume LPI flow starts in 38 seconds while full LPI flow does not occur until 15 seconds later, or 53 seconds total (Ref. 4). This delay time includes allowances for Keowee Hydro Unit starting, Emergency Core Cooling Systems (ECCS) pump starts, and valve openings. Similarly, the RB Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system analyzed.

Accident analyses rely on automatic ESPS actuation for protection of the core temperature and containment pressure limits and for limiting off site dose levels following an accident. These include LOCA, and MSLB events that result in RCS inventory reduction or severe loss of RCS cooling.

The ESPS channels satisfy Criterion 3 of 10 CFR 50.36 (Ref. 5).

LCO

The LCO requires three analog channels of ESPS instrumentation for each Parameter in Table 3.3.5–1 to be OPERABLE in each ESPS digital automatic actuation logic channel. Failure of any instrument renders the affected analog channel(s) inoperable and reduces the reliability of the affected Functions.

BASES

BACKGROUND (continued)

includes allowances for Keowee Hydro Unit startup and loading, ECCS pump starts, and valve openings. Similarly, the reactor building (RB) Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system.

The ESPS automatic initiation of Engineered Safeguards (ES) Functions to mitigate accident conditions is assumed in the accident analysis and is required to ensure that consequences of analyzed events do not exceed the accident analysis predictions. Automatically actuated features include HPI, LPI, RB Cooling, RB Spray, and RB Isolation.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on automatic ESPS actuation for protection of the core and RB and for limiting off site dose levels following an accident. The digital automatic actuation logic is an integral part of the ESPS.

The ESPS digital automatic actuation logic channels satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

The digital automatic actuation logic channels are required to be OPERABLE whenever conditions exist that could require ES protection of the reactor or the RB. This ensures automatic initiation of the ES required to mitigate the consequences of accidents.

The required Function is provided by two associated digital channels as indicated in the following table:

Function	Associated Channels
HPI and RB Non-Essential Isolation, Keowee Emergency Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input	1 & 2
LPI and RB Essential isolation	3 & 4
RB Cooling~~,~~ and RB Essential isolation~~,~~ ~~and Penetration Room Vent.~~	5 & 6
RB Spray	7 & 8

Delete

3.7 PLANT SYSTEMS

3.7.10 Penetration Room Ventilation System (PRVS)

LCO 3.7.10 Two PRVS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PRVS train inoperable.	A.1 Restore PRVS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each PRVS train for $\geq$ 15 minutes.	31 days
SR 3.7.10.2	Perform required PRVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.10.3	Verify each PRVS train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.10.4	Verify one PRVS train can maintain a negative pressure ≥ 0.06 inches water gauge relative to atmospheric pressure during operation at a flow rate of ≥ 900 cfm and ≤ 1100 cfm.	18 months on a STAGGERED TEST BASIS
SR 3.7.10.5	Verify the PRVS filter cooling bypass valve can be opened.	18 months

Delete

Delete

B 3.7 PLANT SYSTEMS

B 3.7.10 Penetration Room Ventilation System (PRVS)

BASES

BACKGROUND

The PRVS filters air from the area of the active penetration rooms during the recirculation phase of a loss of coolant accident (LOCA).

The PRVS consists of two independent, redundant trains. Each train consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated carbon adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system. The system initiates filtered ventilation of the Reactor Building penetration rooms area following receipt of an Engineered Safeguards actuation signal (ESAS).

The PRVS is a standby system. During emergency operations, the PRVS valves are realigned, and fans are started to begin filtration. Upon receipt of the ESAS signal(s), the stream of ventilation air discharges through the system filter trains. The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and carbon adsorbers.

The PRVS is discussed in the UFSAR, Sections 6.5.1, 9.4.7, and 15.4.7 (Refs. 1, 2, and 3, respectively).

APPLICABLE SAFETY ANALYSES

The design basis of the PRVS is established by the Maximum Hypothetical Accident (MHA). In such a case, the system limits radioactive releases to within 10 CFR 100 (Ref. 7) requirements and personnel doses in the Control Room are maintained within the limits of 10 CFR 20 (Ref. 4). The analysis of the effects and consequences of an MHA is presented in Reference 3. No credit is taken in the analysis for any reduction in Control Room Dose provided by the PRVS filters.

The PRVS also actuates following a large and small break LOCA, in those cases where the unit goes into the recirculation mode of long term cooling, and to cleanup releases of smaller leaks, such as from valve stem packing.

Following a LOCA, an ESAS starts the PRVS fans and opens the dampers located in the penetration room outlet ductwork.

The PRVS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 5).

Delete

BASES (continued)

LCO

Two independent and redundant trains of the PRVS are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train coincident with loss of offsite power.

The PRVS is considered OPERABLE when the individual components necessary to maintain the penetration room filtration are OPERABLE in both trains.

A PRVS train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and carbon adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Ductwork, valves, and dampers are OPERABLE, and air flow can be maintained.

In addition, the penetration room boundaries, including the integrity of the walls, floors, ceilings, ductwork, and access doors, must be maintained within the assumptions of the design analysis.

APPLICABILITY

In MODES 1, 2, 3, and 4, the PRVS is required to be OPERABLE consistent with the OPERABILITY requirements of the containment.

In MODES 5 and 6, the PRVS is not required to be OPERABLE since the containment is not required to be OPERABLE.

ACTIONS

A.1

With one PRVS train inoperable, action must be taken to restore the PRVS train to OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the PRVS safety function. However, the overall reliability is reduced because a single failure in the OPERABLE PRVS train could result in loss of PRVS function.

The 7 day Completion Time is appropriate because the risk contribution is less than that of the ECCS (72 hour Completion Time), and this system is not a direct support system for the ECCS. The 7 day Completion Time is based on the low probability of an accident occurring during this time period, and ability of the remaining train to provide the required capability.

Delete

BASES

ACTIONS (continued)

B.1 and B.2

If the required Action and associated Completion Time are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. The 31 day Frequency is based on known reliability of equipment and the two train redundancy available.

SR 3.7.10.2

This SR verifies that the required PRVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance and carbon adsorber efficiency. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.10.3

This SR verifies that each PRVS train starts and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with the guidance in Reference 6.

SR 3.7.10.4

This SR verifies the integrity of the penetration rooms area. The ability of the PRVS to maintain a negative pressure, with respect to outside atmosphere, is periodically tested to verify proper functioning of the PRVS. During the post accident mode of operation, the PRVS is

Delete

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10.4 (continued)

designed to maintain a slight negative pressure in the penetration rooms with respect to outside atmosphere to prevent unfiltered LEAKAGE. The PRVS is designed to maintain this negative pressure at a flow rate of 1000 ± 10% cfm from the area. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration SRs.

SR 3.7.10.5

Operating the PRVS filter bypass valve is necessary to ensure that the system functions properly. The OPERABILITY of the PRVS filter bypass valve is verified if it can be opened. An 18 month Frequency is consistent with the guidance in Reference 6.

REFERENCES

1. UFSAR, Section 6.5.1.
2. UFSAR, Section 9.4.7.
3. UFSAR, Section 15.15.
4. 10 CFR 20.
5. 10 CFR 50.36.
6. Regulatory Guide 1.52.
7. 10 CFR 100.

Delete

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Pool Ventilation System (SFPVS)

LCO 3.7.17 Two SFPVS trains shall be OPERABLE.

-----NOTES-----

1. LCO 3.0.3 is not applicable.
2. Not applicable during reracking operations with no fuel in the spent fuel pool.

APPLICABILITY: During movement of fuel in the spent fuel pool.
During crane operations with loads over the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SFPVS train inoperable.	A.1 Place OPERABLE SFPVS train in operation.	Immediately
	OR	
	A.2.1 Suspend movement of fuel in the spent fuel pool	Immediately
	AND	
	A.2.2 Suspend crane operations with loads over the spent fuel pool.	Immediately

(continued)

Delete

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two SFPVS trains inoperable.	B.1.1 Suspend movement of fuel in the spent fuel pool.	Immediately
	AND	
	B.1.2 Suspend crane operations with loads over the spent fuel pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Operate each SFPVS train for ≥ 15 minutes.	31 days
SR 3.7.17.2	Perform required SFPVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

Delete

B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Pool Ventilation System (SFPVS)

BASES

BACKGROUND

Ventilation air for the Spent Fuel Pool Area is supplied by an air handling unit which consists of roughing filters, steam heating coil, cooling coil supplied by low pressure service water, and a centrifugal fan. In the normal mode of operation, the air from the Spent Fuel Pool Area is exhausted directly to the unit vents by the general Auxiliary Building exhaust fans. The filtered exhaust system consists of a single filter train and two 100 percent capacity vane axial fans. The filter train utilized is the Reactor Building Purge Filter Train. The Unit 2 Reactor Building purge filter train is used for the combined Unit 1 and 2 Spent Fuel Pool Ventilation System, The Unit 3 Reactor Building purge filter train is used for the Unit 3 SFP Ventilation System. The filter train is comprised of prefilters, HEPA filters, and charcoal filters. To control the direction of air flow, i.e., to direct the air from the Fuel Pool Area to the Reactor Building Purge Filter Train, a series of pneumatic motor operated dampers are provided along with a crossover duct from the Fuel Pool to the filter train.

The SFPVS is discussed in the UFSAR, Section 9.4.2, (Ref. 1).

APPLICABLE SAFETY ANALYSES

The analysis of the limiting fuel handling accident, the cask drop accident, given in Reference 2, assumes that a certain number of fuel assemblies are damaged. The DBA analysis for the cask drop accident, does not assume operation of the SFPVS. These assumptions and the analysis are consistent with the guidance provided in Regulatory Guide 1.25 (Ref. 3).

The SFPVS does not satisfy the criteria in 10 CFR 50.36

LCO

Two redundant trains of the SFPVS are required to be OPERABLE to ensure that at least one is available, assuming a single failure that disables the other train.

Delete

BASES

LCO (continued)

An SFPVS train is considered OPERABLE when its associated:

1. Fan is OPERABLE;
2. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and
3. Ductwork and dampers are OPERABLE, and air flow can be maintained.

The LCO is modified by two Notes. Note 1 states LCO 3.0.3 does not apply. If moving fuel or conducting crane operations with load over the storage pool while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving fuel or conducting crane operations with load over the storage pool while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown. Note 2 states the requirements of this LCO is not applicable during reracking operations with no fuel in the spent fuel pool. With no fuel in the spent fuel pool, the potential release of radioactive material to the environs resulting from crane operations with load over the storage pool is substantially reduced.

APPLICABILITY

During movement of fuel in the fuel handling area or during crane operations with loads over the spent fuel pool, the SFPVS is always required to be OPERABLE.

ACTIONS

A.1 and A.2

With one SFPVS train inoperable, the OPERABLE SFPVS train must be started immediately with its discharge through the associated reactor building purge filter or fuel movement in the spent fuel pool and crane operations with loads over the spent fuel pool suspended. This action ensures that the remaining train is OPERABLE, and that any active failures will be readily detected.

