Entergy

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Randy K. Edington
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July 20, 2001

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

Subject: River Bend Station
Docket No. 50-548
License No. NPF-47
License Amendment Request (LAR) 2000-24 Rev. 1, Operational Conditions For Handling Irradiated Fuel in the Primary Containment.

File Nos.: G9.5, G9.42

References: Letter from R. K. Edington to USNRC, "License Amendment Request (LAR) 2000-24, Operational Conditions For Handling Irradiated Fuel in the Primary Containment," dated January 24, 2001.

RBEXEC-01-030
RBF1-01-0138
RBG-45757

Gentlemen:

By letter dated January 24, 2001 (Reference 1), Entergy Operations, Inc. applied for amendment of the River Bend Station (RBS) Operating License, No. NPF-47. This letter revises that request in response to discussions held with the NRC staff at a meeting on May 9, 2001. The original attachments supporting this request have been included in their entirety with changes denoted by revision bars in the right hand margin. The scope of the Technical Specification changes proposed by Reference 1 has not been changed.

In this revision, Entergy makes commitments to compensatory actions as delineated in TSTF-51 in lieu of relying on the maintenance rule, 10 CFR 50.65(a)(4), as a basis for managing the impact of an open containment configuration. The revised attachment also includes supplemental information concerning the current license basis fuel handling accident analysis. In addition, a set of marked-up drawings is enclosed as requested to aid in your review.

A001

The proposed changes were previously evaluated in accordance with 10CFR50.91(a)(1) using criteria in 10CFR50.92(c). The conclusion that these changes involve no significant hazards considerations remain valid for this revision. The bases for these determinations are included in Attachment 2 to this submittal. Attachment 3 includes marked-up TS pages indicating the specific proposed changes and Attachment 4 includes marked-up pages of the Bases for your information.

Entergy requests that the change be approved for the upcoming refueling outage (RF10) currently scheduled for Fall, 2001. Entergy plans to use this amendment to open the containment hatch for transport of materials and equipment during refueling outages. Entergy requests that the effective date for this TS change to be within 60 days of approval. Although this request is neither exigent nor emergent, your prompt review is requested.

Pursuant to 28 U.S.C.A. Section 1746, I declare under penalty of perjury that the foregoing is true and correct.

Executed on July 20, 2001.

Very truly yours,

Randall K. Edge

RKE/RJK/RWB
Attachments (4)
Enclosures (3)

cc: U. S. Nuclear Regulatory Commission
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Attachment 1

Commitment Identification Form

Commitment	One-Time Action	Continuing Compliance
During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) will be assessed, with respect to filtration and monitoring of releases from the fuel. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay.		X
A single normal or contingency method to promptly close primary or secondary containment penetrations will be established. Such prompt methods need not completely block the penetration or be capable of resisting pressure. Contingency plans for prompt closure of openings will include the following: –Equipment and tools needed to facilitate closure will be staged, –Personnel responsible for closure will be knowledgeable and trained in the procedures for establishing building integrity, –The closure response team will be accompanied by a Radiation Protection (RP) technician for radiation protection monitoring, –Hoses and cables routed through openings will employ a means to allow rapid, safe disconnect and removal, and –One door in each airlock will be capable of expeditious closure		X

ATTACHMENT 2

TO

LETTER NO. RBG-45757

PROPOSED TECHNICAL SPECIFICATION

AND

RESPECTIVE SAFETY ANALYSES

IN THE MATTER OF AMENDING

LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

DESCRIPTION OF PROPOSED CHANGES

Attachment 3 provides marked-up Technical Specification (TS) and Operating License (OL) pages to indicate the proposed changes to the following:

TS Table 3.3.7.1-1, CRFA System Instrumentation
TS 3.6.1.10, Primary Containment - Shutdown
TS 3.7.2, CRFA System
TS 3.7.3, Control Room AC System
TS 5.5.2, Primary Coolant Sources Outside Containment
OL Condition 2.C.(17).

The primary focus of the request concerns the APPLICABILITY requirements for selected specifications associated with CORE ALTERATIONS and with the handling of irradiated fuel in the primary containment and the fuel building.

LCO 3.6.1.10 requires the primary containment to be OPERABLE during CORE ALTERATIONS, during operations with the potential for draining the reactor vessel (OPDRVs), and during movement of irradiated fuel assemblies in the primary containment. Entergy proposes to revise the OPERABILITY requirements of TS 3.6.1.10 to require the primary containment to be OPERABLE only during OPDRVs and movement of "recently" irradiated fuel assemblies in the primary containment. The requirement for primary containment OPERABILITY during CORE ALTERATIONS is deleted. In addition, OPERABILITY will no longer be required during the handling of irradiated fuel that has undergone a sufficient period of natural radioactive decay. The period of sufficient radioactive decay will be defined in the TS Bases. Entergy also proposes to delete OL condition 2.C.(17). This condition is no longer needed for opened air locks or any other primary containment openings because 10CFR50.65(a)(4) now requires licensees to assess and manage the risk associated with SSCs being removed from service during normal shutdown operations.

