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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Attention: Document Control Desk

Subject: GGNS Pilot Full-Scope Application of NUREG-1465 Alternative Source Term Insights, LDC 1999-082
Docket No. 50-416
License No. NPF-29

Reference: 1) GNRO-99/00077, GGNS Letter to NRC, Pilot Limited-Scope Application of NUREG-1465 Alternative Source Term Insights, dated November 3,1998

GNRO-2000/20005

Gentlemen:

Grand Gulf is a pilot plant in the collaborative efforts of the Nuclear Regulatory Commission (NRC), Nuclear Energy Institute (NEI), and the Electric Power Research Institute (EPRI) for the implementation of the NRC research efforts documented in the NUREG. Attached is a request to revise the GGNS licensing basis to implement the alternative source term. This represents a full-scope application.

Because this change would implement aspects associated with the alternative source term, the NRC has deemed it to involve an unreviewed safety question. This letter requests a license amendment in accordance with 10CFR50.90. A statement of No Significant Hazards Considerations is included in Attachment 1. A detailed discussion of the requested change is also included in that attachment. Marked up copies of the affected Technical Specification pages showing the requested changes are provided for your review in Attachment 2. Copies of changes to the Technical Specification Bases pages supporting the above changes are included for your information in Attachment 3. Design basis analyses and reports utilizing the alternative source term as an input are included for your review in Attachments 4 through 9.

We request that the review fees associated with the NRC evaluation of this license amendment submittal be waived. This request is made pursuant to 10CFR170.11(b)(1), which governs exemptions from fees granted upon the initiative of the NRC. This request is based on: 1) the participation of Grand Gulf Nuclear Station as a pilot plant and as a member of the NEI Task Force supporting the development of the proposed rule and associated regulatory guide, and 2) the technical information and support provided by GGNS during the NRC re-baselining analysis effort associated with the AST development work. We appreciate the opportunity to participate in the NUREG-1465 evaluation efforts both as a pilot application plant and as a rebaselining analysis subject. We look forward to continued cooperation on this project as the NRC Rulemaking Plan is implemented. We support the revised source term initiative and believe it is an important step toward risk-informed regulatory policy. If you have any questions regarding this submittal, please contact Jerry Roberts at 601-437-6710.

Yours truly,

William A Eaton

/FGB

Attachment 1:	Discussion of Proposed Changes
Attachment 2:	Markups of Affected Technical Specification Pages
Attachment 3:	Markups of Affected Technical Specification Bases Pages
Attachment 4:	CRDA Analysis
Attachment 5:	LOCA Dose Analysis
Attachment 6:	Design Basis FHA Radiological Analysis
Attachment 7:	Suppression Pool pH and Iodine Re-evolution Methodology Engineering Report
Attachment 8:	Suppression Pool pH Analysis
Attachment 9:	Doses from Iodine Re-evolution
Attachment 10:	Affirmation

cc: (see next page)

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Attachment 1

Discussion of Proposed Changes

PURPOSE

Grand Gulf Nuclear Station requests the NRC review and approval of this proposed request to revise our licensing basis to utilize the alternative accident source term described in NUREG-1465 (Reference 4). The current basis utilizes a source term determined in accordance with TID-14844 (Reference 5). This request has been developed considering the recently approved rulemaking and draft regulatory guidance (DG-1081) developed by the NRC (References 7 and 15). As the implementation of the alternative source term involves an unreviewed safety question, GGNS requests a license amendment in accordance with 10CFR50.90 approving the use of the new source term.

This request also proposes revisions to several Technical Specifications that include special applicability wording to invoke safety controls during shutdown operations. Similar changes to several specifications were recently proposed (see Reference 16) based on the results of a revised Fuel Handling Accident analysis performed using the original source term. That submittal introduced the term "recently irradiated fuel assemblies" into the specifications and initially established its definition as fuel that had been used in the reactor and was within an eight-day period following shutdown. Those changes were approved in amendment 139 to the GGNS Operating License. The FHA analyses have been revised again in preparation for this submittal, this time utilizing the alternative source term. This submittal proposes to expand the use of the term to several more specifications and also to redefine the term to involve only a seven-day decay period after shutdown. The relaxation of shutdown safety controls had been the subject of three previous GGNS submittals (References 18, 19, and 20). The changes proposed herein update the earlier request to incorporate alternative source term concepts and are consistent with the changes proposed in Reference 16 and with the industry proposed changes of TSTF-51.

SCOPE

This submittal represents a full-scope implementation of the new source term. Design basis accident analyses 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. The impact of the new source term on plant equipment has been evaluated based on comparison of the new radiation environment to the current radiation environment specified for the qualification of the equipment.

Grand Gulf Nuclear Station (GGNS) is a pilot plant in the effort to evaluate the potential application of the insights regarding the alternative source term. In fact, GGNS is also the BWR evaluated by the NRC in the revised source term rebaselining effort using the NUREG-1465 insights. The results of that evaluation are documented in Reference 3. As noted above, this submittal represents a full-scope application of the new source term methodology; it complements a limited-scope application submitted earlier (Reference 2).

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 alternative source term should be reviewed and approved by the NRC in the form of a license amendment. This submittal is presented for NRC review and approval consistent with the intent of the objectives of the pilot program and agreements made between the NRC and the pilot program licensees. The requested change has not been subject to a 10CFR50.59 review; it is being conservatively submitted as an unreviewed safety question in accordance with guidance noted in the draft rulemaking. In addition, there are changes to the Technical Specifications associated with this request.

Further, this request is based in part on a BWROG report [Reference 6] that has been recently approved by the NRC [see Reference 28]. GGNS had previously submitted [Reference 1] for approval the report justifying a time to cladding breach of 121 seconds for the BWR fleet. In addition, GGNS has also made a submittal of a limited scope application of the alternative source term insights; that submittal was based on the timing of the radioactive release [Reference 2]. This request is consistent the approach and methodology used in those submittals to apply the concepts associated with the alternative source term.

SUMMARY OF CHANGES

The implementation of the new source term involves changes to the following Technical Specifications, Technical Specification Bases, and Operating License condition:

Technical Specification	Affected pages
1.1 Definitions	1.0-3
3.3.6.1 Primary Containment and Drywell Isolation Instrumentation	Bases change only
3.3.6.2 Secondary Containment Isolation Instrumentation	Bases change only
3.3.7.1 CRFA Instrumentation	3.3-73 through 3.3-76
3.6.1.3 Primary Containment Isolation Valves (PCIVs)	3.6-17 (and Bases change)
3.6.4.1 Secondary Containment	3.6-44 (and Bases change)
3.6.4.2 Secondary Containment Isolation Valves (SCIVs)	Bases change only
3.6.4.3 Standby Gas Treatment (SGT) System	Bases change only
3.7.3 Control Room Fresh Air System (CRFA)	3.7-6 through 3.7-8
3.7.4 Control Room AC System	3.7-9 through 3.7-11
3.8.2 AC Sources – Shutdown	3.8-18 through 3.8-20
3.8.5 DC Sources – Shutdown	3.8-31 through 3.8-33
3.8.8 Distribution Systems – Shutdown	3.8-40

Operating License Condition	**Affected pages**
2.C(38)	15

Each of the changes is discussed in more detail below. Markups of the Technical Specification pages illustrating the specific changes are provided in Attachment 2.

Technical Specification 1.1 - Definitions

Changes are proposed to two definitions in this section – DOSE EQUIVALENT I-131 and L_a. DOSE EQUIVALENT I-131 is defined as that concentration of I-131 (in units of microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. These words will not change. The Technical Specification definition goes on, however, to state: "The thyroid dose conversion factors (DCFs) used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites."" This happens to be the only reference to TID-14844 which appears in the Technical Specifications. This dose equivalent terminology is in turn used in Specification 3.4.8 and in Table 6.11.4-1, which appears in both the Technical Requirements Manual and the Offsite Dose Calculation Manual. The design basis dose analyses performed in support of this submittal utilize DCFs taken from Federal Guidance Report (FGR) 11 (Reference 21). It is proposed that the TID reference in the definition simply be reworded to refer to this report.

The maximum allowable primary containment leakage rate, L_a, is currently defined as 0.437% of the primary containment air weight per day at the calculated peak containment pressure. It is proposed that the value for the allowable leakage rate be increased to 0.682%. Much of this increase is associated with the requested increase in the MSIV leak rate, which is summarized in the discussion below of the proposed change to SR 3.6.1.3.8. It also includes a 10% increase in the allowable containment leak rate from other sources (i.e., from 0.350% to 0.385%.) This increased leakage has been evaluated in the LOCA dose analysis (Reference 12); the resulting offsite and control room doses are well within the acceptance criteria of the recently issued 10CFR50.67 and GDC 19.

Technical Specification Bases 3.3.6.1, 3.3.6.2, 3.6.1.3, 3.6.4.1, 3.6.4.2, and 3.6.4.3

(Note – The changes summarized here deal only with a revision to previously proposed changes to the Bases for these specifications. Implementation of the alternative source term also involves additional changes to Technical Specifications 3.6.1.3 and 3.6.4.1 that are discussed separately below.)

Revisions to these six specifications were requested in Reference 16 to relax the operational constraints during an outage. That submittal, which was approved in amendment 139 to the GGNS Operating License, proposed that the Applicability Statements of these selected specifications be revised to utilize the term "recently irradiated fuel assemblies." The use of this terminology recognizes that fission product decay after shutdown serves to reduce the dose consequences of fuel handling accidents. That submittal noted that the revision to the Fuel Handling Accident calculation utilizing the original GGNS licensing basis TID-14844 source term yielded an eight-day decay requirement before the operational constraints could be relaxed. The actual length of the required decay period was to be identified in the Bases for these specifications.

These specifications are not being further revised in this submittal. Rather, they have been included here to inform the NRC that the eight-day minimum decay requirement based on the original source term can be further reduced to a seven-day period with the use of the alternative source term being requested here. This demonstrates a benefit of the alternative source term. Again, while these specifications are unaffected by this change, GGNS intends to revise the Bases for these specifications to reflect the new interpretation of the term "recently irradiated fuel assemblies" as fuel which has been irradiated in the reactor within the previous seven days.

As an additional related aspect of the alternative source term, GGNS has determined that the term "recently irradiated fuel assemblies" may now be applied to several other specifications. These include the last five specifications listed in the table above. Those specific changes are discussed below.

Technical Specification 3.3.7.1 – CRFA Instrumentation

This specification requires the operability of the instrumentation associated with the initiation of the Control Room Fresh Air (CRFA) System. This system provides for the isolation of the Control Room and for the recirculation and filtration of the Control Room environment. Currently, the instrumentation addressed by this specification includes:

- Reactor Vessel Level – Low-Low,
- Drywell Pressure – High,
- Control Room Ventilation Radiation – High, and
- Manual Initiation.

With the implementation of the alternative source term, the only safety function to be required of the CRFA system is manual control room isolation. Analyses performed in support of this submittal (References 11 and 12) made no assumptions regarding automatic control room isolation. Instead, they credited manual action to isolate the control room. As discussed later, the analysis of the fuel handling accident (Reference 14) assumed no control room isolation. It is proposed that all automatic control room isolation features be deleted from the scope of the Technical Specifications. Implementation of this change will also involve:

1) The deletion of the three instruments providing the automatic isolation input signal from Table 3.3.7.1-1,
2) the deletion of the Surveillance Requirements 3.3.7.1.1 through 3.3.7.1.5, which are associated with establishing the operability for those instruments proposed for deletion,
3) the deletion of ACTIONs A, B1, C and D, which are also no longer applicable when the automatic isolation instruments are deleted from the specification. The wording of the proposed ACTION statement retains the intent of ACTIONs B.2 (which becomes A) and E (which becomes B),

4) the deletion of Table 3.3.7.1-1. All of the information from this table has now been incorporated into the revised specification, and
5) the revision of the LCO to clearly reflect that only the manual isolation function of the system is addressed by this specification.

In summary, this specification will require the operability of only the Manual Initiation instrumentation and will retain only current ACTIONs B.2 and E, and the current Surveillance Requirement 3.3.7.1.6, which establishes the operability of this device. The markup included in Attachment 2 reflects the extensive changes being proposed to this specification.

Surveillance Requirement 3.6.1.3.8 - MSIV Leakage Rate

(Note – Specification 3.6.1.3 was also discussed above with regard to the relaxation of operational constraints during shutdown and the related impact on the Bases of this specification. See above change summary for Technical Specifications 3.3.6.1, 3.3.6.2, 3.6.1.3, 3.6.4.1, 3.6.4.2, and 3.6.4.3. The change described below affects a current surveillance requirement and is based on the LOCA dose analysis [Reference 12] performed to support the implementation of the alternative source term.)