If the system is not placed in operation, this action requires suspension of fuel movement and suspension of crane operation with loads over the spent fuel pool, which precludes a fuel handling accident. This action does not preclude the movement of fuel assemblies or crane loads to a safe position.

Delete

BASES

ACTIONS (continued)	B.1 When two trains of the SFPVS are inoperable during movement of fuel in the spent fuel pool, the unit must be placed in a condition in which the LCO does not apply. This Action involves immediately suspending movement of fuel assemblies in the spent fuel pool and suspension of crane operations with loads over the spent fuel pool. This does not preclude the movement of fuel or crane loads to a safe position.
SURVEILLANCE REQUIREMENTS	SR 3.7.17.1 Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system. Systems without heaters need only be operated through the associated reactor building purge filters at a design flow ± 10% for ≥ 15 minutes to demonstrate the function of the system. The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy. SR 3.7.17.2 This SR verifies that the required SFPVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.
REFERENCES	1. UFSAR, Section 9.4.2. 2. UFSAR, Section 15.11. 3. Regulatory Guide 1.25.

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch is capable of being closed ~~and held in place by a minimum of four bolts~~;

b. One door in each air lock is capable of being closed; and

~~NOTE~~

~~An emergency air lock door is not required to be closed when a temporary cover plate is installed.~~

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere ~~either:~~ is capable of being closed.

~~1. closed by a manual, non-automatic power operated or automatic isolation valve, blind flange, or equivalent, or~~

~~2. capable of being closed by an OPERABLE Reactor Building Purge supply and exhaust isolation signal.~~

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1	Suspend CORE ALTERATIONS	Immediately
	<u>AND</u>		
	A.2	Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.3.2	Verify each required Reactor Building Purge supply and exhaust isolation valve that is not locked, sealed or otherwise secured in the isolation position actuates to the isolation position on an actual or simulated high radiation actuation signal.	Once each refueling outage prior to CORE ALTERATIONS or movement of irradiated fuel assemblies within containment

Delete

INSERT 1

During movement of irradiated fuel assemblies within containment, administrative controls shall ensure that appropriate personnel are aware that the equipment hatch is open, that a specific individual(s) is designated and available to close the equipment hatch cover following a required evacuation of containment, and that any obstruction(s) (e.g., cables and hoses) that could prevent closure of the equipment hatch cover be capable of being quickly removed. Should a fuel handling accident occur inside containment, the equipment hatch will be closed following evacuation. For closure, the equipment hatch cover will be in place with a minimum of four bolts securing the cover to the sealing surface. Good engineering practice dictates that the bolts required be approximately equally spaced.

INSERT 2

During movement of irradiated fuel assemblies within containment, administrative controls shall ensure that appropriate personnel are aware that both personnel airlock doors are open, that a specific individual(s) is designated and available to close the airlock door following a required evacuation of containment, and any obstruction (s)(e.g., cables and hoses) that could prevent closure of an airlock door be capable of being quickly removed. Should a fuel handling accident occur inside containment, at least one of the personnel and/or emergency air lock doors will be closed following evacuation.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. In order to make this distinction, the penetration requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that specified escape paths are closed or capable of being closed. Since there is no significant potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained within the requirements of 10CFR50.67 ~~10 CFR 100~~. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be Capable of being closed ~~held in place by at least four bolts~~. ~~Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced~~.

Insert 1

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown

requires that one door in each air lock be capable of being closed.

BACKGROUND (continued)

when containment OPERABILITY is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment ingress and egress is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure ~~is required; therefore,~~ The door interlock mechanism may remain disabled. ~~but one air lock door must always remain closed. Placement of a temporary cover plate in the emergency air lock is an acceptable means for providing containment closure.~~

Insert 2

~~The temporary cover plate is installed and sealed against the inner emergency air lock door flange gasket. The temporary cover plate is visually inspected to ensure that no gaps exist. All cables, hoses and service air piping run through the sleeves on the temporary cover plate will also be installed and sealed. The sleeves will also be inspected to ensure that no gaps exist. Leak testing is not required prior to beginning fuel handling operations. Therefore, visual inspection of the temporary cover plate over the emergency air lock satisfies the requirement that the air lock be closed, which constitutes operability for this requirement.~~

The requirements on containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted ~~from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.~~ to within regulatory limits.

The Reactor Building Purge System includes a supply penetration and exhaust penetration. During MODES 1, 2, 3, and 4, two valves in each of the supply and exhaust penetrations are secured in the closed position. The system is not subject to a Specification in MODE 5.

In MODE 6, large air exchanges are necessary to support refueling operations. The purge system is used for this purpose. ~~and two valves in each penetration flow path may be closed on a unit vent high radiation signal.~~

Insert 2A

~~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, non-automatic power operated valve, manual isolation valve, blind flange, or equivalent. Equivalent isolation methods may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier for the containment penetration(s) during fuel movements.~~

INSERT 2A

During movement of irradiated fuel assemblies within containment, administrative controls shall ensure that appropriate personnel are aware that containment isolation valves are open, that a specific individual (s) is designated and available to close the valve following a required evacuation of containment, and any obstruction (s) (e.g., cables and hoses) that could prevent closure of a containment isolation valve be capable of being quickly removed. Should a fuel handling accident occur inside containment, containment isolation valves will be closed on at least one side. Isolation may be achieved by a closed automatic isolation valve, non-automatic power operated valve, manual isolation valve, blind flange, or equivalent.

BASES (continued)

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of 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. 2). A minimum fuel transfer canal water level and the minimum decay time of 72 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity subsequent to a fuel handling accident results in doses that are within the guideline values specified in 10 CFR 50.67 ~~100~~. The design basis for fuel handling accidents has historically separated the radiological consequences from the containment capability. The NRC staff has treated the containment capability for fuel handling conditions as a logical part of the "primary success path" to mitigate fuel handling accidents, irrespective of the assumptions used to calculate the radiological consequences of such accidents (Ref. 2).

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36.

LCO

This LCO reduces the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity from containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be capable of being closed. ~~except for the OPERABLE containment purge and exhaust penetrations.~~ For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are capable of being closed. ~~isolable by the RB purge isolation signal~~.

~~This LCO is modified by a note indicating that an emergency air lock door is not required to be closed when a temporary cover plate is installed.~~

APPLICABILITY

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

BASES (continued)

ACTIONS

A.1 and A.2

capable of being closed,

With the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere not ~~in the required status, including the Containment Purge and Exhaust Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the unit must be placed in a condition in which the isolation function is not needed. This is accomplished by~~ immediately suspend~~ing~~ CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude moving a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

is either in that position or is open under administrative control.

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position ~~is in that position.~~ Also the Surveillance will demonstrate that each open penetration's valve operator has motive power, which will ensure each valve is capable of being closed.

The Surveillance is performed every 7 days during the CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations.

As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

Delete

SR 3.9.3.2

This Surveillance demonstrates that each containment purge supply and exhaust isolation valve that is not locked, sealed or otherwise secured in the isolation position actuates to its isolation position on an actual or simulated high radiation signal. The frequency requires the isolation capability of the reactor building purge valves to be verified functional once each refueling outage prior to CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. This ensures that this

Delete

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.2 (continued)

function is verified prior to CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. This Surveillance 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.

REFERENCES

1. UFSAR, Section 15.11.
2. NRC letter to RG & E dated December 7, 1995, R.E. Ginna Nuclear Power Plant Conversion to Improved Standard Technical Specifications - Resolutions of Ginna Design Basis for Refueling Accidents.

5.5 Programs and Manuals

5.5.2 Containment Leakage Rate Testing Program (continued)

This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. Containment system visual examinations required by Regulatory Guide 1.163, Regulatory Position C.3 shall be performed as follows:

1. Accessible concrete surfaces and post-tensioning system component surfaces of the concrete containment shall be visually examined prior to initiating SR 3.6.1.1 Type A test. These visual examinations, or any portion thereof, shall be performed no earlier than 90 days prior to the start of refueling outages in which Type A tests will be performed. The validity of these visual examinations will be evaluated should any event or condition capable of affecting the integrity of the containment system occur between the completion of the visual examinations and the Type A test.

2. Accessible interior and exterior surfaces of metallic pressure retaining components of the containment system shall be visually examined at least three times every ten years, including during each shutdown for SR 3.6.1.1 Type A test, prior to initiating the Type A test.

Type B and C testing shall be implemented in the program in accordance with the requirements of 10 CFR 50, Appendix J, Option A.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 59 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.~~25~~.20% of the containment air weight per day.

Leakage rate acceptance ~~criteria are~~ criterion is:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

~~b. Leakage > 0.50 L_a shall be to the penetration room.~~

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5 Programs and Manuals (continued)

5.5.11 Secondary Water Chemistry

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.12 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2.

The VFTP is applicable to the ~~Penetration Room Ventilation System (PRVS), the~~ Control Room Ventilation System (CRVS) Booster Fan Trains, and the Spent Fuel Pool Ventilation System (SFPVS).

~~a. Demonstrate, for the PRVS, that a dioctyl phthalate (DOP) test of the high efficiency particulate air (HEPA) filters shows ≥ 99% removal when tested in accordance with ANSI N510-1975 at the system design flow rate ± 10%.~~

a. ~~b.~~ Demonstrate, for the CRVS Booster Fan Trains, that a DOP test of the HEPA filters shows ≥ 99.5% removal when tested at in accordance with ANSI N510-1975 at the system design flow rate ± 10%.

~~c.~~ b. ~~Demonstrate, for the PRVS, that a halogenated hydrocarbon test of the carbon adsorber shows ≥ 99% removal when tested in accordance with ANSI N510-1975 at the system design flow rate ± 10%.~~

5.5 Programs and Manuals

5.5.12 Ventilation Filter Testing Program (VFTP) (continued)

~~d.~~ b. Demonstrate, for the CRVS Booster Fan Trains, that a halogenated hydrocarbon test of the carbon adsorber shows ≥ 99% removal when tested at in accordance with ANSI N510-1975 at the system design flow rate ± 10%.

~~e.~~ c. Demonstrate, for the CRVS Booster Fan Trains, ~~PRVS and SFPVS~~, that a laboratory test of a sample of the carbon adsorber shows ≥ ~~90%~~ 97.5% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 (30°C, 95% RH).

~~f. Demonstrate, for the PRVS, that the pressure drop across the combined HEPA filters and carbon adsorber banks is < 6 in. of water at the system design flow rate ± 10%.~~

~~g.~~ d. Demonstrate, for the CRVS Booster Fan Trains, that the pressure drop across the pre-filter is ≤ 1 in. of water and the pressure drop across the HEPA filters is ≤ 2 in. of water at the system design flow rate ± 10%.