Proposed changes to TS Table 3.3.7.1-1, TS 3.7.2, and TS 3.7.3 also delete the OPERABILITY requirements during CORE ALTERATIONS. In addition, editorial corrections are proposed regarding the description of the area where irradiated fuel is handled (i.e., the fuel building rather than secondary containment) to be consistent with Amendment 113 and similar TSs such as TS 3.8.2, 3.8.5, and 3.8.8.

The proposed change to TS 5.5.2 deletes the Standby Gas Treatment (SGT) system from the list of systems included in the "Primary Coolant Sources Outside Containment" leakage control program.

Proposed changes to the following Bases pages have also been included as Attachment 4 for information. Entergy will incorporate the appropriate changes to the Bases in accordance with TS 5.5.11, the TS Bases Control Program.

B 3.3.7.1, CRFA System Instrumentation
B 3.6.1.10, Primary Containment – Shutdown
B 3.7.2, CRFA System
B 3.7.3, Control Room AC System

BACKGROUND

The primary containment is not required to be OPERABLE during shutdown conditions (MODES 4 and 5) except during situations for which significant releases of radioactive material can be postulated. The primary containment is currently required by TS 3.6.1.10 to be OPERABLE during CORE ALTERATIONS, during the movement of irradiated fuel assemblies inside the primary containment, and during operations with the potential for draining the reactor vessel (OPDRVs). The safety function of the primary containment during these situations is to contain the fission products that could occur from an accident inside the primary containment such that doses at the site boundary remain within limits.

Entergy desires to revise TS 3.6.1.10 to allow the primary containment requirements to be relaxed during CORE ALTERATIONS and during the handling of irradiated fuel that has undergone a sufficient period of natural radioactive decay. This allowance has already been approved for the containment air locks (TS 3.6.1.2, Amendment 85), the fuel building (TS 3.6.4.5, Amendment 113), and the fuel building ventilation system (TS 3.6.4.7, Amendment 113). Entergy plans to use this allowance during the next refueling outage to open the containment equipment hatch to transport equipment and materials from the outside area into the containment. This transport path would also require the removal of the shield building equipment hatch. However, no TS changes are needed to open the shield building equipment hatch.

The NRC has generally determined that the Fuel Handling Accident (FHA) dose consequences are acceptable when conservatively calculated doses remain less than 25% of 10 CFR 100 guidelines, as reflected in the Standard Review Plan, NUREG 0800. For BWRs, including River Bend Station (RBS), these dose analyses have typically concluded that the handling of irradiated fuel assemblies is acceptable to begin 24 hours after entry into a plant shutdown. These analyses credited certain ESF mitigating systems such as containment, ventilation and filtration systems. The acceptance criteria of SRP 15.7.4, paragraph 11.5, allow a containment to be open to the environment during fuel handling if the capability for prompt detection of radiation and automatic isolation is provided and reflected in the analysis of the FHA radiological consequences.

An alternative to the SRP approach of crediting mitigating systems is to take credit for a longer radioactive decay period of irradiated fuel and the water level that covers the fuel bundles as a natural defense in depth method. In this manner, OPERABILITY requirements for certain mitigating systems could be relaxed after a specified fuel decay period. RBS previously requested such changes by License Amendment Request (LAR) 95-04 (Reference 1).

LAR 95-04 provided a revised offsite dose calculation showing that the consequences of a FHA would remain less than 25% of the 10 CFR 100 guidelines once the fuel had undergone radioactive decay for 11 days. The analysis took no credit for the primary containment, fuel building, or installed ventilation and filtration systems.

Following the RBS submittal and similar submittals by other BWR6 licensees, a collaboration of the four BWR6 plants met at the request of the NRC Staff to develop a joint approach for the proposed changes. The Staff also expressed a desire for the issue to be resolved more generically than just for the BWR6 plants. The Perry Nuclear Power Station was selected as a pilot for this effort. However, the Staff deferred taking complete action on the requests because ongoing activities that related to the shutdown rule was anticipated to provide staff guidance on TS requirements during plant shutdown.

In the interim, the NRC Staff partially approved Entergy's request by issuing Amendment 85 (Reference 2) to the RBS operating license. This amendment only approved the opening of the containment personnel air locks (TS 3.6.1.2). The opening of the air locks was also conditioned upon certain administrative controls required by new license condition 2.C. (17).

The concerns over the initial proposals have since been resolved generically with the NRC staff through the NEI Technical Specification Task Force (TSTF) and the proposed changes were issued as Revision 2 to TSTF traveler 51. The Perry Nuclear Power Station received approval for the remaining proposed changes on March 11, 1999 by issuance of Amendment 102 to NPF-58.

The proposed changes for the RBS TS in this request do not completely implement TSTF-51. This request focuses primarily on relaxations to the primary containment APPLICABILITY requirements during shutdown. Changes to the fuel building and fuel building ventilation system requirements were previously approved for RBS by issuance of Amendment 113. Proposed changes included in TSTF-51 affecting the Control Room Fresh Air system, the Control Room Air Conditioning system, and associated electrical support systems are only partially included in this request because some of the TSTF changes have not been evaluated by the current Fuel Handling Accident (FHA) analysis of record. We do believe that changes to the analysis would support these additional TS changes and may pursue these changes at a later date.