This surveillance is associated with the specification for the operability of the primary containment isolation valves. It currently establishes the leak rate acceptance criterion for the MSIVs to be less than or equal to 100 scfh through all four main steam lines when tested at the calculated peak containment pressure. It is proposed that the allowable leak rate be increased to less than or equal to 100 scfh per main steam line with a total leak rate through all four main steam lines of less than or equal to 250 scfh. This increased leak rate value has been considered in the revised LOCA dose analysis (Reference 12) and the dose consequences were determined to be acceptable.

GGNS is not requesting the deletion of the Main Steam Isolation Valve Leakage Control System. Further, GGNS is not applying the steam line deposition methodology reported in NEDC-31858P [Reference 27]. As such, statements made in NEDC-31858P regarding a reduced "as-left" leak rate for any valve found to exceed the 100 scfh criterion, are not considered to be applicable to this GGNS application.

Surveillance Requirement 3.6.4.1.3 - Secondary Containment Drawdown

This surveillance is associated with the specification for the operability of the secondary containment. It currently establishes an allowable drawdown time of 120 seconds in which the Standby Gas Treatment System must be capable of drawing a vacuum in the Secondary Containment. It is proposed that the allowable drawdown time be increased to 180 seconds. The LOCA dose analysis performed in support of this submittal (Reference 12) utilized the revised assumption of 180 seconds of release prior to the establishment of the required drawdown in the secondary containment.

Technical Specification 3.7.3 – Control Room Fresh Air System

The Control Room Fresh Air (CRFA) System provides for the isolation of the ventilation flowpaths and for the recirculation and filtration of the Control Room atmosphere. The dose analyses of the LOCA and CRDA [References 11 and 12] were performed considering the alternative source term took credit only for the isolation function of the CRFA system. No credit was taken for either iodine or particulate removal by the CRFA filters. On this basis, it is proposed that this specification be revised to address only the isolation function. The proposed revision is worded to be similar to the specifications for the primary and secondary containment isolation valves. In addition, the analysis of the Fuel Handling Accident [Reference 14] using the alternative source term takes no credit for the CRFA isolation or filtration. The Fuel Handling Accident represents the only accident during Modes 4 and 5 that can result in a significant release of radioactivity. On this basis, the Applicability of the revised specification also deletes the fuel movement and CORE ALTERATIONS periods altogether. Since these calculations no longer take credit for the recirculation and filtration functions of the CRFA, these components no longer meet any of the criteria of 10CFR50.36(c)(2)(ii) for inclusion in the Technical Specifications.

Technical Specifications 3.7.4, 3.8.2, 3.8.5, and 3.8.8

The Applicability Statements for each of the above LCOs are proposed to be modified from "when handling irradiated fuel assemblies" to "when handling *recently* irradiated fuel assemblies". Also, revised wording of both the Conditions and Required Actions are proposed to be consistent with the change in the LCO Applicability Statement. The net result of this proposal is to establish a new term for that irradiated fuel that no longer contains sufficient fission products to require the operability of accident mitigation systems to meet the accident analysis assumptions. This new term is then used to define the conditions where fuel handling activities may represent situations in which significant radioactive releases can be postulated and to refine the appropriate operability requirements for the associated safety systems. The actual definition of the term recently irradiated fuel assemblies will be included in the Bases for each of these specifications and is described further in the Discussion section below.

The use of the term "recently irradiated fuel assemblies" provides a mechanism for applying a cutoff in fission product decay to the various specifications where the concept applies. The term is a plant-specific parameter that will be evaluated each fuel cycle. It will be defined in the Bases of the applicable specifications. For the current fuel cycle, the term will be defined as those assemblies that have been in a critical reactor core within the previous seven days. The 7-day period to be discussed in the Technical Specification Bases has been shown by analysis to provide sufficient decay such that, assuming the design basis fuel handling accident, radiological consequences are within the acceptance criteria of 10CFR50.67 and General Design Criteria 19 [Reference 7].

In addition, it is proposed that the Applicability Statement for 3.7.4 be modified to no longer require the LCO to be met during CORE ALTERATIONS. Revised wording of both the Conditions and Required Actions is proposed to be consistent with the change in the Applicability Statement. As described in the UFSAR [Ref. 24], the accidents postulated to occur during core alterations, in addition to fuel handling accident, are:

- inadvertent criticality due to a control rod removal error or continuous control rod withdrawal error during refueling, and
- the inadvertent loading and operation of a fuel assembly in an improper location.

These events are not postulated to result in fuel cladding integrity damage. Since the only accident postulated to occur during CORE ALTERATIONS that results in a significant radioactive release is the fuel handling accident, the proposed Technical Specification requirements deleting the CORE ALTERATIONS constraint is acceptable. The LCO Applicability Statements related to operations with a potential for draining the reactor vessel are unaffected by the proposed changes.

The changes proposed to these specifications are similar to those that had been proposed to specifications for other ESF systems in Reference 16 and approved in amendment 139 to the GGNS Operating License. That submittal was based on the use of the original TID-14844 source term. It proposed changes to Technical Specifications 3.3.6.1, 3.3.6.2, 3.6.1.3, 3.6.4.1, 3.6.4.2, and 3.6.4.3. The changes to these affected specifications are consistent with those changes proposed by the Technical Specification Task Force in TSTF-51.

License Condition 2.C(38) - Control Room Leak Rate

The GGNS Operating License includes a condition that establishes the allowable control room leak rate. It also includes an increased allowable leak rate which would be permissible should construction on Unit 2 restart. It is proposed that the allowable leak rate be increased from 590 cfm to 1200 cfm. Further, based on the implementation of the alternative source term, the dose consequence analysis of the Fuel Handling Accident [Reference 14] has demonstrated that there need be no infiltration restrictions during shutdown. Also, because the Construction Permit for Unit 2 has been revoked [Reference 29], there is no longer a need to retain the Unit 2 construction contingency; it is proposed that the second sentence be deleted. Therefore, it is proposed that the wording of this License Condition be revised to read as follows:

> EOI shall operate Grand Gulf Unit 1 during Modes 1 through 3 with an allowable control room leak rate not to exceed 1200 cfm.

DISCUSSION

Licensing Basis - Limiting Events

The design basis analyses of the three limiting events at GGNS have been revised in support of this submittal. The current licensing basis discussion of these accidents is included in the UFSAR in Section 15.6.5 for the Loss-of-Coolant Accident, 15.4.9 for the Control Rod Drop Accident, and 15.7.4 and 15.7.6 for the Fuel Handling Accidents in the Auxiliary Building and the Containment, respectively. The revised radiological evaluations of these events are developed in References 11, 12, and 14, which are included as attachments to this submittal.

In addition to the alternative source term inputs, several assumptions have been made in these analyses which differ from those described in the UFSAR. These revised assumptions form the basis for the changes requested to the Technical Specifications and the License Condition. Some of the key changes are:

1. MSIV leakage rates have been increased to consider a total leakage rate from all four main steam lines of up to 250 scfh.

2. Secondary containment drawdown time has been increased in the LOCA analysis from 2 to 3 minutes.

3. Control Room inleakage has been assumed to increase from 590 cfm to 1200 cfm for both of the events that can occur at power (i.e., the LOCA and the CRDA). An additional 10 cfm to account for opening of the doors, as recommended in Regulatory Guide 1.78, is also still considered. No credit has been assumed for the automatic isolation of the Control Room after these events; rather, manual isolation within 20 minutes after the accident has been assumed. Note that for the Fuel Handling Accident, no credit is taken for the isolation of the Control Room; outside air is continuously drawn through the normal ventilation system at a rate of 2000 cfm during this event.

4. The containment leak rate is assumed to be 0.385% rather than the current 0.35%.

5. SGTS bypass flow has been reduced from 50 cfm to 1 cfm based on design changes that have been incorporated into the plant design.

Licensing Basis - Other Events

As described in GGNS SAR Section 15.6.4, the main steamline break (MSLB) outside containment would release reactor coolant to the environment during the 5.5 seconds before the MSIVs are fully closed. Although NUREG-1465 does not affect the isotopic activity in the reactor coolant, the proposed change in the definition of DOSE EQUIVALENT I-131 would result in a different iodine inventory in the reactor coolant due to the application of the FGR-11 dose conversion factors rather than those in TID-14844. Consequently, a sensitivity evaluation is performed below to evaluate the impact of these revised dose conversion factors on the radiological consequences of this event.

As shown in the following table, the FGR-11 DCFs are considerably different from those reported in TID-14844 and would result in more flexibility in allowed iodine concentration in the reactor coolant. This change results in a 74% increase in the allowable iodine concentrations for the Technical Specification Action Level of 0.2 μCi/g Dose Equivalent I-131 as shown below. The impact of this change is demonstrated by considering the product of the iodine DCF and allowable activity concentration. As shown below, the higher allowable iodine concentration is offset by the lower FGR-11 DCF such that the total thyroid dose would be less with the coolant iodine concentration calculated with the FGR-11 DCFs. No changes are being proposed that would impact the amount of coolant released in this event.

Table 1
Dose Conversion Factor Comparison

	Thyroid Dose Conversion Factor (Rem/Ci)		Allowable Iodine Concentration (μCi/g) at 0.2 μCi/g Dose Equivalent I-131		Product (Rem/g)	
Isotope	**TID-14844**	**FGR11**	**TID DCFs**	**FGR11 DCFs**	**TID-14844**	**FGR11**
I-131	1.48E6	1.080E6	4.93E-02	8.58E-02	7.30E-02	9.27E-02
I-132	5.35E4	6.438E3	4.90E-01	8.51E-01	2.62E-02	5.48E-03
I-133	4.00E5	1.798E5	3.23E-01	5.62E-01	1.29E-01	1.01E-01
I-134	2.50E4	1.066E3	9.36E-01	1.63E+00	2.34E-02	1.74E-03
I-135	1.24E5	3.130E4	4.75E-01	8.26E-01	5.89E-02	2.59E-02
					3.11E-01	2.27E-01

This argument demonstrates that, as a result of the proposed changes, the offsite dose in the event of an MSLB would be expected to decrease. Control room doses are not currently reported for the MSLB event. However, a conservative estimate of the control room thyroid dose can be made by multiplying the EAB thyroid dose (currently reported in UFSAR Table 15.6-4) by the ratio of the control room χ/Q value (based on a release from the steam tunnel blowout panels) to the EAB χ/Q value. This calculation conservatively ignores the offsite dose reduction due to the application of the FGR11 DCFs as developed above for the offsite dose. As calculated below, the control room thyroid dose is not expected to exceed 12.6 Rem which is within the 30 rem requirement associated with the existing GDC 19.

Control Room X/Q based on release from blowout panels

$$5.82\,\text{Rem} * \frac{2.07\text{E}-3}{9.56\text{E}-4} = 12.6\,\text{Rem Thyroid}$$

Current EAB Thyroid Dose

EAB χ/Q

Other events involving reactor coolant only releases (i.e., no fuel failures), such as the feedwater line break outside containment (UFSAR Section 15.6.6) or the MSIV closure event (UFSAR Section 15.2.4) are bounded by the releases associated with the MSLB outside containment.

A number of other events in UFSAR Chapter 15 involve fuel failure and gap release. These events include:

- Pressure Controller Failure – Closed (SAR 15.2.1)
- Recirculation Pump Seizure in Single-Loop Operation (SAR 15.3.3)
- Misplaced Bundle Accident (SAR 15.4.7)

The fuel failures for these events were assumed to occur as a result of fuel rods experiencing departure from nucleate boiling. However, as demonstrated in Reference 1, fuel rods must become uncovered for a significant period of time (>1 minute) for cladding failure to occur. None of the above events would result in the core becoming uncovered. On this basis, and consistent with the discussion in Section 3.6 of Draft Guide 1081, the above non-LOCA events are not postulated to involve any gap release and have not been the subject of a detailed re-analysis as part of this submittal.