~~h. Demonstrate, for the SFPVS, that a dioctyl phthalate (DOP) test of the high efficiency particulate air (HEPA) filters shows ≥ 99% removal when tested in accordance with ANSI N510-1975 at the system design flow rate ± 10%.~~

~~i. Demonstrate, for the SFPVS, that a halogenated hydrocarbon test of the carbon adsorber shows ≥ 99% removal when tested in accordance with ANSI N510-1975 at the system design flow rate ± 10%.~~

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.13 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the waste gas holdup tanks and the quantity of radioactivity contained in waste gas holdup tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined. The liquid radwaste quantities shall be determined by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

ATTACHMENT 2

Proposed new Technical Specifications Pages with Insertion Instructions

Remove Page	Insert Page
3.3.6-1	3.3.6-1
B 3.3.6-2	B 3.3.6-2
B 3.3.5-1	B 3.3.5-1
B 3.3.5-2	B 3.3.5-2
B 3.3.5-6	B 3.3.5-6
B 3.3.7-2	B 3.3.7-2
3.7.10-1	-----------
3.7.10-2	----------
B 3.7.10-1	----------
B 3.7.10-2	----------
B 3.7.10-3	----------
B 3.7.10-4	----------
3.7.17-1	----------
3.7.17-2	----------
B 3.7.17-1	----------
B 3.7.17-2	----------
B 3.7.17-3	----------
3.9.3-1	3.9.3-1
3.9.3-2	----------
B 3.9.3-1	B 3.9.3-1
B 3.9.3-2	B 3.9.3-2
B 3.9.3-3	B 3.9.3-3
B 3.9.3-4	B 3.9.3-4
B 3.9.3-5	---------
5.0-8	5.0-8
5.0-21	5.0-21
5.0-22	5.0-22

3.3 INSTRUMENTATION

3.3.6 Engineered Safeguards Protective System (ESPS) Manual Initiation

LCO 3.3.6 Two manual initiation channels of each one of the ESPS Functions below shall be OPERABLE:

a. High Pressure Injection, Reactor Building (RB) Non-Essential Isolation, Keowee Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input (ES Channels 1 and 2);

b. Low Pressure Injection and RB Essential Isolation (ES Channels 3 and 4);

c. RB Cooling and RB Essential Isolation (ES Channels 5 and 6);

d. RB Spray (ES Channels 7 and 8).

APPLICABILITY: MODES 1 and 2,
MODES 3 and 4 when associated engineered safeguard equipment is required to be OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ESPS Functions with one channel inoperable.	A.1 Restore channel to OPERABLE status.	72 hours

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ESPS manual initiation ensures that the control room operator can rapidly initiate ES Functions. The manual initiation trip Function is required as a backup to automatic trip functions and allows operators to initiate ESPS whenever any parameter is rapidly trending toward its trip setpoint.

The ESPS manual initiation functions satisfy Criterion 3 of 10 CFR 50.36 (Ref. 1).

LCO

Two ESPS manual initiation channels of each ESPS Function shall be OPERABLE whenever conditions exist that could require ES protection of the reactor or RB. Two OPERABLE channels ensure that no single random failure will prevent system level manual initiation of any ESPS Function. The ESPS manual initiation Function allows the operator to initiate protective action prior to automatic initiation or in the event the automatic initiation does not occur.

The required Function is provided by two associated channels as indicated in the following table:

Function	Associated Channels
HPI and RB Non-Essential Isolation, Keowee Emergency Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input	1 & 2
LPI and RB Essential isolation	3 & 4
RB Cooling and RB Essential isolation	5 & 6
RB Spray	7 & 8

APPLICABILITY

The ESPS manual initiation Functions shall be OPERABLE in MODES 1 and 2, and in MODES 3 and 4 when the associated engineered safeguard equipment is required to be OPERABLE. The manual initiation channels are required because ES Functions are designed to provide protection in these MODES. ESPS initiates systems that are either reconfigured for decay heat removal operation or disabled while in MODES 5 and 6. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components. Adequate time is available to evaluate unit conditions and to respond by manually operating the ES components, if required.

B 3.3 INSTRUMENTATION

B 3.3.5 Engineered Safeguards Protective System (ESPS) Analog Instrumentation

BASES

BACKGROUND

The ESPS initiates necessary safety systems, based on the values of selected unit Parameters, to protect against violating core design limits and to mitigate accidents.

ESPS actuates the following systems:

- High pressure injection (HPI);
- Low pressure injection (LPI);
- Reactor building (RB) cooling;
- RB Spray;
- RB Isolation; and
- Keowee Hydro Unit Emergency Start.

The ESPS operates in a distributed manner to initiate the appropriate systems. The ESPS does this by determining the need for actuation in each of three analog channels monitoring each actuation Parameter. Once the need for actuation is determined, the condition is transmitted to digital automatic actuation logic channels, which perform the two-out-of-three logic to determine the actuation of each end device. Each end device has its own automatic actuation logic, although all digital automatic actuation logic channels take their signals from the same bistable in each channel for each Parameter.

Four Parameters are used for actuation:

- Low Reactor Coolant System (RCS) Pressure;
- Low Low RCS Pressure;
- High RB Pressure; and
- High High RB Pressure.

BASES

BACKGROUND (continued)

LCO 3.3.5 covers only the analog instrumentation channels that measure these Parameters. These channels include all intervening equipment necessary to produce actuation before the measured process Parameter exceeds the limits assumed by the accident analysis. This includes sensors, bistable devices, operational bypass circuitry, and output relays. LCO 3.3.6, "Engineered Safeguards Protective System (ESPS) Manual Initiation," and LCO 3.3.7, "Engineered Safeguards Protective System (ESPS) Digital Automatic Actuation Logic Channels," provide requirements on the manual initiation and digital automatic actuation logic Functions.

The ESPS contains three analog channels. Each analog channel provides input to digital logic channels that initiate equipment with a two-out-of-three logic on each digital logic channel. Each analog channel includes inputs from one analog instrumentation channel of Low RCS Pressure, Low Low RCS Pressure, High RB Pressure, and High High RB Pressure. Digital automatic actuation logic channels combine the three analog channel trips to actuate the individual Engineered Safeguards (ES) components needed to initiate each ES System. Figure 7.5, UFSAR, Chapter 7 (Ref. 1), illustrates how analog instrumentation channel trips combine to cause digital logic channel trips.

The following matrix identifies the analog instrumentation (measurement) channels and the Digital Automatic Actuation Logic Channels actuated by each.

Digital Logic Channels	Actuated Systems/ Functions	RCS PRESS LOW	RCS PRESS LOW LOW	RB PRESS HIGH	RB PRESS HIGH HIGH
1 and 2	HPI and RB Non-Essential Isolation, Keowee Emergency Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input	X		X	
3 and 4	LPI and RB Essential isolation		X	X	
5 and 6	RB Cooling and RB Essential isolation			X	
7 and 8	RB Spray				X

The ES equipment is generally divided between the two redundant digital actuation logic channels. The division of the equipment between the two digital actuation logic channels is based on the equipment redundancy and

BASES

APPLICABLE SAFETY ANALYSES (continued)

Reactor Building Spray, Reactor Building Cooling, and Reactor Building Isolation

The ESPS actuation of the RB coolers and RB Spray have been credited in RB analysis for LOCAs, both for RB performance and equipment environmental qualification pressure and temperature envelope definition. Accident dose calculations have credited RB Isolation and RB Spray.

Keowee Hydro Unit Emergency Start

The ESPS initiated Keowee Hydro Unit Emergency Start has been included in the design to ensure that emergency power is available throughout the limiting LOCA scenarios.

The small break LOCA analyses assume a conservative 48 second delay time for the actuation of HPI and LPI in UFSAR, Chapter 15 (Ref. 4). The large break LOCA analyses assume LPI flow starts in 38 seconds while full LPI flow does not occur until 15 seconds later, or 53 seconds total (Ref. 4). This delay time includes allowances for Keowee Hydro Unit starting, Emergency Core Cooling Systems (ECCS) pump starts, and valve openings. Similarly, the RB Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system analyzed.

Accident analyses rely on automatic ESPS actuation for protection of the core temperature and containment pressure limits and for limiting off site dose levels following an accident. These include LOCA, and MSLB events that result in RCS inventory reduction or severe loss of RCS cooling.

The ESPS channels satisfy Criterion 3 of 10 CFR 50.36 (Ref. 5).

LCO

The LCO requires three analog channels of ESPS instrumentation for each Parameter in Table 3.3.5–1 to be OPERABLE in each ESPS digital automatic actuation logic channel. Failure of any instrument renders the affected analog channel(s) inoperable and reduces the reliability of the affected Functions.

Only the Allowable Value is specified for each ESPS Function in the LCO. Nominal trip setpoints are specified in the setpoint calculations. The nominal trip setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is

BASES

BACKGROUND (continued)

includes allowances for Keowee Hydro Unit startup and loading, ECCS pump starts, and valve openings. Similarly, the reactor building (RB) Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system.

The ESPS automatic initiation of Engineered Safeguards (ES) Functions to mitigate accident conditions is assumed in the accident analysis and is required to ensure that consequences of analyzed events do not exceed the accident analysis predictions. Automatically actuated features include HPI, LPI, RB Cooling, RB Spray, and RB Isolation.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on automatic ESPS actuation for protection of the core and RB and for limiting off site dose levels following an accident. The digital automatic actuation logic is an integral part of the ESPS.

The ESPS digital automatic actuation logic channels satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

The digital automatic actuation logic channels are required to be OPERABLE whenever conditions exist that could require ES protection of the reactor or the RB. This ensures automatic initiation of the ES required to mitigate the consequences of accidents.

The required Function is provided by two associated digital channels as indicated in the following table:

Function	Associated Channels
HPI and RB Non-Essential Isolation, Keowee Emergency Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input	1 & 2
LPI and RB Essential isolation	3 & 4
RB Cooling and RB Essential isolation	5 & 6
RB Spray	7 & 8

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch is capable of being closed;

b. One door in each air lock is capable of being closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere is capable of being closed.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS <u>AND</u> A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately Immediately

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required containment penetration is in the required status.	7 days

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. In order to make this distinction, the penetration requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that specified escape paths are closed or capable of being closed. Since there is no significant potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained within the requirements of 10 CFR 50.67. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be capable of being closed.

During movement of irradiated fuel assemblies within containment, administrative controls shall ensure that appropriate personnel are aware that the equipment hatch is open, that a specific individual(s) is designated and available to close the equipment hatch cover following a required evacuation of containment, and that any obstruction(s) (e.g., cables and hoses) that could prevent closure of the equipment hatch cover be capable of being quickly removed. Should a fuel handling accident occur inside containment, the equipment hatch will be closed following evacuation. For closure, the equipment hatch cover will be in place with a minimum of four bolts securing the cover to the sealing

BASES

BACKGROUND (continued)

surface. Good engineering practice dictates that the bolts required be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment OPERABILITY is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment ingress and egress is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure requires that one door in each air lock be capable of being closed. The door interlock mechanism may remain disabled.

During movement of irradiated fuel assemblies within containment, administrative controls shall ensure that appropriate personnel are aware that both personnel airlock doors are open, that a specific individual(s) is designated and available to close the airlock door following a required evacuation of containment, and any obstruction (s)(e.g., cables and hoses) that could prevent closure of an airlock door be capable of being quickly removed. Should a fuel handling accident occur inside containment, at least one of the personnel and/or emergency air lock doors will be closed following evacuation.