Accident Analyses and Results

In developing this submittal, GGNS revised the design basis accident analyses. The analyses for those accident scenarios determined to represent controlling cases for the dose results are included for NRC review in this submittal as Attachments 4 through 6. Specifically,

Control Rod Drop Accident (Reference 11)	Attachment 4
LOCA analysis (Reference 12)	Attachment 5
FHA – Radiological Analysis (Reference14)	Attachment 6

These calculations include the details of the design inputs and assumptions used in applying the alternative source term concepts to the GGNS. The dose results from these analyses are summarized below in Table 2. In addition to the accident analyses above, several other calculations were prepared to address topics of interest in establishing the alternative source term as the design basis for GGNS. These calculations are also attached for NRC review:

Suppression Pool pH and Iodine Re-Evolution Methodology (Reference 8)	Attachment 7
Suppression Pool pH Analysis (Reference 9)	Attachment 8
Doses from Iodine Re-evolution (Reference 10)	Attachment 9

Table 2
Dose Results Using the Alternative Source Term

	EAB/LPZ Dose (rem TEDE)			Control Room Dose (rem TEDE)	
Accident	Calculated	Regulatory Limit		Calculated	Regulatory Limit
LOCA	19.6 / 13.0	25		4.92	5
CRDA	0.27 / 0.13	6.25		0.16	5
FHA	2.50 / note 2	6.25		4.67	5

Notes for Table 2 –

1) Regulatory Limits are the accident dose acceptance criteria taken from Rulemaking for "Use of Alternative Source Terms at Operating Reactors" (Reference 7) and Draft Regulatory Guide DG-1081 (Reference 15). The offsite dose limit is from 10CFR50.67(b)(2)i and the control room dose limit is from 10CFR50, Appendix A, GDC 19.
2) LPZ dose not calculated for the Fuel Handling Accident. The EAB atmospheric dispersion factor is 4.9-times that of the LPZ.

Control Room Habitability

The proposed changes include several items associated with the control room envelope and its ventilation system. Each of the changes, including the removal from technical specifications of the automatic isolation instrumentation and the Control Room Fresh Air System fans and filters and the increase in the allowable in-leakage rate, has been modeled and evaluated in the design analyses prepared in support of this submittal. These conservative calculations demonstrate that the changes are acceptable and the control room envelope remains habitable following a design basis accident. The post-accident dose rates to the operators in the control room are within the regulatory acceptance criteria of General Design Criteria 19. While these changes represent a relaxation of the design features, EOI remains committed to ensuring plant personnel are adequately protected from any hazard that may affect their performance following a design basis event.

The above changes represent a proposal to revise the licensing basis for GGNS in the area of control room design. EOI is aware of NRC concerns that the design and operation of some control rooms may not be consistent with its design and licensing bases. EOI is actively participating in the industry initiative to develop guidance to aid plants in demonstrating that their design and licensing bases are understood and are satisfied. The tentative schedule for the industry effort is to issue final guidance for NRC review and endorsement by late November 2000. GGNS would suggest that the review of the proposed changes to its licensing basis may proceed to completion independent of the NRC review and acceptance of the industry effort.

Equipment Qualification

The NRC, in the Federal Register notice of the final Alternative Source Term Rule, provided a discussion of the topic of equipment qualification. Excerpts from the notice are paraphrased in the first two paragraphs below.

The re-baselining study prepared by the NRC staff (Reference 3) considered the impact of an AST on analyses of the postulated integrated radiation doses for plant components exposed to containment atmosphere radiation sources and those exposed to containment sump radiation sources. The study also concluded that the increased concentration of cesium in the containment sump could result in an increase in the postulated integrated radiation doses for certain plant components subject to equipment qualification. Further, the NRC has determined that it is necessary to consider the potential impact of the postulated cesium concentration in the containment sump water as it applies to all operating power reactors, not just to those licensees amending their design basis to use an AST.

Since the postulated increase in the post-accident integrated dose occurs well into the event scenario (i.e., well beyond 30 days), there is no adverse effect on equipment relied upon to perform safety functions immediately following an accident. Rather, this issue was found to affect equipment that is required to be operable longer than about 30 days to 4 months after an accident. As such, the NRC determined that continued plant operation does not pose an immediate threat to public health and safety. Also, should such long-term equipment fail there will not be an undue threat to public health and safety as protective actions for the public would have already been implemented by the time the postulated failure could occur. In addition, the time period between the onset of the event and the projected failure allows compensatory measures to be taken to prevent the equipment failure or to restore the degraded safety function. The NRC plans to evaluate this issue as a generic safety issue to determine whether further regulatory actions are justified. The final regulatory guide is expected to reflect the resolution of this generic safety issue.

Anticipating further NRC review of this issue and to ascertain the potential impact on the plant, GGNS has performed an evaluation of the impact of the implementation of AST on plant equipment. This evaluation has been performed qualitatively based on a comparison of the anticipated new radiation environment to the current radiation environment specified for the qualification testing of the equipment. The current equipment qualification data packages have not been revised. The acceptability of this approach of not revising the design basis Equipment Qualification analyses was suggested at the June 1999 NRC Workshop on the Alternative Radiological Source Term. In discussing a proposed draft of a new regulatory guidance document (DG-1081), the NRC noted it was not necessary to revise all design analyses; only those affected by the scope of any proposed changes should be revised. However, where sensitivity analyses or evaluations demonstrated that the current design to TID-14844 source term enveloped the new source term, re-analysis was not necessary. GGNS has reviewed the changes being proposed here and concluded that none of the changes, other than the use of the AST itself, is expected to have any adverse impact on radiation doses to equipment.

The major impact of the new source term data with respect to equipment qualification is an increase in the integrated dose contribution from the radioisotopes in the suppression pool fluid. Based on the conclusions from the revised source term rebaselining effort [Reference 3], the AST containment atmosphere gamma and beta integrated doses are expected to be enveloped by those GGNS calculated based on the TID-14844 source term. Even the doses from the suppression pool are enveloped by the original GGNS calculations for a period of about 145 days. The coping duration for EQ purposes at GGNS, however, is 180 days. The integrated dose after 180 days is conservatively estimated to be 12.5% higher than that based on the original source term data per the EQ studies performed in the rebaselining analyses [Reference 13]. GGNS equipment has been qualitatively evaluated to demonstrate that there is adequate margin in the actual test dose to conclude that the equipment would continue to be qualified.

Dose Acceptance Criteria

The dose consequences associated with accident analyses revised for this submittal are presented in terms of "total effective dose equivalent" – TEDE. While the original regulatory guidance had categorized allowable values for thyroid, whole body, and skin doses, the new rule for the Alternative Source Term, 10CFR50.67, establishes the TEDE criteria. In the new rule, the GDC 19 acceptance criteria have been expanded to include a 5 rem TEDE criterion for plants requesting the use of the alternative source term. The GGNS accident analyses in support of this submittal utilize the alternative source term and have been evaluated against the new acceptance criteria. With the issuance of the new Rule, GGNS is able to consider the use of the new dose acceptance criteria without the need for any exemption request to the statements of the original regulations.

Risk Justification

The impact of the proposed changes on the public health risk profiles was evaluated using the GGNS probabilistic safety assessment (PSA) models. The evaluation is summarized in Table 3.

The conclusions from this evaluation are that the public health risk impact in terms of each of the following risk metrics:

- Core Damage Frequency (CDF),
- Large Early Release Frequency (LERF), and
- Latent Cancer Fatalities (LCF)

is negligible. These conclusions are consistent with the risk evaluations/impact of the AST applications discussed in SECY-98-154 and NUREG/CR-6418 [References 3 and 30].

Table 3
Risk Evaluation of Changes

Proposed Change Using AST	Changes or Potential Changes to PSA Model	Resulting Incremental Change to Plant Risk Profile		
		CDF [Level 1]	LERF [Level 2]	LCF [Level 3] [Note 1]
Relaxed secondary containment and control room requirements during refueling	This change does not impact the risk metrics evaluated with GGNS ORAM calculations (core damage risk and boiling risk). ORAM does not calculate the LERF risk profile; however, the LERF profile during refueling is expected to be much lower than that at normal at-power operation. This is primarily due to the energy associated with any release and the time frames associated with core damage and subsequent releases. By definition, LERF applies only when the releases occur prior to evacuation; for refueling conditions, the long time periods before core damage and release can occur allows for prior evacuation.	None	None [Note 2]	None [Note 2]
Relax secondary containment drawdown time by 50%	This change does not affect the CDF models. Secondary containment performance is present as an event tree question in the containment event trees (CET) in the level 2 PSA. The impact of drawdown time is, however, inconsequential to the accident progression as quantified by the CET. This change will not impact any of the questions, accident progression, or branching probabilities in the CETs; therefore, there is no measurable impact on the plant LERF profile	None	None	None
Control Room (CR) inleakage relaxation	The CRFA leakage or the recirculation, filtration, and isolation functions do not contribute to the CDF or LERF in any measurable manner.	None	None	None
Deletion from TS of automatic CR isolation		None	None	None
Deletion from TS of CRFA fans and filters		None	None	None
Increase MSIV leak rate from 25/100 scfh to 100/250 scfh	Impact of MSIV leak rate on the level 2 accident progression phenomena and model results is not measurable. This conclusion is based on the significantly larger severe accident fission product releases involved in the	None	None	None

Proposed Change Using AST	Changes or Potential Changes to PSA Model	Resulting Incremental Change to Plant Risk Profile		
		CDF [Level 1]	LERF [Level 2]	LCF [Level 3] [Note 1]
	LERF metric when compared to the relatively small MSIV leak rates. Therefore, the MSIV leak rates do not contribute to LERF.			
Increase allowable containment leak rate by 10%	This change does not affect • Any of the elements of the CDF (Level 1) PSA model • Any of the containment event tree questions or branching probabilities in the Level 2 PSA model Impact of containment leak rate on the level 2 model is negligible compared to the severe accident releases; increased leakage also has the potential to (very slightly) delay the time at which ultimate containment failure pressure is reached.	None	None	None

Notes for Table 3

1: GGNS has recently completed a limited scope Level 3 PSA study using the MACCS2 computer code. A review of the input to the Level 3 PSA model was performed to assure that there would be no significant changes to these input values as a result of the proposed Tech Spec changes, thus supporting the conclusions in this column, *i.e.,* no change to the latent cancer fatalities.

2: GGNS does not have LERF or Level 3 PSA models for shutdown; these conclusions were extrapolated from the CDF (Level 1) results for this scenario.

Supplemental Risk Discussion – Shutdown Controls

The following discussion of shutdown risk is provided to supplement the analysis and justification of the changes to relax the operational constraints during shutdown. It is applicable primarily to those Technical Specifications affected by proposed changes regarding the terminology "recently irradiated fuel assemblies." This discussion was also included in Reference 16 with the original submittal of similar changes.

The containment and associated engineered safety feature systems are only required by the Technical Specifications during the specific events which are postulated to result in a significant release of radioactivity (e.g., fuel handling accident, drain down). As a result, the requirements of the Technical Specifications are based on the plant being in specified conditions and are not based on providing requirements associated with shutdown risk considerations. Shutdown risk issues are instead addressed by utility outage management programs that follow the guidance of NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management" [Reference 25]. NUMARC

91-06 Section 4.5 discusses the need to assure that secondary containment closure can be achieved to prevent fission product release during severe accidents. NUMARC 91-06 also identifies that the time to effect closure should be consistent with plant conditions (e.g., reactor coolant system inventory and decay heat load). Consistent with the industry's commitment in the letter from NUMARC's President, Mr. Byron Lee, Jr., to Mr. James M. Taylor of the NRC [Reference 26], GGNS has administrative controls in place to meet the recommendations of NUMARC 91-06 Section 4.5 for extended loss of decay heat removal events.

In the draft NUMARC 93-01 guideline, Section 11.2.6.5, "Safety Assessment for Removal of Equipment from Service During Shutdown Conditions," under the subheading of "Containment - Primary (PWR)/Secondary (BWR)", the following guidance is provided.

> "... for plants which obtain amendments to modify Technical Specification requirements on primary or secondary containment operability and ventilation system operability during fuel handling or core alterations, the following guidelines should be included in the assessment of systems removed from service:
>
> - During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay.
>
> - A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure. "

The purpose of the "prompt methods" mentioned above is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored.

In the interim period until the revision to NUMARC 93-01 is endorsed as a formal industry position, GGNS has adopted these provisions for controlling the removal from service of systems, structures and components (SSC's) that are currently required by Technical Specifications during core alteration/fuel handling periods. The GGNS administrative controls include those described in Reference 32.

Also, in accordance with Technical Specification 3.9.6, RPV Water Level – Irradiated Fuel, handling irradiated fuel in the reactor vessel can only occur when the water level in the reactor cavity is at the high water level. Thus, the proposed changes only affect containment requirements during relatively low risk times during refueling outages. Therefore, the proposed changes do not significantly increase the shutdown risk.

Additionally, the proposed Technical Specification changes do not affect the requirements to have the containment systems operable any time the unit is in MODE 1, 2, or 3 regardless of whether fuel handling is occurring in the spent fuel pool.

This change does not impact the GGNS ORAM calculations of risk metrics (core damage risk and boiling risk). ORAM does not calculate the Large Early Release Frequency (LERF) risk profile. Of those accidents during Modes 4 and 5 which are postulated to result in a release, the fuel handling accident produces a small release and the loss of shutdown cooling event is a much more slowly evolving scenario that allows evacuation prior to release. Therefore, the LERF profile during this operation is essentially zero.

RELATED REQUESTS

GGNS has made other submittals related to the revised accident source term or Technical Specifications affected here which are either currently under review by NRR or have been recently approved.

BWROG Report on Gap Release Timing

By letter dated May 6, 1997 (Reference 1), GGNS submitted a report prepared by the BWROG entitled "Prediction of the Onset of Fission Gas Release from Fuel in Generic BWR" (Reference 6). Based on assumptions described in the NUREG, it was recognized that additional analysis was needed to better establish the BWR-specific gap release characteristics. The report presented a conservative analysis determining the minimum time to fuel perforation for a generic BWR following a DBA LOCA with no emergency core cooling system (ECCS) injection. NUREG-1465 assumed the coolant activity phase lasted 30 seconds (based solely on PWR analyses), but recognized that plant specific analyses could justify longer times (see page 8 of Reference 4). The BWROG report was commissioned so that the BWR fleet would not be unduly penalized by the overly conservative assumptions made in NUREG-1465.

While this report was submitted on the GGNS docket, it was intended that the results could be applied on a generic basis and utilized by any BWR. The analysis was performed using a limiting plant configuration and fuel type. NRC-approved codes were used to calculate the minimum duration of the coolant activity phase described in the NUREG. The BWR coolant activity release phase, which represents the period of time from the start of the accident until the initiation of fuel perforation and the attendant gap release, is calculated to last 121 seconds. This conclusion has been utilized in the analyses supporting this full-scope submittal. The inputs, assumptions, and results of the analyses performed for this full-scope submittal are compatible with those in the BWROG report.

Two other letters (References 18 and 23) submitted additional information to the NRC to support the review of the GGNS request for generic approval of the use of the BWROG report. An NRC Safety Evaluation accepting the use of this report for reference by BWRs was issued in Reference 28.

Limited Scope Application of NUREG-1465

Another related outstanding submittal is the limited-scope application of the NUREG-1465 insights presented in letters dated November 3, 1998 and October 6, 1999 (References 2 and 31). That submittal was made as a part of the pilot program to evaluate the use of specific insights in NUREG-1465 to make licensing basis changes. GGNS credited the results of the BWROG report discussed above and proposed an increase in the allowable closure time for those primary containment isolation valves for which the basis for the closure requirement is only loss of coolant accident dose mitigation. That is, valves whose closure times may be restricted based on high energy line break or thermodynamic considerations were not affected by the request. The inputs, assumptions, and results of the analyses performed for this full-scope submittal are compatible with those used in that limited scope application. While NRC approval had not been received when this submittal was being prepared, approval was expected in January 2000.

Fuel Handling Accident Operational Conditions

A submittal (Reference 16) was made to the NRC regarding the results of a revised Fuel Handling Accident analysis performed using the original source term. This request was followed up with supplemental submittals providing information requested by the NRC (References 17 and 32). Based on the assumptions made in that analysis and the acceptable dose results, GGNS had proposed to relax selected constraints imposed during shutdown. The request recognized the benefit of radioactive decay in mitigating the consequences of accidents during shutdown; it was determined that, eight days after shutdown selected safety functions which had been imposed by the Technical Specifications were no longer needed. The requested changes were approved in amendment 139 to the GGNS Operating License.

NO SIGNIFICANT HAZARDS CONSIDERATIONS

This proposed amendment to the Grand Gulf Nuclear Station (GGNS) Technical Specifications (TS) revises those specifications affected by the implementation of the alternative source term concepts in accordance with NUREG 1465. In addition, based on the alternative source term, changes are proposed to selected specifications associated with handling irradiated fuel in the primary or secondary containment and CORE ALTERATIONS. Specifically, the proposal uses a new term to describe irradiated fuel that contains sufficient fission products to require operability of accident mitigation systems to meet the accident analysis assumptions. The alternative source term changes affect the definitions, and the specifications for the Control Room Fresh Air System, MSIV leakage surveillance, Standby Gas Treatment System surveillance, and revises a license condition to increase the allowable control room inleakage. The specifications affected by the relaxation of the shutdown controls include those for the Control Room HVAC system, and the electrical AC Sources, DC Sources and Distribution Systems during shutdown.

The Commission has provided standards for determining whether a no significant hazards consideration exists as stated in 10CFR50.92(c). A proposed amendment to an operating license involves a no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

Entergy Operations Inc. has evaluated the no significant hazards considerations in its request for a license amendment. In accordance with 10CFR50.91(a), Entergy Operations Inc. is providing the analysis of the proposed amendment against the three standards in 10CFR50.92(c). A description of the no significant hazards considerations determination follows:

1. The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated.

   The alternative source term does not affect the design or operation of the facility; rather, once the occurrence of an accident has been postulated the new source term is an input to evaluate the consequences. The implementation of the alternative source term has been evaluated in revisions to the analyses of the limiting design basis accidents at Grand Gulf Nuclear Station. Based on the results of these analyses, it has been demonstrated that, even with the requested Technical Specification and Operating License changes, the dose consequences of these limiting events are within the regulatory guidance currently proposed by the NRC for use with the alternative source term. This guidance is presented in NUREG 1465, in the draft rulemaking for 10CFR50.67, and in the associated draft Regulatory Guide DG-1081.

   A new term to describe irradiated fuel is used to establish operational conditions where specific activities represent situations where significant radioactive releases can be postulated. These operational conditions are consistent with the design basis analysis. Because the equipment affected by the revised operational conditions is not considered an initiator to any previously analyzed accident, inoperability of the equipment cannot increase the probability of any previously evaluated accident. The proposed requirements bound the conditions of the current design basis fuel handling accident analysis which concludes that the radiological consequences are within the acceptance criteria of NUREG 0800, Section 15.7.4 and General Design Criteria 19. As noted above, with the alternative source term implementation, the acceptance criteria are also being revised. The results of the revised Fuel Handling Accident demonstrate that the dose consequences are within the currently proposed NRC regulatory guidance. This guidance is presented in NUREG 1465, in the draft rulemaking for 10CFR50.67, and in the associated draft Regulatory Guide DG-1081.

   Therefore, the proposed changes do not significantly increase the probability or consequences of any previously evaluated accident.

2. The proposed changes would not create the possibility of a new or different kind of accident from any previous analyzed.

   The alternative source term does not affect the design, functional performance, or operation of the facility or of any equipment within the facility. Similarly, it does not affect the design or operation of any equipment or systems involved in the mitigation of any accidents. The proposed changes to the Technical Specifications and the Operating License, while they revise certain performance requirements, do not involve any physical modifications to the plant. Therefore, the proposed changes associated with the alternative source term do not create the possibility of a new or different kind of accident from any previous analyzed.

The new term to describe irradiated fuel is used to establish operational conditions where specific activities represent situations where significant radioactive releases can be postulated. These operational conditions are consistent with the design basis analyses. The relaxation of selected shut down controls has been modeled in revised analyses. The proposed changes do not introduce any new modes of plant operation and do not involve physical modifications to the plant. Therefore, the proposed changes related to shutdown controls based on the alternative source term do not create the possibility of a new or different kind of accident from any previous analyzed.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously analyzed.

3. The proposed changes do not involve a significant reduction in a margin of safety.

The changes above are basically associated with the implementation of a new licensing basis for Grand Gulf Nuclear Station. Approval of the basis change from the original source term in accordance with TID-14844 to the new alternative source term of NUREG-1465 is requested by this submittal. The results of the accident analyses revised in support of this submittal, and considering the requested Technical Specification and Operating License changes, are subject to revised acceptance criteria. These analyses have been performed using conservative methodologies as outlined in the currently proposed regulatory guidance. Safety margins and analytical conservatisms have been evaluated and are well understood. The analyzed events have been carefully selected and margin has been retained to ensure that the analyses adequately bound all postulated event scenarios. The dose consequences of these limiting events are within the acceptance criteria also found in the latest regulatory guidance. This guidance is presented in NUREG 1465, in the approved rulemaking for 10CFR50.67, and in the associated draft Regulatory Guide DG-1081.

The proposed changes continue to ensure that the doses at the exclusion area and low population zone boundaries as well as control room, are within the corresponding regulatory limit. In a similar way, the results of the existing analyses demonstrated that the dose consequences were within the applicable NRC-specified regulatory limit. Specifically, the margin of safety for these accidents is considered to be that provided by meeting the applicable regulatory limit, which, for most events, is conservatively set below the 10CFR100 limit. With respect to the control room personnel doses, the margin of safety is the difference between the 10CFR100 limits and the regulatory limit defined by 10CFR50, Appendix A, Criterion 19 (GDC 19).

Therefore, because the proposed changes continue to result in dose consequences within the applicable regulatory limits, they are considered to not result in a significant reduction in a margin of safety.

Based on the above evaluation, operation in accordance with the proposed amendment involves no significant hazards considerations.

REFERENCES

1) Letter from W.K. Hughey to NRC Document Control Desk; "Submittal of BWROG Report- Prediction of the Onset of Fission Gas Release from Fuel in Generic BWR. Application of NUREG-1465 Source Terms for Grand Gulf Nuclear Station Rebaselining Study", GNRO-97/034, May 6, 1997.

2) Letter from W.K. Hughey to NRC Document Control Desk; "GGNS Pilot Application Submittal of the NUREG-1465 Revised Source Term Insights," GNRO-98/085, November 3, 1998.

3) NRC Letter, L. Joseph Callan to The Commissioners; "Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors", SECY 98-154, June 30, 1998.

4) U.S. Nuclear Regulatory Commission; "Accident Source Terms For Light-Water Nuclear Power Plants", NUREG-1465, February 1995.

5) U.S. Atomic Energy Commission; "Calculation of Distance Factors For Power And Test Reactor Sites", TID-14844, March 1962.

6) General Electric Company Report; "Prediction of the Onset of Fission Gas Release from Fuel in Generic BWR", July 1996.

7) Federal Register, Volume 64, Number 246, pages 71990 through 72002, Final Rule – Use of Alternative Source Terms at Operating Reactors, dated December 23, 1999.

8) GGNS Engineering Report, "Suppression Pool pH and Iodine Re-Evolution Methodology," Revision 1.

9) GGNS Calculation XC-Q1111-98013, Revision 0, "Suppression Pool pH Analysis."

10) GGNS Calculation XC-Q1111-98014, Revision 1, "Doses From Iodine Re-evolution."

11) GGNS Calculation XC-Q1111-98016, Revision 0, "Control Rod Drop Accident Radiological Analysis with Revised Source Terms."

12) GGNS Calculation XC-Q1111-98017, Revision 0, "LOCA Dose Analysis with Revised Source Terms."

13) Sandia National Laboratories, Letter Report, "Evaluation of Radiological Consequences of Design Basis Accidents at Operating Reactors Using the Revised Source Term", dated September 28, 1998.

14) GGNS Calculation XC-Q1111-98019, Revision 0, "Design Basis Fuel Handling Accident Radiological Analysis with Revised Source Terms."

15) NRC Draft Regulatory Guide, DG-1081, "Alternative Radiological Source Terms for Evaluating the Radiological Consequences of Design basis Accidents at Boiling and Pressurized Water Reactors."

16) Letter GNRO-99/00049, Fuel Handling Accident Operational Conditions, Proposed Amendment to the Operating License (LDC 1999-051), dated June 23, 1999.

17) Letter GNRO-99/00063, Information Supporting the Review of Fuel Handling Accident Operational Conditions, Proposed Amendment to the Operating License (LDC 1999-051), dated August 6, 1999.

18) Letter GNRO-94/00131, Fuel Handling Accident Operational Conditions, Proposed Amendment to the Operating License (PCOL-93/08), dated November 9, 1994.

19) Letter GNRO-95/00090, Fuel Handling Accident Operational Conditions, Proposed Amendment to the Operating License (PCOL-93/08 Revision 1), dated August 4, 1995.

20) Letter GNRO-96/00048, Fuel Handling Accident Operational Conditions, Proposed Amendment to the Operating License, Additional Information, Dated April 24, 1996.

21) Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", Second Printing 1989.

22) Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil", 1993.