The requirements on containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted to within regulatory limits.

The Reactor Building Purge System includes a supply penetration and exhaust penetration. During MODES 1, 2, 3, and 4, two valves in each of the supply and exhaust penetrations are secured in the closed position. The system is not subject to a Specification in MODE 5.

In MODE 6, large air exchanges are necessary to support refueling operations. The purge system is used for this purpose.

During movement of irradiated fuel assemblies within containment, administrative controls shall ensure that appropriate personnel are aware that containment isolation valves are open, that a specific individual (s) is designated and available to close the valve following a required evacuation of containment, and any obstruction (s) (e.g., cables and hoses) that could prevent closure of a containment isolation valve be

BACKGROUND (continued)

capable of being quickly removed. Should a fuel handling accident occur inside containment, containment isolation valves will be closed on at least one side. Isolation may be achieved by a closed automatic isolation valve, non-automatic power operated valve, manual isolation valve, blind flange, or equivalent.

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of 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. 2). A minimum fuel transfer canal water level and the minimum decay time of 72 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity subsequent to a fuel handling accident results in doses that are within the guideline values specified in 10 CFR 50.67. The design basis for fuel handling accidents has historically separated the radiological consequences from the containment capability. The NRC staff has treated the containment capability for fuel handling conditions as a logical part of the "primary success path" to mitigate fuel handling accidents, irrespective of the assumptions used to calculate the radiological consequences of such accidents (Ref. 2).

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36.

LCO

This LCO reduces the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity from containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be capable of being closed. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are capable of being closed.

APPLICABILITY

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

BASES (continued)

ACTIONS

A.1 and A.2

With the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere not capable of being closed, immediately suspend CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude moving a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is either in that position or is open under administrative control. Also the Surveillance will demonstrate that each open penetration's valve operator has motive power, which will ensure each valve is capable of being closed.

The Surveillance is performed every 7 days during the CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations.

As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

REFERENCES

1. UFSAR, Section 15.11.

2. NRC letter to RG & E dated December 7, 1995, R.E. Ginna Nuclear Power Plant Conversion to Improved Standard Technical Specifications - Resolutions of Ginna Design Basis for Refueling Accidents.

5.5 Programs and Manuals

5.5.2 Containment Leakage Rate Testing Program (continued)

This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. Containment system visual examinations required by Regulatory Guide 1.163, Regulatory Position C.3 shall be performed as follows:

1. Accessible concrete surfaces and post-tensioning system component surfaces of the concrete containment shall be visually examined prior to initiating SR 3.6.1.1 Type A test. These visual examinations, or any portion thereof, shall be performed no earlier than 90 days prior to the start of refueling outages in which Type A tests will be performed. The validity of these visual examinations will be evaluated should any event or condition capable of affecting the integrity of the containment system occur between the completion of the visual examinations and the Type A test.

2. Accessible interior and exterior surfaces of metallic pressure retaining components of the containment system shall be visually examined at least three times every ten years, including during each shutdown for SR 3.6.1.1 Type A test, prior to initiating the Type A test.

Type B and C testing shall be implemented in the program in accordance with the requirements of 10 CFR 50, Appendix J, Option A.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 59 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of the containment air weight per day.

Leakage rate acceptance criterion is:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5 Programs and Manuals (continued)

5.5.11 Secondary Water Chemistry

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.12 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2.

The VFTP is applicable to the Control Room Ventilation System (CRVS) Booster Fan Trains.

a. Demonstrate, for the CRVS Booster Fan Trains, that a DOP test of the HEPA filters shows $\geq$ 99.5% removal when tested at in accordance with ANSI N510-1975 at the system design flow rate $\pm$ 10%.

b. Demonstrate, for the CRVS Booster Fan Trains, that a halogenated hydrocarbon test of the carbon adsorber shows $\geq$ 99% removal when tested at in accordance with ANSI N510-1975 at the system design flow rate $\pm$ 10%.

5.5 Programs and Manuals

5.5.12 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate, for the CRVS Booster Fan Trains that a laboratory test of a sample of the carbon adsorber shows ≥ 97.5% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 (30°C, 95% RH).

d. Demonstrate, for the CRVS Booster Fan Trains, that the pressure drop across the pre-filter is ≤ 1 in. of water and the pressure drop across the HEPA filters is ≤ 2 in. of water at the system design flow rate ± 10%.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.13 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the waste gas holdup tanks and the quantity of radioactivity contained in waste gas holdup tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined. The liquid radwaste quantities shall be determined by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

The program shall include:

a. The limit for concentration of hydrogen in the waste gas holdup tanks and a surveillance program to ensure the limit is maintained. The limit shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each waste gas holdup tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual at the nearest exclusion area boundary, in the event of an uncontrolled release of the tank's contents.

ATTACHMENT 3

Description of the Proposed Changes and Technical Justification for Oconee Nuclear Station

INTRODUCTION

This submittal describes the evaluations conducted to assess the radiological consequences of implementing the alternative source term (AST) methodology for the Oconee Nuclear Station. The accident source term is a significant aspect of the design and licensing basis of a plant. As an input to the accident analyses that form the basis for the design and operation of the unit, a change in the source term can impact both the postulated accident consequences and the margin of safety. For this reason, the NRC has determined that any change to the design basis to use an accident source term that is different from the accident source term used in the original design and licensing of the facility should be reviewed and approved by the NRC in the form of a license amendment.

REASON FOR THE REQUEST

There are two reasons for this request. The first is to implement the AST rule in accordance with 10 CFR 50.67 and the relevant guidance provided in Regulatory Guide 1.183. The second is to request approval of the AST analysis methodology for the Oconee Station that will justify simplification of the Ventilation System testing requirements and amend secondary containment operability requirements during core alterations or fuel movement. The changes include the following revisions:

- The Penetration Room Ventilation System (PRVS) is removed from Technical Specifications (TS) because the PRVS filtration will not be credited in licensing analyses that determine Control Room and off-site doses.
- The Spent Fuel Pool Ventilation System (SFPVS) is removed from TS because the SFPVS will not be credited in licensing analyses that determine Control Room and off-site doses.
- During certain refueling operations, the containment air locks and/or the equipment hatch and penetrations providing

direct access from the containment atmosphere to the outside atmosphere will be permitted to be unisolated under administrative controls. Additionally, the requirement to maintain an operable automatic isolation capability for the Reactor Building Purge System during refueling is being removed from TS.

- The allowable value for the Reactor Building leakage rate is lowered from 0.25 w%/day to 0.20 w%/day.
- The requirement to measure Reactor Building leakage in excess of 50% of L_a to the penetration room is removed from TS.
- The Ventilation Filter Testing Program (VFTP) is revised to remove all references to the PRVS and SFPVS and their testing requirements.
- The VFTP acceptance criterion for the CRVS Booster Fan trains is revised to require $\geq$ 97.5% radioactive methyl iodide removal.

This submittal represents a full-scope implementation of the new source term. Design Basis Accident analyses, specifically, the Maximum Hypothetical Accident (MHA) and the Fuel Handling Accident (FHA) have been revised to define the impact of the new source term on doses to the public at the site boundary and to the operator in the Control Room.

DISCUSSION OF CHANGES

TS 3.3.6 Engineered Safeguards Protective System (ESPS) Manual Initiation

The ESPS initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and mitigate accidents. PRVS is one of the systems that is actuated by the ESPS. Since the PRVS will not be credited for Control Room and off-site doses based on the revised radiological analyses of the MHA, the PRVS is being removed as an ESPS function from TS 3.3.6.

B 3.3.6 Engineered Safeguards Protective System (ESPS) Manual Initiation

This section is being revised to remove all references to the PRVS based on the aforementioned discussion.

B 3.3.5 Engineered Safeguards Protective System (ESPS) Analog Instrumentation

This section is also affected by the removal of the PRVS from the ESPS and is being revised to remove all references to the PRVS.

B 3.3.7 Engineered Safeguards Protective System (ESPS) Digital Automatic Actuation Logic Channels

This section is also affected by the removal of the PRVS from the ESPS and is being revised to remove all references to the PRVS.

TS 3.7.10 Penetration Room Ventilation System (PRVS)

This section is being removed from TS. The PRVS will not be credited for evaluating potential Control Room and off-site doses. This change results in an operational efficiency that is achievable from implementing the AST. The revised radiological analyses of the MHA are performed without taking credit for the PRVS filter system and the results of this analysis show that the offsite and Control Room doses remain below the Regulatory Guide 1.183 limits. Removal of this system from TS eliminates the requirement to demonstrate the effectiveness of this system in operation. This simplifies testing design and performance tasks.

B 3.7.10 Penetration Room Ventilation System (PRVS)

The bases for TS 3.7.10 is also being removed from TS.

TS 3.7.17 Spent Fuel Pool Ventilation System (SFPVS)

This section is being removed from TS. The SFPVS will not be credited for evaluating potential Control Room and off-site doses. This change results in an operational efficiency that is achievable from implementing the AST. The revised radiological analyses of the FHA are performed without taking credit for the SFPVS filter system and the results of this analysis show that the offsite and Control Room doses remain below the Regulatory Guide 1.183 limits.

B 3.7.17 Spent Fuel Pool Ventilation System (SFPVS)

The bases for TS 3.7.17 is also being removed from TS.

TS 5.5.2 Containment Leakage Rate Testing Program

The revised radiological analysis of the MHA takes no credit for the PRVS filter system. The existing MHA dose analysis assumes a reactor building leakage rate of 0.25 w%/day for the first 24 hours. This is followed by 0.125 w%/day for the duration of the accident. In addition, 50% of the 0.25 w%/day leakage passes through the PRVS filtration system. The remaining 50% of the 0.25 w%/day is assumed to pass directly to the atmosphere.

However, the revised MHA analysis assumes a Reactor Building leakage rate of 0.20 w%/day for the first 24 hours. Following the guidance provided in Regulatory Guide 1.183, this rate is then reduced to 0.1 w%/day for the duration of the accident.

Additionally, as described in the analysis modifications, it is assumed that the PRVS bypass is 100 percent. Without taking credit for PRVS filtration, the doses are still within the limits of 10 CFR 50.67. This eliminates the need to demonstrate the effectiveness of the PRVS filtration.

TS 5.5.12 Ventilation Filter Testing Program (VFTP)

The revised MHA analysis assumes that the filter efficiencies for the Control Room intake iodine filters are 99% for particulate, 95% for organic, and 99% for elemental iodine. Based on this assumption, the test acceptance criterion for radioactive methyl iodide removal for the CRVS Booster Fan trains is changed from ≥ 90% to ≥ 97.5%.

TS 3.9.3 Containment Penetrations

This change is another operational enhancement and cost reduction that results from the implementation of the AST. This change revises the requirements for containment closure during fuel movement and refueling operations. The revised analysis of the FHA inside Containment using an AST supports the proposed TS change.

The Oconee containment is equipped with two containment air locks and an equipment hatch. Presently, TS 3.9.3 requires that a minimum of one door in each of the containment air locks, as well as other containment penetrations, including the equipment hatch, are closed during core alterations or movement of irradiated fuel within the Reactor Containment Building. The purpose of this requirement is to mitigate the consequences of a fuel handling accident inside containment. Current FHA analysis assumptions are reflected in Section 15.11 of the Oconee Updated Final Safety Analysis Report (UFSAR). Duke has revised the analysis for the FHA utilizing AST methodology in accordance with 10 CFR 50.67 and Regulatory Guide 1.183. The revised analysis assumes that the containment air lock doors and other penetrations, including the equipment hatch, are open at the time of the accident.

The LCO for TS 3.9.3 is being revised to allow the containment air lock doors, the equipment hatch and each penetration providing direct access from the containment atmosphere to the outside atmosphere to remain open during fuel movement and refueling operations. The proposed TS would require that during fuel movement and refueling operations the equipment hatch, at

least one door in each air lock, and any penetration providing direct access from the containment atmosphere to the outside atmosphere are each capable of being closed.

During the movement of irradiated fuel, administrative controls will be in place to assure that appropriate personnel are aware that both personnel air lock doors and the equipment hatch are open, that a specific individual(s) is designated and available to close an airlock door and the equipment hatch cover following a required evacuation of containment, and that any obstructions(s) (e.g., cables and hoses) that could prevent closure of the equipment hatch cover or an airlock door be capable of being removed quickly. Should a FHA occur inside containment, at least one of the personnel and/or emergency airlock doors and the equipment hatch will be closed following evacuation of the Reactor Building. For closure, the equipment hatch cover will be in place with a minimum of four bolts securing the cover to the sealing surface. These administrative control requirements are identified in the proposed change.

Additionally, the methods of isolation for penetrations providing direct access from the containment atmosphere to the outside atmosphere are being removed from the LCO and are being incorporated into the bases of the TS. The existing note concerning the emergency air lock door is being removed. Administrative controls will ensure that appropriate personnel are aware of the open status of the penetration flow path(s) during core alterations or movement of irradiated fuel assemblies within containment, and specified individuals are designated and readily available to isolate the flow path in the event of a FHA. Isolation of the penetration flow path(s) may be achieved by a closed automatic isolation valve, non-automatic power operated valve, manual isolation valve, blind flange or equivalent.

Surveillance Requirement 3.9.3.2 is also being removed from TS. This change involves eliminating the requirement to maintain an operable automatic isolation capability for the Reactor Building Purge system while retaining operability requirements for the radiation monitors (to provide fuel handling accident

identification) during refueling. Automatic isolation occurs in response to high radiation signals from containment area and airborne radiation monitors. Currently, this surveillance requires testing of this system and the radiation monitors immediately prior to refueling. The revised FHA analysis inside containment assumes failure of the purge isolation function and therefore already models radioactive releases through the purge pathway. This change is proposed to provide flexibility in refueling operations. The revised FHA analysis takes no credit for operation of the purge isolation function.

During a refueling outage, additional work inside the containment continues during fuel movement and core alterations, requiring frequent cycling of the containment air lock doors in order to enter and exit containment. Repeated cycling and heavy use of the containment air lock doors necessitates substantial maintenance on the containment air lock doors and components. Thus, leaving the air lock doors open should increase the reliability and lifetime of the air locks. Other mechanical penetrations are utilized to provide access for equipment and communications cables during a refueling outage. The proposed change to leave the air lock doors and the equipment hatch open during core alterations or movement of irradiated fuel would increase personnel and emergency air lock door reliability, provide greater efficiency in the movement of personnel and equipment, and result in decreased outage critical path time. This results in significant cost savings over the life of the plant. It will also result in reductions in radiation worker dose for routine tasks requiring containment access during these periods.

Should a FHA occur, it would take a number of cycles of the personnel and emergency air lock doors to evacuate personnel from within containment. This would delay the evacuation of personnel from containment. Containment could be evacuated more expeditiously with the personnel and emergency air lock doors open and with the equipment hatch open than with these doors closed, thereby enhancing personnel safety. This would reduce dose to the workers in the event of an accident while maintaining acceptable doses to the public.

This change will improve the reliability of the personnel and emergency air lock doors, improve personnel safety, and allow more efficient plant refueling outages.[1]

B3.9.3 Containment Penetrations

The Bases for TS 3.9.3 are being revised to designate the penetrations to be open during fuel movement along with the administrative controls applicable to these conditions.

PLANT MODIFICATIONS

Plant modifications will be implemented in concert with the implementation of the AST and the changes to the plant controls described above. These modifications include:

Control Room Outside Air Intake Relocation

The existing Control Room outside air intakes for the shared Unit 1 & 2 Control Room and the separate Unit 3 Control Room will be relocated. Control Room intakes will be moved from the roof of the Auxiliary Building to the roof of the Turbine Building. Dual intakes will be installed for each Control Room. Intakes will be located at the northeast and southeast corners of the Turbine Building for each Control Room.

---

[1] The NRC has approved a similar change for Arkansas Nuclear One, Unit 1 on September 20,1996 (Amendment No. 184) and Unit 2 on September 28,1995 (Amendment No. 166) for personnel and emergency air lock doors. Additionally, the NRC approved a similar change for Arkansas Nuclear One, Units 1 and 2, on April 16, 1999 (Amendment Nos. 195 and 203, respectively) for the equipment hatch. NUREG 1430, Revision 2 (STS for Babcock and Wilcox Plants) was also used as guidance in proposing this change.

Oconee presently has one outside air intake for each Control Room that is located on the roof of the Auxiliary Building, approximately 75 feet from the base of the unit vent. Due to the close proximity of the potential release points to the air intake, there is a significant contribution to the calculated post-accident operator dose from unit vent releases. A detailed sensitivity study of the impact of intake locations was performed in order to determine the dose impact. The degree of dispersion, and hence the amount of contamination present in the pressurization air, is modeled in the dose calculation using the ARCON96 dispersion modeling program. Current NRC guidance relating to the application of ARCON96 has been used. Moving the intake farther from the unit vent and providing multiple intakes reduces the potential Control Room operator dose.

Caustic Addition System

The existing active Caustic Addition System will be replaced with a passive Caustic Addition System. The existing Caustic Addition System is used to add caustic to the Emergency Sump during post-LOCA conditions in order to raise the pH of sump water injected into the reactor vessel by ECCS injection systems, and sprayed into the reactor containment by the Reactor Building Spray System. The current dose analysis assumes that caustic will be injected into the Emergency Sump within 30 minutes of shifting to the Recirculation mode of emergency cooling operation. Additionally, the dose analysis assumes that the pH will be raised to a pH level between 7.0 - 8.0 within 24 hours after shifting to the Recirculation mode of emergency cooling operation. Raising the pH of the injection and Reactor Building Spray water will decrease the amount of iodine re-volatilization during accident conditions, inhibit the generation of hydrogen gas, and reduce the likelihood of stress assisted corrosion of susceptible components in contact with borated water.

The existing system has limited capacity to add caustic in a timely manner. Dose analysis results are improved when the sump

pH is normalized to a value above 7.0 at an earlier time. The existing system requires Operator action outside the Control Room to control sump pH following an accident.

The new passive Caustic Addition System is a series of baskets that will contain solid caustic or tri-sodium phosphate. The chemicals will dissolve as the water level in the Reactor Building basement increases following a LOCA. The vertical sides of each basket will be constructed of a stainless steel mesh, which will prevent the loss of chemicals during normal plant operation, but will allow water to dissolve the chemicals. The top and bottom of each basket will consist of stainless steel plates. A sufficient number of baskets will be positioned in areas of the Reactor Building basement to achieve the desired sump water pH following a LOCA.

The new system will add caustic to the Emergency Sump at the beginning of the event. This feature is beneficial in that it raises the pH sooner, thereby minimizing the dose consequences by eliminating re-suspension of iodine subsequent to swapover. The new system will be QA-1 and seismically qualified. The passive design of the system will remove the need for Operator action to control post-accident pH and the design will not be susceptible to single active failures.

The chief difference between the existing active Caustic Addition System and the new passive Caustic Addition System is the passive design feature. That is, the new system will perform the same function of assuring that sump pH is controlled as assumed in the dose analysis. This function is presently controlled by the Selected Licensee Commitment (SLC) process. The affected SLC will be revised to reflect the controls necessary to implement the new passive design.

High Pressure Injection/Low Pressure Injection (HPI/LPI) Relief Valve Discharge Re-route

The Letdown Storage Tank (LDST) relief valve, HP-79, currently relieves to the RC Bleed Hold-up tank, which is located in the Auxiliary Building. A modification will be implemented to route the discharge of the HP-79 back to the Reactor Building Emergency Sump (RBES). A four inch line will be routed from the discharge port of HP-79 to a new four inch drain header line that will tap into the drain line that drains the RBES through penetration 40. The new four inch drain header line will contain two check valves to prevent RBES drain water from entering the line. These check valves will serve as the containment isolation valves. This modification will eliminate the potential for sump fluid leaking to the LDST, lifting the LDST relief valve and entering the Auxiliary Building.

RADIOLOGICAL EVENT RE-ANALYSIS AND EVALUATION

Regulatory Guide 1.183 requires that the DBA LOCA be re-analyzed, at a minimum, for full implementation of the AST. Oconee has also re-analyzed the FHA accident for this License Amendment Request. The Oconee design basis MHA currently described in the ONS UFSAR Section 15.15 represents the limiting design basis accident dose consequences for Oconee Units 1, 2 and 3. This is evident upon examination of the source term released for each accident analyzed in the UFSAR as compared to that of the LOCA. The extent of core inventory released during a LOCA is higher than for all other accidents. In addition, the atmospheric dispersion factors (χ/Qs) for all accidents are reduced due to the relocation of control room intakes at Oconee planned as part of the plant modifications. A system of dual Control Room intake locations will be installed, further reducing these χ/Qs by a factor of two.

This section describes the re-analysis of the design basis radiological analyses for the MHA and the FHA. These analyses

have incorporated the features of the AST, including the Total Effective Dose Equivalent (TEDE) analysis methodology and modeling of plant systems and equipment operation that influence the events. The calculated radiological consequences are compared with the revised limits provided in 10 CFR 50.67(b)(2), and as specified per the additional guidance in Regulatory Guide 1.183 for the FHA event. Dose calculations are performed for the exclusion area boundary (EAB) for the worst 2-hour period, for the low population zone (LPZ) and Control Room for the duration of the accident (30 days). All the radiological consequence calculations for the AST were performed by Duke Energy with the LOCADOSE computer code system.[2] Bechtel Corporation developed the LOCADOSE codes for the purpose of analyzing doses from transport of radioactive materials through multi-region systems. The dose acceptance criteria that apply for implementing the AST are provided in the table below.