23) Letter from William Rasin, Duke Engineering and Services, to the Document Control Desk, Information Supporting the Review of Generic Alternate Source Term Request", referencing the GGNS Docket 50-416, dated July 16, 1999.

24) GGNS Updated Final Safety Analysis Report, various sections.

25) NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management, December, 1991.

26) Letter GNRO-99/00057, Information Supporting Review of Generic Alternate Source Term Request, dated July 14, 1999.

27) General Electric Technical Report, NEDC-31858P, BWROG Report For Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems.

28) NRC Letter to W.A. Eaton from S.P. Sekerak, "Acceptance of Boiling Water Reactor Owners' Group (BWROG) Report, "Prediction of the Onset of Fission Gas Release from Fuel in Generic BWR," report dated July 1996, letter dated September 9, 1999.

29) NRC Letter to W. Cottle from T.R. Quay, "Revocation of the Construction Permit," dated August 21, 1991 (assigned letter number GNRI-91/00176).

30) NUREG/CR-6418, "Risk-Importance of Containment and Related ESF System Performance Requirements," prepared by H.P. Nourbaksh, et al.

31) Letter GNRO-99/00077, "Pilot Limited Scope Application of NUREG-1465 Alternative Source Term Insights," dated October 6, 1999.

32) Letter GNRO-99/00075, "Fuel Handling Accident Operational Conditions," dated October 4, 1999.

Attachment 2

Markups of Affected Technical Specification and Operating License Pages

Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989.

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites."
EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME	The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial movement of the associated turbine stop valve or the turbine control valve to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured, except for the breaker arc suppression time, which is not measured but is validated to conform to the manufacturer's design value.
ISOLATION SYSTEM RESPONSE TIME	The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
L_a	The maximum allowable primary containment leakage rate, L_a, shall be ~~0.437%~~ 0.682% of primary containment air weight per day at the calculated peak containment pressure (P_a).

(continued)

3.3 INSTRUMENTATION

3.3.7.1 Control Room Fresh Air (CRFA) System Instrumentation

LCO 3.3.7.1 The CRFA System instrumentation for ~~each Function in Table 3.3.7.1-1~~ manual isolation shall be OPERABLE.

APPLICABILITY: ~~According to Table 3.3.7.1-1.~~ Modes 1,2, and 3
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 ~~Enter the Condition referenced in Table 3.3.7.1-1 for the channel.~~ Place channel in trip.	~~Immediately~~ 24 hours
B. ~~As required by Required Action A.1 and referenced in Table 3.3.7.1-1.~~ Required Action and associated Completion Time	~~B.1 --------NOTE-------- Only applicable for Function 1. ---------------------- Declare associated CRFA subsystem inoperable.~~ B.1 Close associated isolation dampers	~~1 hour from discovery of loss of CRFA initiation capability in both trip systems~~ 1 hour
	~~AND~~	
	~~B.2 Place channel in trip.~~	~~24 hours~~

~~(continued)~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	C.1	Declare associated CRFA subsystem inoperable.	1 hour from discovery of loss of CRFA initiation capability in both trip systems
	<u>AND</u>		
	C.2	Place channel in trip.	12 hours
D. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	D.1	Declare associated CRFA subsystem inoperable.	1 hour from discovery of loss of CRFA initiation capability in both trip systems
	<u>AND</u>		
	D.2	Place channel in trip.	6 hours
E. Required Action and associated Completion Time of Condition B, C, or D not met.	E.1	Place the associated CRFA subsystem in the isolation mode of operation.	1 hour
	<u>OR</u>		
	E.2	Declare associated CRFA subsystem inoperable.	1 hour

DELETED

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------

1. ~~Refer to Table 3.3.7.1-1 to determine which SRs apply for each Function.~~

~~2.~~ When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided ~~the associated Function maintains CRFA initiation capability.~~ CR isolation capability is maintained.

SURVEILLANCE		FREQUENCY
~~SR 3.3.7.1.1~~	~~Perform CHANNEL CHECK.~~	~~12 hours~~
~~SR 3.3.7.1.2~~	~~Perform CHANNEL FUNCTIONAL TEST.~~	~~92 days~~
~~SR 3.3.7.1.3~~	~~Calibrate the trip units.~~ DELETED	~~92 days~~
~~SR 3.3.7.1.4~~	~~Perform CHANNEL CALIBRATION.~~	~~12 months~~
~~SR 3.3.7.1.5~~	~~Perform CHANNEL CALIBRATION.~~	~~18 months~~
SR 3.3.7.1.~~6~~ 1	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months

Table 3.3.7.1-1 (page 1 of 1)
Control Room Fresh Air System Instrumentation

DELETED

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level – Low Low, Level 2	1,2,3 (a)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.5 SR 3.3.7.1.6	≥ -43.8 inches
2. Drywell Pressure – High	1,2,3	2	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 1.43 psig
3. Control Room Ventilation Radiation Monitors	1,2,3 (a),(b)	2	D	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.6	≤ 5 mR/hr
4. Manual Initiation	1,2,3 (a),(b)	2	B	SR 3.3.7.1.6	NA

(a) During operations with a potential for draining the reactor vessel.

(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in the primary or secondary containment.

TABLE TR3.3.7.1-1

TECHNICAL SPECIFICATION CONTROL ROOM FRESH AIR SYSTEM INSTRUMENTATION TRIP SETPOINTS

DELETED

FUNCTION	TRIP SETPOINT
1. Reactor Vessel Water Level Low Low, Level 2	≥ -41.6 inches*
2. Drywell Pressure - High	≤ 1.23 psig
3. Control Room Ventilation Radiation Monitors	≤ 5 mR/hr
4. Manual Initiation	NA

* See Bases Figure B 3.3.1.1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	18 months
SR 3.6.1.3.8	-----NOTE----- Only required to be met in MODES 1, 2, and 3. ----- Verify leakage rate through all four main steam lines is ≤ ~~100~~ 250 scfh when tested at ≥ P_a.	In accordance with 10 CFR 50, Appendix J, Testing Program
SR 3.6.1.3.9	-----NOTE----- Only required to be met in MODES 1, 2, and 3. ----- Verify combined leakage rate of 1 gpm times the total number of PCIVs through hydrostatically tested lines that penetrate the primary containment is not exceeded when these isolation valves are tested at ≥ 1.1 P_a.	In accordance with 10 CFR 50, Appendix J, Testing Program

leakage rate through each main steam line is ≤ 100 scfh when tested at ≥ P_a, AND the total

Fig 2603 Am 135

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.3	Verify each standby gas treatment (SGT) subsystem will draw down the secondary containment to ≥ 0.25 inch of vacuum water gauge in ≤ ~~120~~ 180 seconds.	18 months on a STAGGERED TEST BASIS
SR 3.6.4.1.4	Verify each SGT subsystem can maintain ≥ 0.266 inch of vacuum water gauge in the secondary containment for 1 hour at a flow rate ≤ 4000 cfm.	18 months on a STAGGERED TEST BASIS

3.7 PLANT SYSTEM

3.7.3 Control Room Fresh Air (CRFA) System

LCO 3.7.3 ~~Two CRFA subsystems shall be OPERABLE.~~
Each Control Room isolation damper shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the primary or secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more flowpaths ~~CRFA subsystem inoperable.~~ with one isolation damper inoperable.	A.1 ~~Restore CRFA subsystem to OPERABLE status.~~ Isolate the affected flowpath	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

------ NOTES ------

1. Control Room inlet and exhaust flowpaths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each flowpath.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during OPDRVs. ~~movement of irradiated fuel assemblies in the primary or secondary containment, during CORE ALTERATIONS, or during OPDRVs.~~	---------------NOTE--------------- LCO 3.0.3 is not applicable.	
	~~C.1 Place OPERABLE CRFA subsystem in isolation mode.~~	~~Immediately~~
	~~OR~~	
	~~C.2.1 Suspend movement of irradiated fuel assemblies in the primary and secondary containment.~~	~~Immediately~~
	~~AND~~	
	~~C.2.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
~~D. Two CRFA subsystems inoperable in MODE 1, 2, or 3.~~	~~D.1 Enter LCO 3.0.3.~~	~~Immediately~~

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(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Two CRFA subsystems inoperable during movement of irradiated fuel assemblies in the primary or secondary containment, during CORE ALTERATIONS, or during OPDRVs.	E.1	Suspend movement of irradiated fuel assemblies in the primary and secondary containment.	Immediately
	<u>AND</u>		
	E.2	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	E.3	Initiate action to suspend OPDRVs.	Immediately

DELETE

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Operate each CRFA subsystem for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.3.2	Perform required CRFA filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.3.~~3~~ 1	Verify each CRFA isolation damper closes ~~subsystem actuates~~ on an actual or simulated initiation signal.	18 months

DELETED

3.7 PLANT SYSTEMS

3.7.4 Control Room Air Conditioning (AC) System

LCO 3.7.4 Two control room AC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the primary or secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One control room AC subsystem inoperable.	A.1	Restore control room AC subsystem to OPERABLE status.	30 days
B. Two control room AC subsystems inoperable.	B.1	Verify control room area temperature ≤ 90°F.	Once per 4 hours
	AND		
	B.2	Restore one control room AC subsystem to OPERABLE status.	7 days
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1	Be in MODE 3.	12 hours
	AND		
	C.2	Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A not met during OPDRVs. ~~movement of irradiated fuel assemblies in the primary or secondary containment, during CORE ALTERATIONS, or during OPDRVs.~~	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	D.1 Place OPERABLE control room AC subsystem in operation.	Immediately
	<u>OR</u>	
	D.2~~.1~~ ~~Suspend movement of irradiated fuel assemblies in the primary and secondary containment.~~	~~Immediately~~
	~~<u>AND</u>~~	
	~~D.2.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~<u>AND</u>~~	
	D.2.3 Initiate action to suspend OPDRVs.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition B not met during OPDRVs. ~~movement of irradiated fuel assemblies in the primary or secondary containment, during CORE ALTERATIONS, or during OPDRVs.~~	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	~~E.1 Suspend movement of irradiated fuel assemblies in the primary and secondary containment.~~	~~Immediately~~
	~~AND~~	
	~~E.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	E.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify each control room AC subsystem has the capability to remove the assumed heat load.	18 months

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources—Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems—Shutdown"; and

b. One diesel generator (DG) capable of supplying one division of the Division 1 or 2 onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.8; and

c. One qualified circuit, other than the circuit in LCO 3.8.2.a, between the offsite transmission network and the Division 3 onsite Class 1E electrical power distribution subsystem, or the Division 3 DG capable of supplying the Division 3 onsite Class 1E AC electrical power distribution subsystem, when the Division 3 onsite Class 1E electrical power distribution subsystem is required by LCO 3.8.8.

APPLICABILITY: MODES 4 and 5,
During movement of irradiated fuel assemblies in the primary or secondary containment.

recently

ACTIONS

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

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO Item a not met.	------------NOTE------------ Enter applicable Condition and Required Actions of LCO 3.8.8, when any required division is de-energized as a result of Condition A. ----------------------------	
	A.1 Declare affected required feature(s) with no offsite power available from a required circuit inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
recently	A.2.2 Suspend movement of irradiated fuel assemblies in the primary and secondary containment.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.4	Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately
B. LCO Item b not met.	B.1	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
recently	B.2	Suspend movement of irradiated fuel assemblies in primary and secondary containment.	Immediately
	<u>AND</u>		
	B.3	Initiate action to suspend OPDRVs.	Immediately
	<u>AND</u>		
	B.4	Initiate action to restore required DG to OPERABLE status.	Immediately
C. LCO Item c not met.	C.1	Declare High Pressure Core Spray System inoperable.	72 hours

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources—Shutdown

LCO 3.8.5 The following shall be OPERABLE:

a. One Class 1E DC electrical power subsystem capable of supplying one division of the Division 1 or 2 onsite Class 1E DC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems - Shutdown";

b. One Class 1E battery or battery charger, other than the DC electrical power subsystem in LCO 3.8.5.a, capable of supplying the remaining Division 1 or 2 onsite Class 1E DC electrical power distribution subsystem(s) when required by LCO 3.8.8; and

c. The Division 3 DC electrical power subsystem capable of supplying the Division 3 onsite Class 1E DC electrical power distribution subsystem, when the Division 3 onsite Class 1E DC electrical power distribution subsystem is required by LCO 3.8.8.

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the primary or secondary containment.