Accident Dose Acceptance Criteria

Accident or Case	Control Room	EAB & LPZ
Design Basis LOCA	5 rem TEDE	25 rem TEDE
Fuel Handling Accident	5 rem TEDE	6.3 rem TEDE

Revised MHA Analysis

An evaluation of the postulated MHA at Oconee Nuclear Station has been performed utilizing AST methodology in accordance with 10 CFR 50.67 and Regulatory Guide 1.183.

[2] Alternative Source term analyses using the LOCADOSE computer code have been submitted previously to the NRC for review in licensing submittals for the Surry Nuclear Power Station. Licensing submittals using LOCADOSE in TID source term applications have been made in support of Palo Verde, San Onofre, and Diablo Canyon Nuclear Power Stations.

This analysis calculates the potential offsite and Control Room doses due to the inhalation of and exposure to contaminated air from containment leakage and RBES leakage. The two types of leakage are modeled separately in order to properly model potential iodine re-volatilization for the containment model, while maintaining proper conservatism for the RBES leakage. The inputs and assumptions that are necessary to model release, transport, and removal of the contaminants follow. The following section presents data involving source term generation and offsite/Control Room transport/deposition inputs that are common for both the containment and RBES models. The subsequent sections present data that is model specific and will vary depending on the type of release.

Source Term Production and Receptor Inputs

The bounding source term for the MHA analysis is based on assumptions specified in Regulatory Guide 1.183. The inventory of fission products in the reactor core and available for release is based on a reactor power of 2620 MWt. This is 102% of the rated value of 2568 MWt.

Data for the calculation of Control Room doses are taken from Regulatory Guide 1.183 unless otherwise noted:

The 0-2 hour Exclusion Area Boundary atmospheric dispersion factor is 2.2E-4 sec/m^3

The Low Population Zone atmospheric dispersion factors are:

0 - 8 hours	2.35E-5 sec/m^3
8 - 24 hours	4.70E-6 sec/m^3
1 - 4 days	1.50E-6 sec/m^3
4 - 30 days	3.30E-7 sec/m^3

Breathing rates used for calculation of offsite doses are:

0 - 8 hours	3.5E-4 m^3/sec
8 - 24 hours	1.8E-4 m^3/sec
1 - 30 days	2.3E-4 m^3/sec

The breathing rate for Control Room Operators is 3.5E-4 m^3/sec for the duration of the accident.

Control Room occupancy factors during the 30-day post-accident period are:

0 to 24 hours = 100%
1 to 4 days = 60%
4 to 30 days = 40%

Based on results from Oconee specific Tracer Gas testing performed in 1998, the following values were used for Control Room unfiltered inleakage:

Units 1 & 2	prior to fan start:	1150 cfm
	after fan start:	150 cfm
Unit 3	prior to fan start:	600 cfm
	after fan start:	100 cfm

It is assumed that the Control Room Operators start the Control Room Ventilation System Booster Fans within 30 minutes of the accident.

Control Room volumes are calculated to be 8.64E+4 cubic feet for the Unit 1 & 2 Control Room and 4.32E+4 cubic feet for the Unit 3 Control Room.

The Emergency Control Room Ventilation Booster Fan intake flowrate is 1350 $\pm$ 135 cfm in each train (2700 $\pm$ 270 cfm in both trains). Since the lower flowrate yields a lower iodine protection factor, the flow is calculated to be 1215 cfm (one train at the minimum flow rate) for both Control Rooms. In this analysis, only one fan is assumed to start when activated by Control Room personnel.

The assumed filter efficiencies for the Control Room intake iodine filters are 99%, for particulate, 95% for organic, and 99% for elemental.

The following atmospheric dispersion factors were calculated for the Control Room radioactivity transport: The bounding (maximum) χ/Q values for all units will be used for each release type. Used in conjunction, the proposed northeast and southeast Control Room intake locations will present a system of dual intakes without manual selection. A factor of 2 reduction in the most limiting χ/Q value will be applied to account for dilution effects associated with a dual inlet configuration. Therefore, the χ/Q values are reduced by a factor of 2 for use in LOCADOSE modeling.

Time (hours)	Equipment Hatch (sec/m^3)	BWST (sec/m^3)	Unit Vent (sec/m^3)
0-8	2.60E-4	1.57E-4	3.34E-4
8-24	1.04E-4	6.05E-4	1.27E-4
24-96	7.80E-5	4.72E-5	9.95E-5
96-720	6.10E-5	3.70E-5	8.05E-5

Containment Release Model

The containment model simulates mixing between the sprayed and unsprayed portions of containment, releases from containment to the penetration room, and containment leakage that bypasses the penetration room and enters the environment directly.

For the first 24 hours the containment leak rate is determined to be 0.2% per day based on the proposed TS. Following Regulatory Guide 1.183, this rate is reduced to 0.1% per day for the remainder of the accident.

In accordance with Regulatory Guide 1.183, the chemical form of radioiodine released into containment is 4.85% elemental, 0.15% organic, and 95% particulate.

No credit is taken for the PRVS filters. It is assumed that 100% of containment leakage bypasses these filters and is released directly to the environment.

The volume of the unsprayed containment is calculated to be 9.17E+5 cubic feet and the sprayed containment volume is 8.66E+5 cubic feet.

Mixing between the sprayed and unsprayed portions of containment is calculated as represented by 30,500 cfm, which is based on an air exchange rate of 2 unsprayed volumes per hour.

The modeling of containment spray is based on Oconee Nuclear Station calculations for Post-Accident Iodine Re-volatilization Analysis and for Post-Accident Containment Atmosphere Iodine Spray Removal Analysis. Elemental iodine spray lambdas were selected in order to produce sprayed and unsprayed iodine inventories that match the stated references. Particulate iodine spray lambdas were calculated as:

Start Time	End Time	Particulate Iodine Spray Lambda (hr^{-1})
1.6 minutes	25 minutes	9.70
25 minutes	720 hours	6.73 initially. Reduces to 0.673 when overall atmospheric concentration falls below 2% of initial

In addition to the removal of radioactivity by the Reactor Building Spray System, Regulatory Guide 1.183 allows for removal by natural processes. A deposition model based on methodology described in NUREG/CR-6189 is included in the LOCADOSE containment model.

RBES Release Model

The RBES model simulates leakage from the emergency sump into the Borated Water Storage Tank (BWST), and Auxiliary Building.

The model follows the guidance presented in Regulatory Guide 1.183.

In accordance with Regulatory Guide 1.183, the chemical form of radioiodine released is 97% elemental and 3% organic.

The minimum initial volume of water in the sump post-LOCA was calculated to be 48,100 cubic feet at a reference temperature of 150° F. This volume was then reduced with each time step to account for sump volume loss due to release paths.

Based on Oconee SLC 16.6.4 and Regulatory Guide 1.183, a value of 4 gph is used for the ECCS leakage to the Auxiliary Building for the sump models. Oconee SLC 16.6.4 states that the maximum allowable leakage from the LPI System components shall not exceed two gallons per hour, while Regulatory Guide 1.183 requires that the specified leakage is increased by a factor of two for conservatism. A flashing value for this fluid of 10% is used in the analysis, since Regulatory Guide 1.183 specifies 10% to be used in cases where the fluid temperature is greater than 212° F and when the calculated value is less than 10%. For this analysis, the calculated flash fraction was less than 10%, so a value equal to 10% was used.

BWST flow characteristics are based on time-dependent iodine release models that produce a partition factor based on the chemical properties of the leaking fluid. ECCS backleakage from the sump to the BWST was set to 5 gpm, with no refill. For the accident, the no refill option is the most limiting because it reduces the amount of time the system is in the injection mode. The average BWST release rates to the environment are modeled in LOCADOSE by using a filtered node with filter efficiencies set to (1 - release ratio). This release ratio represents the amount of iodine which partitions into the BWST vapor space.

Results

The results for the MHA analysis are presented below. The doses are within the prescribed limits set forth in 10 CFR 50.67 and

Regulatory Guide 1.183. These are 5 REM TEDE for the Control Room and 25 REM TEDE for the EAB and LPZ. The Control Room dose is for the Unit 1&2 Control Room and is more limiting than the dose calculated for the Unit 3 Control Room. For clarity, only the most limiting value will be reported.

	Containment Model	RBES Model	Total rem TEDE
EAB	9.2	0.1	9.3
LPZ	4.7	0.1	4.8
Control Room	2.4	0.7	3.1

Revised Fuel Handling Accident (FHA) Analysis

An evaluation of the postulated FHA at Oconee Nuclear Station has been performed utilizing AST methodology in accordance with 10 CFR 50.67 and following the guidance of Regulatory Guide 1.183. This calculation evaluates possible dose consequences to the Control Room operators as a result of FHA in the Spent Fuel Pool (SFP) area or in containment. No credit is taken for containment integrity in terms of containment isolation. No credit is taken for the Reactor Building Purge Exhaust System filtration prior to release to the environment. For SFP building accidents, no credit is taken for the SFP Ventilation System filtration prior to release to the environment.

This calculation does not take credit for any radioactive decay during the transit time from a SFP or transfer canal to a Control Room. This is conservative. The source terms used in this calculation are bounding, given that the total amount of noble gases released from the breached fuel assemblies and all the iodines that reach the pool surface are transported to the unit vent, equipment hatch, and roll-up doors. No credit is taken for filtration through the equipment hatch or the roll-up doors, deposition, plateout, holdup nor decay.

The events analyzed involve confinement failure of all fuel pins in either a single fuel assembly or an array of fuel assemblies. The single fuel assembly event may occur in the SFP, the area outside of the SFP but inside the SFP Building, or in containment. The multiple fuel assembly events are postulated to occur only in the SFP. All the gases released from the fuel-cladding gap are assumed released within the first two hours of the accident.

A bounding source term was used for the FHA. Varying cycle lengths, initial enrichments and burnups were used, in conjunction with a peaking factor of 1.65 and power uncertainty of 2% for single fuel assembly accidents. For a fuel handling event involving multiple fuel assemblies, the source term is based on core averaged fuel with a peaking factor of 1.2. Minimum decay times were assumed to maximize the bounding source term.

For the purposes of calculating the Control Room operator dose, nuclides from FHA that occur in containment are released through the most limiting pathway. Nuclides released in the SFP Building are released to the building environment and then transported to a Control Room either through a unit vent or a roll-up door.

Control Room doses are calculated using the methodology consistent with Regulatory Guide 1.183 using dose conversion factors taken from Federal Guidance Report No. 11 and 12. Dose calculations to the Control Room operators as a result of a FHA are calculated using the LOCADOSE computer code and compared to a 5-rem TEDE limit. The Control Room operator dose estimates are calculated using site-specific atmospheric dispersion factors for the proposed dual Control Room air intake locations.

In accordance with Regulatory Guide 1.183, Appendix B.1.3, the chemical form of radioiodine released from the fuel to the water is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. The CsI released from the fuel is assumed to completely dissociate in the water, and the iodine

instantaneously re-evolves as elemental iodine due to the low pH of the water.

Major assumptions in the FHA are:

- Fuel assemblies that are breached in a FHA are assumed damaged to the extent that the entire gap inventory of the damaged fuel rods is released to the surrounding water. This is a conservative assumption since the quantity of gap inventory that escapes a breached fuel rod depends on several factors such as the extent of the breach, the axial location of the breach, and the orientation of the fuel rod.

- There is no credit taken for any pool retention of noble gases or organic iodine as the gas bubbles up through the water (DF = 1.0).

- No credit is taken for containment closure.

- No credit is taken for the SFP Ventilation System following the initiation of the fuel handling event.

- For single assembly fuel handling events, the fuel-cladding gap source term assumes a peak rod average fuel burnup of 62,000 MWD/MTU, with an axial power peaking factor of 1.65.

- For multiple assembly fuel handling events, the total fuel gap source term assumes a power peaking factor of 1.2 and a core average inventory.

- Inorganic iodine is treated as elemental iodine in this analysis.

- The release of the radionuclides from the SFP or containment is modeled to occur over a two-hour time frame and is confirmed with the LOCADOSE code. A ratio of 10:1, node volume to exhaust flow rate from the node will produce a two-

hour release that exhausts greater than 99.99% of the activity from the node.

- Control Room volumes are calculated to be 8.64E+4 cubic feet for the Unit 1 & 2 Control Room and 4.32E+4 cubic feet for the Unit 3 Control Room.

- No credit is taken in the dose calculation for any potential filtration of effluents through the roll-up door following the release of the effluents from the SFP. In addition, no credit is taken for potential deposition/plateout that should occur as effluents are released from the SFP.

- Based on results from Oconee specific Tracer Gas testing performed in 1998, the following values were used for Control Room unfiltered inleakage:

  Units 1 & 2

  Unfiltered inleakage prior to fan start = 1150 cfm
  Unfiltered inleakage after fan start = 150 cfm

  Unit 3

  Unfiltered inleakage prior to fan start = 600 cfm
  Unfiltered inleakage after fan start = 100 cfm

- The time at which the emergency Control Room ventilation booster fans establish positive pressure in the Control Room is 30 minutes following initiation of the fuel handling accident. This timing is applicable to both Unit 1&2 and Unit 3 Control Rooms. Prior to initiation of these fans, all flow into the Control Rooms is treated as unfiltered.

- The breathing rate used for Control Room Operators is 3.5E-4 m^3/sec for the duration of a fuel handling accident.

- Control Room Occupancies are :
  0 - 24 hours 100%

1 - 4 days	60%
4 - 30 days	40%

- The 0-2 hour Exclusion Area Boundary atmospheric dispersion factor is 2.2E-4 sec/m^3

- The Low Population Zone atmospheric dispersion factors are :

0 - 8 hours	2.35E-5 sec/m^3
8 - 24 hours	4.70E-6 sec/m^3
1 - 4 days	1.50E-6 sec/m^3
4 - 30 days	3.30E-7 sec/m^3

- Breathing rates used for calculation of offsite doses are:

0 - 8 hours	3.5E-4 m^3/sec
8 - 24 hours	1.8E-4 m^3/sec
1 - 30 days	2.3E-4 m^3/sec

- The emergency Control Room Ventilation booster fan intake flow rate is 1350 ± 135 cfm in each train (2700 ± 270 cfm in both trains). Since the lower flow rate yields a lower iodine protection factor, the flow is assumed to be 1215 cfm (one train at the minimum flow rate) for both Control Rooms. In this analysis, only one fan is assumed to start when activated by Control Room personnel.

- The assumed filter efficiencies for the Control Room intake iodine filters (active when booster fans are in operation) are 99% for particulate, 95% for organic, and 99% for elemental iodine.

- An alternative source term is calculated based on guidance provided in Regulatory Guide 1.183. In accordance with Regulatory Guide 1.183, Appendix B.2, decontamination factors of 500 and 1 are applied for the elemental and organic iodine, respectively, with a minimum depth of water above the damaged fuel of 23 feet or greater. Oconee TS 3.9.6 requires the fuel transfer canal water level to be maintained greater than or equal to 21.34 feet above the top of the reactor vessel flange.

Therefore, the elemental iodine decontamination factor is reduced accordingly to 430.1.

- Control Room χ/Q values have been calculated as shown in the table below. The bounding (maximum) χ/Q values for all units will be used for each release type. Used in conjunction, the proposed Northeast and Southeast Control Room intake locations will present a system of dual intakes without manual selection. A factor of 2 reduction in the most limiting χ/Q value is applied to account for dilution effects associated with a dual inlet configuration. Therefore, the χ/Q values are reduced by a factor of 2 for use in LOCADOSE modeling.

Parameter	**Value from χ/Q Calculation**	**Factor of 2 Reduction**
Unit Vent Atmospheric Dispersion Factor-χ/Q (0-2 hour)	8.70E-04 sec/m^3	4.35E-04 sec/m^3
Equipment Hatch Atmospheric Dispersion Factor-χ/Q (0-2 hour)	6.35E-04 sec/m^3	3.18E-04 sec/m^3
SFP Roll-up Door Atmospheric Dispersion Factor-χ/Q (0-2 hour)	2.88E-04 sec/m^3	1.44E-04 sec/m^3

In conclusion, the TEDEs to Control Room Operators are summarized in the table below for each case described using the AST. The alternative source term TEDE is calculated in accordance with guidance provided in Regulatory Guide 1.183.

According to the table below, the maximum TEDE calculated for either Control Room following a fuel handling event is 2.0-rem. The event is the transport cask drop in the spent fuel pool with

transport from the Unit 2 vent to the Unit 1 & 2 Control Room. This value satisfies the 5-rem occupational TEDE limit.

Calculated Doses to Control Room Operators due to Fuel Handling Events				
Case	**Source**	**Unit and Release Point**	**Control Room Unit Destination**	TEDE (rem)
1	Fuel Assembly Accident in SFP	Unit 2 Unit Vent	Unit 1&2	1.3
2	Fuel Assembly Accident in SFP	Unit 3 Roll-Up Door	Unit 1&2	0.5
3	Fuel Assembly Accident in SFP	Unit 2 Unit Vent	Unit 3	0.9
4	Fuel Assembly Accident in SFP	Unit 3 Roll-Up Door	Unit 3	0.3
5	Fuel Assembly Accident in Containment	Unit 2 Unit Vent	Unit 1&2	0.8
6	Fuel Assembly Accident in Containment	Unit 3 Unit Vent	Unit 3	0.5
7	Transport Cask Drop in SFP	Unit 2 Unit Vent	Unit 1&2	2.0
8	Transport Cask Drop in SFP	Unit 2 Unit Vent	Unit 3	1.4
9	ISFSI Cask Drop in SFP	Unit 2 Unit Vent	Unit 1&2	0.9
10	ISFSI Cask Drop in SFP	Unit 2 Unit Vent	Unit 3	0.6
11	ISFSI Cask Drop in SFP	Unit 3 Roll-Up Door	Unit 1&2	0.3
12	ISFSI Cask Drop in SFP	Unit 3 Roll-Up Door	Unit 3	0.2

The TEDEs for the offsite doses are summarized in the table below for each case described using the AST. According to the table below, the maximum TEDE occurring at the EAB or LPZ following a fuel handling event is 1.3-rem. Again, the limiting event is the transport cask drop in the SFP. This value satisfies the 6.3-rem offsite TEDE limit.

Calculated Doses for EAB and LPZ due to Fuel Handling Events				
Case	**Source**	**Unit and Release Point**	EAB TEDE (rem)	LPZ TEDE (rem)
1	Fuel Assembly Accident in SFP	Unit 2 Unit Vent	0.9	0.1
2	Fuel Assembly Accident in SFP	Unit 3 Roll-Up Door	0.9	0.1
5	Fuel Assembly Accident in Containment	Unit 2 Unit Vent	0.6	0.1
7	Transport Cask Drop in SFP	Unit 2 Unit Vent	1.3	0.2
9	ISFSI Cask Drop in SFP	Unit 2 Unit Vent	0.6	0.1
11	ISFSI Cask Drop in SFP	Unit 3 Roll-Up Door	0.6	0.1

In the event of a FHA, actual Control Room and offsite doses will be less than the results displayed because at least one door in each airlock, the equipment hatch door, as well as other

open penetrations, will be closed following an evacuation of containment. Since the FHA analyses were performed with containment open, they show that it is not necessary to have containment closure in order to show acceptable offsite or Control Room operator doses following a FHA.

The added provision for isolation methods for other penetrations is considered an administrative change consistent with B&W Owners Group Standard Technical Specifications, NUREG-1430, April 1995. This change enhances the existing Oconee TS by explicitly recognizing approved isolation methods.

Atmospheric Dispersion Factors (χ/Q)

Atmospheric dispersion factors (χ/Q) provide normalized concentrations of Oconee site effluents at specified locations onsite or at distances from the site. As a result, these factors are primary inputs to radiological dose assessments for accident analyses. χ/Q values were analyzed for dual Control Room intake locations proposed as part of the proposed modification package at Oconee Nuclear Station. These intake locations will be used by both the Unit 1 & 2 Control Room and the Unit 3 Control Room. The new intakes will replace the separate single intakes currently used by each of these Control Rooms.

Control Room χ/Q values at the Control Room air intake locations are developed for radionuclide releases from the following release points:

- Unit vents
- Reactor Building penetrations outside of the penetration rooms
- Atmospheric dump valves (ADVs)
- Main steam safety valves (MSSVs)
- Main steam line breaks (MSLBs)
- Fuel Handling Building roll-up doors
- BWST Vent

The ARCON96 code is used to calculate Control Room χ/Q values.

The ARCON96 methodology has been supplemented by draft NRC guidance on the use of ARCON96 provided to the NEI Control Room Habitability Task Force in June, 2000. The conditions and the prescribed acceptable input values from this document are applied to χ/Q determinations in this calculation.

The new Control Room air intake locations will be on the northeast and southeast corners of the Turbine Building. These locations have been selected to allow the use of dual intakes on the roof of the Turbine Building. The same intake locations will be used for both the Unit 1 & 2 Control Room and the Unit 3 Control Room. The elevation of both intakes is 25.9 meters. The Control Room intake is based on the preliminary design two feet above the roof elevation of 879.5 feet. These locations will be referred to by the designations "NE" and "SE".

Key assumptions used in calculating χ/Q values are:

- Radionuclide effluent concentrations at the Control Room air intake location resulting from continuous releases from an Oconee unit vent, Reactor Building penetration, or secondary-side release path can be conservatively estimated by the Gaussian plume models for continuous releases incorporated in the ARCON96 computer code. Therefore, the following assumptions are made:

    a) The release is continuous, constant, and of sufficient duration to establish a representative mean concentration, and
    b) The material being released is reflected by the ground.

- The treatment of calm wind transport and diffusion is based on the assumptions that:

    a) Air motions do not cease even when the mean wind velocity approaches zero, and
    b) The receptor is assumed to be directly downwind of the release point, regardless of the wind direction.

Site meteorological data digitally recorded for the 5 year period from 1/1991 through 12/1995 (e.g. wind speed, wind direction, atmospheric stability class) is judged to provide a representative characterization of site meteorological conditions for the purposes of estimating 95% confidence level atmospheric dispersion factors.