ACTIONS

-----------------------------------NOTE-----------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required battery charger inoperable.	------------NOTE------------ Entry into MODE 4 or 5, or commencing movement of recently irradiated fuel is not allowed, except entry into MODE 4 or 5 can be made as part of a unit shutdown. ----------------------------	
	A.1 Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	1 hour AND Once per 8 hours thereafter
B. Required Action and associated Completion Time of Condition A not met.	B.1 Declare associated battery inoperable.	Immediately
C. One or more required DC electrical power subsystems inoperable for reasons other than Condition A.	C.1 Declare affected required feature(s) inoperable.	Immediately
	OR	
	C.2.1 Suspend CORE ALTERATIONS. AND	Immediately
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued) recently	C.2.2 Suspend movement of irradiated fuel assemblies in the primary and secondary containment.	Immediately
	<u>AND</u>	
	C.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	<u>AND</u>	
	C.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	-------------------NOTE------------------ The following SRs are not required to be performed: SR 3.8.4.4, SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8. ---	
	For DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.4 SR 3.8.4.7 SR 3.8.4.2 SR 3.8.4.5 SR 3.8.4.8. SR 3.8.4.3 SR 3.8.4.6	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Distribution Systems—Shutdown

LCO 3.8.8 The necessary portions of the Division 1, Division 2, and Division 3 AC and DC electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the primary or secondary containment.

ACTIONS

-----------------------------------NOTE-----------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required AC or DC electrical power distribution subsystems inoperable.	A.1	Declare associated supported required feature(s) inoperable.	Immediately
	<u>OR</u>		
	A.2.1	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	A.2.2	Suspend movement of recently irradiated fuel assemblies in the primary and secondary containment.	Immediately
	<u>AND</u>		
			(continued)

(a) Include an emergency override of the test mode of the Division 3 HPCS diesel generator to permit response to emergency signals and to return the control of the diesel generator to the emergency standby mode. (Item No. 333, TS 4.8.1.1.2.d.12.b)

(b) Provide the second level undervoltage protection for Division 3 power supply (Item No. 373, TS Table 3.3.3-2).

(c) Incorporate a bypass or coincident logic in all Division 1 and 2 diesel generator protective trips, except for trips on diesel engine overspeed and generator differential current (Item No. 808, TS 4.8.1.1.2.d.16.d).

(38) Control Room Leak Rate (Section 6.2.6, SSER #6)

EOI shall operate Grand Gulf Unit 1 during Modes 1, 2, and 3 with an allowable control room leak rate not to exceed ~~590~~ 1200 cfm. ~~Upon restart of construction of Unit 2 control room, EOI will be permitted to operate at a leak rate of 760 cfm as evaluated in SSER No. 6.~~

(39) Temporary Secondary Containment Boundary Change

For a period of time not to exceed 144 cumulative hours, the provisions of Specification 3/4.6.6.1 may be applied to the railroad bay area including the exterior railroad bay door on the auxiliary building in lieu of the present secondary containment boundaries that isolate the railroad bay area. While the railroad bay area is being used as a secondary containment boundary, the railroad bay door may be opened for the purpose of moving trucks in and out provided the four hour limitation in ACTION a of Technical Specification 3.6.6.1 is reduced to one hour. A fire watch shall be established in the railroad bay area while the door is being used as a secondary containment boundary.

(40) Temporary Ultimate Heat Sink Change

With the plant in OPERATIONAL condition 4, SSW cooling tower basin A may be considered OPERABLE in accordance with Technical Specification 3.7.1.3 with less than a 30 day supply of water (without makeup) during the time that SSW basin B is drained to replace its associated service water pump provided:

(a) SSW basin A water level is maintained greater than or equal to 87".

(b) At least two sources of water (other than normal makeup with one source not dependent on offsite power) are available for makeup to SSW basin A.

This license condition may remain in effect until plant startup following the outage scheduled for fall 1985.

Attachment 3

Markups of Affected Technical Specification Bases Pages

(for Information)

BASES

LCO 3.0.4 (continued)

provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.4 is only applicable when entering MODE 3 from MODE 4, MODE 2 from MODE 3 or 4, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, or 3. The requirements of LCO 3.0.4 do not apply in MODES 4 and 5, or other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual Specifications sufficiently define the remedial measure to be taken.

The ACTIONS for an inoperable required battery charger in LCO 3.8.4, "DC Sources - Operating," and LCO 3.8.5, "DC Sources - Shutdown," include a Note explicitly precluding entry into specific MODEs or other specified conditions of the Applicability while relying on the ACTIONS. With an inoperable required battery charger this Note in LCO 3.8.4 prohibits entry in MODE 1, 2, or 3, except during power decrease and in LCO 3.8.5 prohibits starting movement of recently irradiated fuel, entering MODE 4 from MODE 5, or loading fuel into the vessel if the vessel is defueled.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, either in compliance with LCO 3.0.4, or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.g. Containment and Drywell Ventilation Exhaust Radiation – High (continued)

Four channels of Containment and Drywell Ventilation Exhaust – High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. Two upscale-Hi Hi, one upscale-Hi Hi and one downscale, or two downscale signals from the same trip system actuate the trip system and initiate isolation of the associated containment and drywell isolation valves.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding and to ensure offsite doses remain below 10 CFR 20 and 10 CFR 100 limits.

The Function is required to be OPERABLE during operations with a potential for draining the reactor vessel (OPDRVs) and movement of recently irradiated fuel assemblies in the primary or secondary containment because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure offsite dose limits are not exceeded. Due to radioactive decay, this Function is only required to isolate primary containment during those fuel handling accidents involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~8 days~~ 7 days).

This Function isolates the Group 7 valves.

2.h. Manual Initiation

The Manual Initiation push button channels introduce signals into the primary containment and drywell isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific UFSAR safety analysis that takes credit for this Function. It is retained for the isolation function as required by the NRC in the plant licensing basis.

There are four push buttons for the logic, two manual initiation push buttons per trip system. There is no

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3, 4. Fuel Handling Area Ventilation and Pool Sweep Exhaust Radiation – High High (continued)

channels of Fuel Handling Area Ventilation Exhaust Radiation – High High Function and four channels of Fuel Handling Area Pool Sweep Exhaust Radiation – High High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

The Exhaust Radiation – High High Functions are required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, these Functions are not required. In addition, the Functions are required to be OPERABLE during OPDRVs and movement of recently irradiated fuel assemblies in the primary or secondary containment because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded. Due to radioactive decay, these Functions are only required to isolate secondary containment during those fuel handling accidents involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~8 days~~ 7 days).

5. Manual Initiation

The Manual Initiation push button channels introduce signals into the secondary containment isolation logic that are redundant to the automatic protective instrumentation channels, and provide manual isolation capability. There is no specific UFSAR safety analysis that takes credit for this Function. It is retained for the secondary containment isolation instrumentation as required by the NRC approved licensing basis.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The PCIVs LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory, and establishing the primary containment boundary during major accidents. As part of the primary containment boundary, PCIV OPERABILITY supports leak tightness of primary containment. Therefore, the safety analysis of any event requiring isolation of primary containment is applicable to this LCO.

The DBAs that result in a release of radioactive material for which the consequences are mitigated by PCIVs are a loss of coolant accident (LOCA), a main steam line break (MSLB), and a fuel handling accident involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~8 days~~ 7 days) inside primary containment (Refs. 1 and 2). In the analysis for each of these accidents, it is assumed that PCIVs are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through PCIVs are minimized. Of the events analyzed in Reference 1, the LOCA is the most limiting event due to radiological consequences. An analysis of the affect of the purge valves being open at the initiation of a LOCA has been performed. This condition was found to result in dose contributions of a small fraction of 10 CFR 100. It is assumed that the primary containment is isolated such that release of fission products to the environment is controlled.

PCIVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

PCIVs form a part of the primary containment boundary and some also form a part of the RCPB. The PCIV safety function is related to minimizing the loss of reactor coolant inventory, and establishing the primary containment boundary during a DBA.

The power operated isolation valves are required to have isolation times within limits. Additionally, power operated automatic valves are required to actuate on an automatic isolation signal.

(continued)

LCO (continued)

are listed with their associated stroke times in the applicable plant procedures. Purge valves with resilient seals, MSIVs, and hydrostatically tested valves must meet additional leakage rate requirements. Other PCIV leakage rates are addressed by LCO 3.6.1.1, "Primary Containment," as Type B or C testing.

Valves on the containment airlock bulkhead have a design function as a primary containment isolation when the airlock inner door is inoperable per LCO 3.6.1.2 or during performance of airlock barrel testing or pneumatic tubing testing or at any time the inner airlock door/bulkhead is breached. However, these valves are Primary Containment Isolation Valves as required by LCO 3.6.1.3 at all times.

This LCO provides assurance that the PCIVs will perform their designed safety functions to minimize the loss of reactor coolant inventory, and establish the primary containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, most PCIVs are not required to be OPERABLE. Certain valves are required to be OPERABLE, however, to prevent a potential flow path (the RHR Shutdown Cooling System suction from the reactor vessel) from lowering reactor vessel level to the top of the fuel. These valves are those whose associated isolation instrumentation is required to be OPERABLE according to LCO 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation," Function 5.b. Additional valves are required to be OPERABLE to prevent release of radioactive material during a postulated fuel handling accident involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~8 days~~ 7 days). These valves are those whose associated isolation instrumentation is required to be OPERABLE according to LCO 3.3.6.1, "Function 2.g." (This does not include the valves that isolate the associated instrumentation.)

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow path(s) to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.7 (continued)

each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.1.7 overlaps this SR to provide complete testing of the safety function. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.8

Leakage through any single main steam line must be ~~less than~~ ≤ 100 scfh when tested at a pressure of 11.5 psig.

250

The analyses in Reference 2 is based on leakage that is less than the specified leakage rate. Leakage through all four steam lines must be ≤ ~~100~~ scfh when tested at P_a ~~P_t~~ (11.5 psig). The MSIV leakage rate must be verified to be in accordance with the leakage test requirements of Reference 3, as modified by approved exemptions. A Note is added to this SR which states that these valves are only required to meet this leakage limit in MODES 1, 2 and 3. In the other conditions, the Reactor Coolant System is not pressurized and specific primary containment leakage limits are not required.

SR 3.6.1.3.9

Surveillance of hydrostatically tested lines provides assurance that the calculation assumptions of Reference 2 is met.

This SR is modified by a Note that states these valves are only required to meet the combined leakage rate in MODES 1, 2, and 3 since this is when the Reactor Coolant System is

(continued)

BASES

BACKGROUND (continued)

2. closed by a manual valve, blind flange, rupture disk, or de-activated automatic valve or damper secured in a closed position, except as provided in LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)";

b. All auxiliary building and enclosure building equipment hatches and blowout panels are closed and sealed;

c. The door in each access to the auxiliary building and enclosure building is closed, except for normal entry and exit;

d. The sealing mechanism, e.g., welds, bellows, or O-rings, associated with each secondary containment penetration is OPERABLE; and

e. The standby gas treatment system is OPERABLE, except as provided in LCO 3.6.4.3, "Standby Gas Treatment System."

APPLICABLE SAFETY ANALYSES

7 days

There are three principal accidents for which credit is taken for secondary containment OPERABILITY. These are a LOCA (Ref. 1), a fuel handling accident involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~8 days~~) inside primary containment (Ref. 2), and a fuel handling accident involving the handling of recently irradiated fuel in the auxiliary building (Ref. 3). The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis, and that fission products entrapped within the secondary containment structure will be treated by the SGT System prior to discharge to the environment.

Secondary containment satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES

LCO (continued)

to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained.

APPLICABILITY

In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY.

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the primary or secondary containment. Due to radioactive decay, secondary containment is required to be OPERABLE only during that fuel movement involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~8 days~~ 7 days).

ACTIONS

A.1

If secondary containment is inoperable, it must be restored to OPERABLE status within 4 hours. The 4 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary containment during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring secondary containment OPERABILITY) occurring during periods where secondary containment is inoperable is minimal.

B.1 and B.2

If the secondary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.1.3 and SR 3.6.4.1.4

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. To ensure that all fission products are treated, SR 3.6.4.1.3 verifies that the SGT System will rapidly establish and maintain a pressure in the secondary containment that is less than the lowest postulated pressure external to the secondary containment boundary. This is confirmed by demonstrating that one OPERABLE SGT subsystem will draw down the secondary containment to ≥ 0.25 inches of vacuum water gauge in ≤ ~~120~~ 180 seconds. This cannot be accomplished if the secondary containment boundary is not intact. SR 3.6.4.1.4 demonstrates that each OPERABLE SGT subsystem can maintain ≥ 0.266 inches of vacuum water gauge for 1 hour at a flow rate ≤ 4000 cfm. The 1 hour test period allows secondary containment to be in thermal equilibrium at steady state conditions. Therefore, these two tests are used to ensure secondary containment boundary integrity. Since these SRs are secondary containment tests, they need not be performed with each SGT subsystem. The SGT subsystems are tested on a STAGGERED TEST BASIS, however, to ensure that in addition to the requirements of LCO 3.6.4.3, either SGT subsystem will perform this test. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 15.6.5.
2. UFSAR, Section 15.7.6.
3. UFSAR, Section 15.7.4.