- Site-specific values for horizontal and vertical diffusion coefficients are bounded by values obtained from the Pasquill-Gifford-Turner database, as implemented in the ARCON96 code for all atmospheric stability classes and distances from the release point.

- Initial horizontal and vertical diffusion coefficients are used to define virtual area sources, and are therefore set to the default value of 0.0 meters, for all point releases. For the equipment hatch releases and Fuel Handling Building roll-up door releases, which are treated as area sources, these diffusion coefficients are defined as specified in draft NRC guidance using the dimensions of the release point.

- The surface roughness length is 0.20 meters. In the absence of a site-specific monitoring study, a value of 0.20 meters has been adopted based on the draft NRC guidance for ARCON96.

- The sector averaging factor, α, is 4.3. This is based on draft NRC guidance for ARCON96.

- Effluent releases are characterized as being cold and non-buoyant (i.e., near ambient temperature with little or no steam). The Gaussian model is deemed appropriate for characterizing releases. No credit is taken for buoyancy-induced plume rise. This is based on draft NRC guidance for ARCON96.

- Releases from all locations are modeled as ground-level releases. The release elevations of each of these paths are less than 2.5 times the height of the Reactor Buildings, so they do not meet staff guidance for classification as elevated

releases. Stack flow rate is not used with ground-level releases.

- Based on draft NRC guidance, the building vertical cross-sectional area perpendicular to the wind direction is entered to account for wake effects for ground level releases. The suggested default value of 2,000 square meters is used for the building area.

- Leakage from Reactor Building penetrations located inside the penetration rooms are assumed to be released through the unit vent.

- Leakage from the BWST is conservatively assumed to occur at the BWST vent valve located approximately one foot off the top of the BWST at the BWST centerline. This is based on a worst case release point from the BWST. All other release points from the BWST would be bounded by this release. Therefore there is no implied credit taken for the relief function of the valve.

Meteorological Input Data Requirements

The meteorological data required for ARCON96 χ/Q calculations include wind speed and wind direction measurements from both the lower and upper site meteorological tower locations, plus a measure of the vertical temperature gradient, from which atmospheric stability is inferred. These data represent hourly averages of observed plant conditions, as defined in Regulatory Guide 1.23.

Atmospheric stability is classified by vertical temperature gradient by the method described in RG 1.23. The atmospheric stability classes are defined according to the Pasquill-Turner stability class methodology. This classification method is used to characterize all atmospheric stability conditions for the Oconee site.

Atmospheric stability is determined by the vertical ambient temperature difference between the unit vent release height (60 m)

and the 10-meter level, as defined by the Pasquill stability classification system:

Description	Lapse Rate (°C/50 m)	Pasquill-Gifford Class	ARCON96 Class
Extremely unstable	$\Delta T/\Delta z \leq -0.95$	A	1
Moderately unstable	$-0.95 < \Delta T/\Delta z \leq -0.85$	B	2
Slightly unstable	$-0.85 < \Delta T/\Delta z \leq -0.75$	C	3
Neutral	$-0.75 < \Delta T/\Delta z \leq -0.25$	D	4
Slightly stable	$-0.25 < \Delta T/\Delta z \leq 0.75$	E	5
Moderately stable	$0.75 < \Delta T/\Delta z \leq 2.0$	F	6
Extremely stable	$2.0 < \Delta T/\Delta z$	G	7

Since the Oconee instrumentation program corresponds to the regulatory position in RG 1.23, calm winds are to be assigned the wind speed equal to the vane or anemometer starting speed, whichever is higher. For Oconee, this wind speed is 0.45 meter/second (m/s). The input file includes wind speed data for both the 10 meter and 60 meter heights. The default minimum wind speed of 0.5 m/s will be used in the ARCON96 input per draft NRC guidance.

Atmospheric dispersion factors for all possible source-receptor pairs are analyzed. Releases from each of the release locations for Units 1, 2 and 3 to each of the NE and SE control air intake locations are evaluated to see which is the bounding χ/Q for the plant. The bounding (maximum) χ/Q values for all units will be used for each release type. Used in conjunction, the proposed NE and SE CR intake locations will present a system of dual intakes without manual selection. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", Section 6.4, allows a factor of 2 reduction in the most limiting χ/Q value to account for dilution effects associated with a dual inlet configuration. Therefore, the χ/Q values calculated are to be reduced by a factor of 2 for use in LOCADOSE modeling. The bounding χ/Q values calculated are shown in the following table.

Maximum Bounding χ/Q Values (sec/m^3)

[NOTE: THESE VALUES HAVE NOT BEEN REDUCED BY A FACTOR OF 2.]

	Vent Releases	Main Steam Penetration Releases	Equipment Hatch Releases	ADV Releases
0 to 2 hr	8.70E-04	5.44E-04	6.35E-04	1.65E-03
0 to 8 hr	6.69E-04	4.44E-04	5.21E-04	1.27E-03
8 to 24 hr	2.54E-04	1.74E-04	2.08E-04	4.93E-04
1 to 4 days	1.99E-04	1.33E-04	1.56E-04	3.75E-04
4 to 30 days	1.61E-04	1.02E-04	1.22E-04	3.08E-04

	MSSV Releases	MSLB Releases	FHB Roll-up Door Releases	BWST Releases
0 to 2 hr	1.47E-03	1.12E-03	2.88E-04	3.87E-04
0 to 8 hr	1.13E-03	8.87E-04	2.44E-04	3.14E-04
8 to 24 hr	4.37E-04	3.47E-04	9.70E-05	1.21E-04
1 to 4 days	3.29E-04	2.59E-04	7.45E-05	9.43E-05
4 to 30 days	2.73E-04	2.09E-04	5.66E-05	7.39E-05

Conclusion

The AST defined in Regulatory Guide 1.183 and associated analysis guidance provided has been incorporated into the re-analysis of radiological effects of the MHA and FHA analyses. This represents a full implementation of the AST that will become the new licensing basis source term for assessment of design basis events.

The analysis results from the re-analyzed events meet all of the acceptance criteria as specified in 10 CFR 50.67 and Regulatory Guide 1.183.

Additionally, this LAR proposes changes to TS based on the revised MHA and FHA analyses. These proposed changes involve revising the TS requirements for containment integrity during fuel movement and refueling operations, lowering the allowable Reactor Building volume leakage rate per day limit, removing the PRVS and SFPVS requirements from TS, removing the requirement to measure Reactor Building leakage in excess of 50% of L_a to the Penetration Room and revising the VFTP radioactive methyl iodide removal acceptance criterion for the CRVS Booster Fan trains.

The modifications, described previously, that will be implemented are necessary to support the revised analysis methodology, the margin embedded in the analysis input values and assumptions, and the overall resolution of issues related to Control Room Habitability. Implementation of the AST into the Oconee design basis will be integrated with the completion of the associated plant modifications.

Based on the considerations detailed in this Attachment, the changes proposed in this LAR are concluded to be acceptable for implementation at Oconee.

ATTACHMENT 4

Determination of
No Significant Hazards Considerations
Basis for Determination of No Significant Hazards Considerations

Standards for determining whether a license amendment involves no significant hazards considerations are contained in 10CFR50.92(c). The TS changes and modifications as proposed in this LAR have been evaluated in accordance with 10 CFR 50.92 and determined not to involve any significant hazards considerations.

The proposed LAR includes (1) implementing the AST for accident analysis as described in Regulatory Guide 1.183;
(2) removing the PRVS and the SFPVS from TS because they are no longer being credited for Control Room and off-site doses;
(3) allowing containment airlocks, the equipment hatch and penetrations providing direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative controls and eliminating the requirement to maintain an operable automatic isolation capability for the Reactor Building Purge system during refueling; (4) lowering the Reactor Building leakage rate from 0.25 w%/day to 0.20 w%/day;
(5) removing the TS requirement to measure Reactor Building leakage in excess of 50% of L_a to the penetration room; and (6) revising the VFTP radioactive methyl iodide removal acceptance criterion for the CRVS Booster Fan trains.

Plant modifications are also being proposed in concert with the proposed TS changes and they include relocating the existing Control Room outside air intake from the roof of the Auxiliary Building to the roof of the Turbine Building and installing dual intakes for each Control Room; re-routing HPI/LPI relief valve discharge back into the Reactor Building and replacing the existing Caustic Addition system.

As a result of this evaluation, Duke has concluded:

1) The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The AST and those plant systems affected by implementing the proposed changes to the TS are not assumed to initiate design basis accidents. The AST does not affect the design or

operations of the facility. Rather, the AST is used to evaluate the consequences of a postulated accident. The implementation of the AST has been evaluated in the revisions to the analysis of the design basis accidents for Oconee Nuclear Station. Based on the results of these analyses, it has been demonstrated that, with the requested changes, the dose consequences of these events meet the acceptance criteria of 10 CFR 50.67 and Regulatory Guide 1.183. Therefore, the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2) The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The AST and those plant systems affected by implementing the proposed changes to the TS are not assumed to initiate design basis accidents. The systems affected by the changes are used to mitigate the consequences of an accident that has already occurred. The proposed TS changes and modifications do not significantly affect the mitigative function of these systems. Consequently, these systems do not alter the nature of events postulated in the Safety Analysis Report nor do they introduce any unique precursor mechanisms. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) The proposed amendment will not involve a significant reduction in the margin of safety.

The implementation of the AST, proposed changes to the TS and the implementation of the proposed modifications have been evaluated in the revisions to the analysis of the consequences of the design basis accidents for the Oconee Nuclear Station. Based on the results of these analyses, it has been demonstrated that with the requested changes the dose consequences of these events meet the acceptance criteria of 10 CFR 50.67 and Regulatory Guide 1.183. Thus,

the proposed amendment will not involve a significant reduction in the margin of safety.

ATTACHMENT 5

Environmental Assessment/Impact Statement

10CFR51.22(c)(9) identifies certain licensing and regulatory actions which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released off site; and (3) result in a significant increase in individual or cumulative occupational radiation exposure.

Duke Energy Corporation has reviewed this request and determined that implementation of the AST, proposed TS amendments and modifications meet the eligibility criteria for categorical exclusion set forth in 10CFR51.22(c)(9). Pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows:

1) As demonstrated in Attachment 4 to this letter, the changes proposed by this LAR does not involve a significant hazards consideration.

2) The changes proposed by this LAR do not introduce any new effluents or significantly increase the quantities of existing effluents. The plant systems affected by the proposed changes do not interface with any plant system that is involved in the generation or processing of radioactive fluids. Therefore, implementation of this LAR will not result in a significant change in the types or increase in the amounts of any effluents that may be released offsite.

3) As demonstrated in Attachment 3 to this letter, the changes proposed by this LAR do not result in a significant increase in Control Room operator doses during design basis radiological accidents. In addition, the proposed changes do not result in any physical plant changes or surveillances which would require additional personnel entry into radiation controlled areas.

Therefore, the LAR, as proposed, will not result in a significant increase in either individual or cumulative occupational radiation exposure.