BASES

BACKGROUND (continued)

Analyses have shown that in addition to building leakage paths, the Standby Gas Treatment System (SGTS) has the capacity to maintain secondary containment negative pressure assuming the failure of all nonqualified lines 2 inches and smaller or with the failure of a single nonisolated line as large as 4 inches. As a result, the following lines which penetrate the secondary containment and terminate there (i.e., they do not continue through the secondary containment and also penetrate the primary containment) are provided with a single isolation valve, rather than two, at the secondary penetration:

a. 4-inch makeup water supply line

b. 3-inch domestic water supply line

c. 4-inch RHR backwash line

d. 3-inch backwash transfer pump discharge line

e. 3-inch floor and equipment drain line

The single isolation valve for each of the above lines is an air-operated valve which fails closed; in addition, each operator is provided with redundant solenoid valves which receive actuation signals from redundant sources. In this manner, it is ensured that, given any single failure, only one of the above lines will be nonisolated, which as stated above is within the capacity of the SGTS.

APPLICABLE SAFETY ANALYSES

7 days

The SCIVs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The principal accidents for which the secondary containment boundary is required are a loss of coolant accident (Ref. 1), a fuel handling accident involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~8 days~~) inside primary containment (Ref. 3), and a fuel handling accident involving the handling of recently irradiated fuel in the auxiliary building (Ref. 4). The secondary containment performs no active function in response to each of these limiting

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Maintaining SCIVs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment so that they can be treated by the SGT System prior to discharge to the environment.

SCIVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

SCIVs form a part of the secondary containment boundary. The SCIV safety function is related to control of offsite radiation releases resulting from DBAs.

The power operated isolation dampers and valves are considered OPERABLE when their isolation times are within limits. Additionally, power operated automatic dampers and valves are required to actuate on an automatic isolation signal.

The normally closed isolation dampers and valves, rupture disks, or blind flanges are considered OPERABLE when manual dampers and valves are closed or open in accordance with appropriate administrative controls, automatic dampers and valves are de-activated and secured in their closed position, rupture disks or blind flanges are in place. The SCIVs covered by this LCO, along with their associated stroke times, if applicable, are listed in the applicable plant procedures.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, OPERABILITY of SCIVs is required.

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies. Moving recently irradiated fuel assemblies in the primary or secondary containment may also occur in MODES 1, 2, and 3.

(i.e., fuel that has occupied part of a critical reactor core within the previous 7 days)

(continued)

BASES

BACKGROUND (continued)

humidity of the airstream to less than 70% (Ref. 2). The prefilter removes large particulate matter, while the HEPA filter is provided to remove fine particulate matter and protect the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the final HEPA filter is provided to collect any carbon fines exhausted from the charcoal adsorber.

The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, both enclosure building recirculation fans and both charcoal filter train fans start. SGT System flows are controlled by modulating inlet vanes installed on the charcoal filter train exhaust fans and two position volume control dampers installed in branch ducts to individual regions of the secondary containment.

APPLICABLE SAFETY ANALYSES

The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident and fuel handling accidents. Due to radioactive decay, the SGT System is required to be OPERABLE to mitigate only those fuel handling accidents involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~8 days~~ 7 days) (Ref. 2). For all events analyzed, the SGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment.

The SGT System satisfies Criterion 3 of the NRC Policy Statement.

LCO

Following a DBA, a minimum of one SGT subsystem is required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two operable subsystems ensures operation of at least one SGT subsystem in the event of a single active failure.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, SGT System OPERABILITY is required during these MODES.

(continued)

BASES

APPLICABILITY (continued)

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT System OPERABLE is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the primary or secondary containment. Due to radioactive decay, the SGT System is required to be OPERABLE only during fuel movement involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~8 days~~ 7 days).

ACTIONS

A.1

With one SGT subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE SGT subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SGT subsystem and the low probability of a DBA occurring during this period.

B.1 and B.2

If the SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.3 Control Room Fresh Air (CRFA) System

BASES

BACKGROUND

The CRFA System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA). INSERT B3.7-11A

The safety related function of the CRFA System used to control radiation exposure consists of two independent and redundant high efficiency air filtration subsystems for treatment of recirculated air or outside supply air. Each subsystem consists of a demister, an electric heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a second HEPA filter, a fan, and the associated ductwork and dampers. Demisters remove water droplets from the airstream. Prefilters and HEPA filters remove particulate matter that may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay.

INSERT B3.7-11B

~~In addition to the safety related standby emergency filtration function,~~ parts of the CRFA System are operated to maintain the control room environment during normal operation. Upon receipt of the initiation signal(s) (indicative of conditions that could result in radiation exposure to control room personnel), the CRFA System automatically switches to the isolation mode of operation to prevent infiltration of contaminated air into the control room. A system of dampers isolates the control room; ~~and~~ control room air flow ~~is~~ may be recirculated and processed through either of the two filter subsystems.

The CRFA System is designed to maintain the control room environment for a 30 day continuous occupancy after a DBA, per the requirements of GDC 19. CRFA System operation in maintaining the control room habitability is discussed in the UFSAR, Sections 6.5.1 and 9.4.1 (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The ability of the CRFA System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the UFSAR, Chapters 6 and 15 (Refs. 3 and 4, respectively). ~~The isolation mode of the CRFA System~~

(continued)

Proposed inserts to Bases for TS 3.7.3

INSERT B 3.7-11 A

Redundant isolation dampers in each inlet and exhaust flow path. The system also includes

Insert B 3.7-11 B

With the implementation of the alternative source term (Reference 7), the filtration function is no longer credited in the accident analyses and is not a safety-related function.

Insert B 3.7-12 A

The CRFA System is assumed to isolate the control room following a loss of coolant accident, main steam line break, and control rod drop accident. Analyses of these events have assumed the control room would be isolated for at least three days. At that time, isolation was terminated and the control room was again ventilated with unfiltered outside air. Safety analysis of the fuel handling accident has demonstrated that control room isolation is not required for this accident.

BASES

REPLACE w/ INSERT B 3.7-12 A

APPLICABLE SAFETY ANALYSES (continued)

is assumed to operate following a loss of coolant accident, main steam line break, fuel handling accident, and control rod drop accident. The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference 4. No single active or passive failure will cause the loss of outside or recirculated air from the control room.

The CRFA System satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two redundant dampers in each ventilation flowpath ~~subsystems of the CRFA System~~ are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other ~~subsystem.~~ damper. ~~Total system~~ Failure of both dampers could result in a failure to meet the dose requirements of GDC 19 in the event of a DBA.

~~The CRFA System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated:~~

~~a. Fan is OPERABLE;~~

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

~~c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.~~

DELETED

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors. The control room boundary is maintained when the boundary can be rapidly isolated and established to meet in-leakage limits as outlined in Ref. 6.

APPLICABILITY

In MODES 1, 2, and 3, the CRFA System must be OPERABLE to control operator exposure during and following a DBA, since the DBA could lead to a fission product release.

In MODES 4 and 5, the probability and consequences of a DBA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the CRFA System

(continued)

BASES

isolation function

APPLICABILITY (continued)

OPERABLE is not required in MODE 4 or 5, except ~~for the following situations under which significant radioactive releases can be postulated:~~

~~a.~~ During operations with a potential for draining the reactor vessel (OPDRVs) ~~;~~ and

b. ~~During CORE ALTERATIONS; and~~

~~c.~~ During movement of recently irradiated fuel assemblies in the primary or secondary containment.

ACTIONS

A.1

With one CRFA subsystem inoperable, the inoperable CRFA subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE CRFA subsystem is adequate to perform control room radiation protection. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of CRFA System function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining subsystem can provide the required capabilities.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable ~~CRFA subsystem~~ isolation damper cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 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.

C.1~~, C.2.1, C.2.2, and C.2.3~~

~~The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations.~~

(continued)

BASES

ACTIONS

C.1~~, C.2.1, C.2.2, and C.2.3~~ ℓ (continued)

~~Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.~~

During ~~movement of irradiated fuel assemblies in the primary or secondary containment, during CORE ALTERATIONS, or during~~ OPDRVs, if the inoperable ~~, CRFA subsystem~~ cannot be restored to OPERABLE status within the required Completion Time, the ~~OPERABLE CRFA subsystem may be placed in the isolation mode.~~

isolation damper

This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the primary and secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

unit must be placed in a condition that minimizes risk. To achieve this status, activities that present a potential for a significant release of radioactivity must be suspended immediately.

D.1

If both CRFA subsystems are inoperable in MODE 1, 2, or 3, the CRFA System may not be capable of performing the intended function and the unit is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

E.1, E.2, and E.3

During movement of irradiated fuel assemblies in the primary or secondary containment, during CORE ALTERATIONS, or during

(continued)

BASES

ACTIONS

E.1, E.2, and E.3 (continued)

OPDRVs, with two CRFA subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the primary and secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. If applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that a subsystem in a standby mode starts from the control room on demand and continues to operate. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal operating conditions of this system are not severe, testing each subsystem once every month provides an adequate check on this system. Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. Systems with heaters must be operated for $\geq$ 10 continuous hours with the heaters energized. Furthermore, the 31 day Frequency is based on the known reliability of the equipment and the two subsystem redundancy available.

SR 3.7.3.2

This SR verifies that the required CRFA testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CRFA filter tests are in accordance with Regulatory Guide 1.52 (Ref. 5). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.7.3.2 (continued)~~

~~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.~~

~~SR 3.7.3.3~~

This SR verifies that each ~~CRFA subsystem starts and operates and that the isolation valves~~ of the required dampers closes in ≤ ~~4~~ 10 seconds on an actual or simulated initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.7.1.~~5~~1 overlaps this SR to provide complete testing of the safety function. While this Surveillance can be performed with the reactor at power, operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 6.5.1.
2. UFSAR, Section 9.4.1.
3. UFSAR, Chapter 6.
4. UFSAR, Chapter 15.
5. Regulatory Guide 1.52, Revision 2, March 1978.
6. Engineering Evaluation Request 95/6213, Engineering Evaluation Request Response Partial Response dated 12/18/95.
7. Amendment ___ to GGNS Operating License regarding Alternative Source Term.

B 3.7 PLANT SYSTEMS

B 3.7.4 Control Room Air Conditioning (AC) System

BASES

BACKGROUND	The Control Room AC System provides temperature control for the control room. The Control Room AC System consists of two independent, redundant subsystems that provide cooling and heating of recirculated control room air. Each subsystem consists of heating coils, cooling coils, fans, chillers, compressors, ductwork, dampers, and instrumentation and controls to provide for control room temperature control. The Control Room AC System is designed to provide a controlled environment under both normal and accident conditions. The Control Room AC System operation in maintaining the control room temperature is discussed in the UFSAR, Sections 6.4 and 9.4.1 (Refs. 1 and 2, respectively).
APPLICABLE SAFETY ANALYSES	The design basis of the Control Room AC System is to maintain the control room temperature for a 30 day continuous occupancy. The Control Room AC System components are arranged in redundant safety related subsystems. During emergency operation, the Control Room AC System maintains a habitable environment and ensures the OPERABILITY of components in the control room. A single active failure of a component of the Control Room AC System, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. The Control Room AC System is designed in accordance with Seismic Category I requirements. The Control Room AC System is capable of removing sensible and latent heat loads from the control room, including consideration of equipment heat loads and personnel occupancy requirements to ensure equipment OPERABILITY. INSERT B3.7-17 A The Control Room AC System satisfies Criterion 3 of the NRC Policy Statement.

(continued)

Proposed inserts to Bases for TS 3.7.4

INSERT B 3.7-17 A

The ability of the Control Room AC System to maintain the control room temperature during Modes 1, 2, and 3 is implicitly assumed in the analyses of the design basis accidents (e.g., LOCA, main steam line break). Of the events which can occur in Modes 4 or 5, however, only the potential to drain the reactor vessel is postulated to result in significant radioactive releases.

BASES (continued)

LCO

Two independent and redundant subsystems of the Control Room AC System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits.

The Control Room AC System is considered OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both subsystems. These components include the cooling coils, fans, chillers, compressors, ductwork, dampers, and associated instrumentation and controls. The heating coils are not required for Control Room AC System OPERABILITY.

APPLICABILITY

In MODE 1, 2, or 3, the Control Room AC System must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY limits.

In MODES 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the Control Room AC System OPERABLE is not required in MODE 4 or 5, except~~, for the following situations under which significant radioactive releases can be postulated:~~

~~a.~~ ~~D~~during operations with a potential for draining the reactor vessel (OPDRVs)~~;~~.

~~b. During CORE ALTERATIONS; and~~

~~c. During movement of irradiated fuel assemblies in the primary or secondary containment.~~

ACTIONS

A.1

With one control room AC subsystem inoperable, the inoperable control room AC subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE control room AC subsystem is adequate to perform the control room air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of the control room air conditioning

(continued)

Proposed inserts to Bases for TS 3.7.4

INSERT B 3.7-18A

Due to radioactive decay, the Control Room AC System is only required to be OPERABLE during fuel movement involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 7 days) (Ref. 3).

BASES

ACTIONS

A.1 (continued)

function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining subsystem can provide the required protection, and the availability of alternate cooling methods.

B.1 and B.2

If both control room AC subsystems are inoperable, the Control Room AC System may not be capable of performing its intended function. Therefore, the control room area temperature is required to be monitored to ensure that temperature is being maintained low enough that equipment in the control room is not adversely affected. With the control room temperature being maintained within the temperature limit, 7 days is allowed to restore a control room AC subsystem to OPERABLE status. This Completion Time is reasonable considering that the control room temperature is being maintained within limits, the low probability of an event occurring requiring control room isolation, and the availability of alternate cooling methods.

C.1 and C.2

In MODE 1, 2, or 3, if the control room area temperature cannot be maintained less than or equal to 90°F or if the inoperable control room AC subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 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.

and

D.1~~,~~ D.2~~.1, D.2.2, and D.2.3~~

~~The Required Actions of Condition D are modified by a Note indicating that LCO 3.0.3 does not apply.~~

(continued)

BASES

ACTIONS

D.1 and D.2~~.1, D.2.2, and D.2.3~~ (continued)

If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During ~~movement of irradiated fuel assemblies in the primary or secondary containment, during CORE ALTERATIONS, or during~~ OPDRVs, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE control room AC subsystem may be placed immediately in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, ~~CORE ALTERATIONS and movement of irradiated fuel assemblies in the primary and secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable,~~ actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

E.1~~, E.2, and E.3~~

~~The Required Actions of Condition E.1 are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.~~

(continued)

BASES

ACTIONS

E.1~~, E.2, and E.3~~ (continued)

During ~~movement of irradiated fuel assemblies in the primary or secondary containment, during CORE ALTERATIONS, or during~~ OPDRVs if the Required Action and associated Completion Time of Condition B is not met, action must be taken to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, ~~CORE ALTERATIONS and handling of irradiated fuel in the primary and secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable,~~ actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the heat removal capability of the system is sufficient to remove the control room heat load assumed in the safety analysis. The SR consists of a combination of testing and calculation. The 18 month Frequency is appropriate since significant degradation of the Control Room AC System is not expected over this time period.

REFERENCES

1. UFSAR, Section 6.4.

2. UFSAR, Section 9.4.1.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources—Shutdown

BASES

BACKGROUND	A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources—Operating."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources during MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the primary or secondary containment ensures that:

a. The unit can be ~~maintained~~ in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

Insert B 3.8-35A

In general, when the unit is shut down the Technical Specifications (TS) requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or loss of all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs), which are analyzed in MODES 1, 2, and 3, have no specific analyses in MODES 4 and 5. Worst case bounding events are deemed not credible in MODES 4 and 5 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence significantly reduced or eliminated, and minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCOs for required systems.

During MODES 1, 2, and 3, various deviations from the analysis assumptions and design requirements are allowed within the ACTIONS. This allowance is in recognition that

(continued)

Proposed inserts to Bases for TS 3.8.2

<u>INSERT B 3.8-35A</u>

involving recently irradiated fuel. Due to radioactive decay, AC electrical power is only required to mitigate fuel handling accidents involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 7 days).

<u>INSERT B 3.8-37A</u>

involving recently irradiated fuel

<u>INSERT B 3.8-38A</u>

involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 7 days)

BASES

LCO (continued)

support, assuming a loss of the offsite circuit. Similarly, when the high pressure core spray (HPCS) is required to be OPERABLE, a separate offsite circuit to the Division 3 Class 1E onsite electrical power distribution subsystem, or an OPERABLE Division 3 DG, ensure an additional source of power for the HPCS. This additional source for Division 3 is not necessarily required to be connected to be OPERABLE. Either the circuit required by LCO Item a, or a circuit required to meet LCO Item c may be connected, with the second source available for connection. Together, OPERABILITY of the required offsite circuit(s) and DG(s) ensures the availability of sufficient AC sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents, reactor vessel draindown).

Insert B 3.8-37A

The qualified offsite circuit(s) must be capable of maintaining rated frequency and voltage while connected to their respective ESF bus(es), and accepting required loads during an accident. Qualified offsite circuits are those that are described in the UFSAR and are part of the licensing basis for the plant. The offsite circuit consists of incoming breakers and disconnects to the ESF transformers and the respective circuit path including feeder breakers to all 4.16 kV ESF buses required by LCO 3.8.8.

The required DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage, and accepting required loads. This sequence must be accomplished within 10 seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as: DG in standby with the engine hot and DG in standby with the engine at ambient conditions. Additional DG capabilities must be demonstrated to meet required Surveillances, e.g., capability of the DG to revert to standby status on an ECCS signal while operating in parallel test mode.

Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY. In addition, proper load sequence operation is

(continued)

BASES

LCO (continued)

an integral part of offsite circuit and DG OPERABILITY since its inoperability impacts the ability to start and maintain energized loads required OPERABLE by LCO 3.8.8.

It is acceptable for divisions to be cross tied during shutdown conditions, permitting a single offsite power circuit to supply all required AC electrical power distribution subsystems.

As described in Applicable Safety Analyses, in the event of an accident during shutdown, the TS are designed to maintain the plant in a condition such that, even with a single failure, the plant will not be in immediate difficulty.

APPLICABILITY

The AC sources required to be OPERABLE in MODES 4 and 5 and during movement of irradiated fuel assemblies in the primary or secondary containment provide assurance that:

recently

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel handling accident are available;

Insert B 3.8-38A

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.1.

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

recently

(continued)

BASES

ACTIONS (continued)

A.1

An offsite circuit is considered inoperable if it is not available to one required ESF division. If two or more ESF 4.16 kV buses are required per LCO 3.8.8, division(s) with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, recently irradiated fuel movement, and operations with a potential for draining the reactor vessel. By the allowance of the option to declare required features inoperable with no offsite power available, appropriate restrictions can be implemented in accordance with the affected required feature(s) LCOs' ACTIONS.

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4

With the offsite circuit not available to all required divisions, the option still exists to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend CORE ALTERATIONS, movement of recently irradiated fuel assemblies in the primary and secondary containment, and activities that could potentially result in inadvertent draining of the reactor vessel.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize probability of the occurrence of postulated events. It is further required to initiate action immediately to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient power.

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources—Shutdown

BASES

BACKGROUND

A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources—Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 4 and 5 and during movement of irradiated fuel assemblies in the primary or secondary containment ensures that:

recently

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

INSERT B3.8-60A

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

One DC electrical power subsystem consisting of one battery, one battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the division, associated with Division

(continued)

Proposed inserts to Bases for TS 3.8.5

INSERT B 3.8-60A

involving recently irradiated fuel. Due to radioactive decay, DC electrical power is only required to mitigate fuel handling accidents involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 7 days).

INSERT B 3.8-61A

involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 7 days).

BASES

LCO (continued)

1 or 2 onsite Class 1E DC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems - Shutdown" is required to be OPERABLE. Similarly, when the High Pressure Core Spray (HPCS) System is required to be OPERABLE, the Division 3 DC electrical power subsystem associated with the Division 3 onsite Class 1E DC electrical power distribution subsystem required to be OPERABLE by LCO 3.8.8 is required to be OPERABLE. In addition to the preceding subsystems required to be OPERABLE, a Class 1E battery or battery charger and the associated control equipment and interconnecting cabling capable of supplying power to the remaining Division 1 or 2 onsite Class 1E DC electrical power distribution subsystem(s), when portions of both Division 1 and 2 DC electrical power distribution subsystem are required to be OPERABLE by LCO 3.8.8. This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents and inadvertent reactor vessel draindown).

involving recently irradiated fuel

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 4 and 5 and during movement of irradiated fuel assemblies in the primary or secondary containment provide assurance that:

recently

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

b. Required features needed to mitigate a fuel handling accident are available;

Insert B3.8-61A

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.4.

(continued)

BASES (continued)

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

A.1

Condition A represents one division with a loss of ability to completely respond long term to an event, and a potential loss of ability to remain energized during normal operation. Since eventual failure of the battery to maintain the required battery cell parameters is highly probable, it is imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected division. The additional time provided by the Completion Time is consistent with the capability of the battery to maintain its short term capability to respond to a design basis event.

A Note is added to take exception to the allowance of LCO 3.0.4 to enter MODES or other specified conditions in the Applicability. Even though Condition A Required Actions do not in themselves require a plant shutdown, or require exiting the MODES or other specified conditions in the Applicability, the condition of the DC system is not such that extended operation is expected. Therefore, the Note would require restoration of an inoperable battery charger to OPERABLE status prior to starting up or commencing recently irradiated fuel movement. This exception is not intended to preclude the allowance of LCO 3.0.4 to always enter MODES or other specified conditions in the Applicability as a result of a plant shutdown.

B.1

If the battery cell parameters cannot be maintained within the Category A limits, the short term capability of the battery is also degraded and the battery must be declared inoperable.

(continued)

BASES

ACTIONS (continued)

C.1, C.2.1, C.2.2, C.2.3, and C.2.4

If more than one DC distribution subsystem is required according to LCO 3.8.8, the DC subsystems remaining OPERABLE with one or more DC power sources inoperable for reasons other than an inoperable battery charger may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for draining the reactor vessel. By allowing the option to declare required features inoperable with associated DC power source(s) inoperable, appropriate restrictions are implemented in accordance with the affected system LCOs' ACTIONS. In many instances this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of recently irradiated fuel assemblies, and any activities that could result in inadvertent draining of the reactor vessel).

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystems and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystems should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient power.

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 requires performance of all Surveillances required by SR 3.8.4.1 through SR 3.8.4.8. Therefore, see the corresponding Bases for LCO 3.8.4 for a discussion of each SR.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DC sources from being discharged below their capability to provide the required

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Distribution Systems—Shutdown

BASES

BACKGROUND

A description of the AC and DC electrical power distribution systems is provided in the Bases for LCO 3.8.7, "Distribution Systems—Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC and DC electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the AC and DC electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum AC and DC electrical power sources and associated power distribution subsystems during MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the primary or secondary containment ensures that:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident. INSERT B 3.8-80A

The AC and DC electrical power distribution systems satisfy Criterion 3 of the NRC Policy Statement.

(continued)

Proposed inserts to Bases for TS 3.8.8

INSERT B 3.8-80A

involving recently irradiated fuel. Due to radioactive decay, AC and DC electrical power is only required to mitigate fuel handling accidents involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 7 days).

INSERT B 3.8-81A

involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 7 days).

BASES (continued)

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of Technical Specifications' required systems, equipment, and components—both specifically addressed by their own LCOs, and implicitly required by the definition of OPERABILITY.

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the plant in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving recently irradiated fuel, ~~and inadvertent reactor vessel draindown~~).

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the primary or secondary containment provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel handling accident are available;

INSERT B 3.8-81A

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown or refueling condition.

The AC and DC electrical power distribution subsystem requirements for MODES 1, 2, and 3 are covered in LCO 3.8.7.

(continued)

BASES (continued)

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

<u>A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5</u> recently irradiated

Although redundant required features may require redundant divisions of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem division may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for draining the reactor vessel. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystem LCO's Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made recently (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies in the primary and secondary containment and any activities that could result in inadvertent draining of the reactor vessel).

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the plant safety systems.

Notwithstanding performance of the above conservative Required Actions, a required residual heat removal—shutdown cooling (RHR-SDC) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.4 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR-SDC ACTIONS

(continued)