BASIS FOR PROPOSED CHANGE

Changes to TS 3.6.1.10, "Primary Containment – Shutdown" and License Condition 2.C (17).

The safety design basis for the primary containment during shutdown conditions is that it contain the fission products from a FHA that occurs inside the primary containment such that doses at the site boundary remain within limits. To maintain the containment as a leak tight barrier during certain shutdown activities, TS 3.6.1.10 requires the containment to be OPERABLE during movement of irradiated fuel assemblies in the primary containment, during CORE ALTERATIONS, and during operations with the potential for draining the reactor vessel (OPDRVs). In general, for the primary containment to be considered OPERABLE, all penetrations required to be closed during accident conditions must be closed, the primary containment air locks must be OPERABLE and all equipment hatches must be closed.

The Improved Standard Technical Specifications for the BWR6s, NUREG 1434, does not include a TS 3.6.1.10 section or requirements for primary containment during shutdown. These TS requirements, however, are applicable to RBS and the Perry Nuclear Power Plant due to containment and secondary containment designs. Although TSTF-51 Rev. 2, does not include generic proposed changes to the current requirements of TS 3.6.1.10, the Entergy proposed changes for the RBS TS are consistent with the TSTF-51 concepts and have already been approved for the Perry Nuclear Power Plant. There are two changes proposed for TS 3.6.1.10. First, Entergy proposes to delete the OPERABILITY requirements during CORE ATERATIONS and secondly, to change the OPERABILITY requirements to allow the primary containment to be inoperable when handling fuel that has not been recently irradiated.

The deletion of the TS requirements during CORE ALTERATIONS is based on the premise that the only accident postulated to occur during CORE ALTERATIONS that results in fuel damage and radioactive release is the FHA. The other accidents postulated to occur are: inadvertent criticality (due to a control rod removal error or continuous control rod withdrawal error during refueling) and the inadvertent loading of and subsequent operation with a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage during the shutdown conditions. Therefore, the primary containment is only needed during shutdown to mitigate the FHA event. The proposed changes do not impact the TS requirements for systems needed to prevent or mitigate CORE ALTERATION events other than the FHA. They also do not affect the requirements needed to mitigate potential vessel draindown events or systems needed for decay heat removal.

The proposal to relax the primary containment requirements during handling of fuel that has not been recently irradiated is based on a radioactive decay period that takes advantage of the reduced radionuclide inventory available for release in the event of a FHA. This portion of TSTF-51 is only applicable to those licensees who have demonstrated by analysis that after a sufficient radioactive decay period has occurred, off-site doses resulting from the FHA remain below the Standard Review Plan (SRP) guideline limits (well within 10 CFR 100). In addition, TSTF-51 states that licensees adding the term "recently" must make certain commitments which are consistent with draft NUMARC 93-01, Revision 3. The following address compliance with these two conditions of the TSTF.

FHA Analysis

The Entergy proposed TS changes are based on the current FHA analysis of record approved by Amendment 110. No new or revised analysis is needed for this request.

Entergy submitted a FHA analysis by Reference 1, which postulated the drop of an irradiated fuel assembly onto the reactor core and assumed that the primary containment equipment hatch was open to the environment. The analysis demonstrated that for the worst case bundle drop, the regulatory dose guidelines of SRP 15.7.4 were satisfied after a decay period of 11 days or more without credit for ESF mitigating systems. This analysis was used as a basis for the issuance of Amendment 85 (Reference 2), which relaxed the TS OPERABILITY requirements containment air locks after a fuel assembly radioactive decay period of 11 days.

Entergy later proposed additional changes to the FHA analysis, which were approved by Amendment 110. Among other changes, the revised analysis accounted for power uprate core design and credited the control room filtration system. The analysis demonstrated that for the FHA inside containment with the equipment hatch open, the radiological dose consequences remained well below the dose criteria of 10 CFR 100 with regard to offsite doses and GDC 19 of 10 CFR 50, Appendix A with regard to control room personnel doses. This analysis also formed the basis for Amendment 113 (Reference 4) for relaxing the fuel building and fuel building ventilation system requirements after a fuel assembly radioactive decay period of 11 days.

The results of the Entergy analysis and the confirmatory NRC analysis justifying issuance of Amendment 110 are presented in the following table. These results are included in the NRC SER for Amendment 110 and are provided here for continuity purposes.

AMENDMENT 110 CALCULATED RADIOLOGICAL CONSEQUENCES **(in Rem)**		
Exclusion Area Boundary		
	Dose	SRP Limits
Whole Body	0.2	6
Thyroid	67	75
Low Population Zone		
Whole Body	0.1	6
Thyroid	8.8	75
Control Room		
Whole Body	0.1	5
Thyroid	3.6	30

The River Bend Station atmospheric dispersion factors (X/Qs) used in the current analysis for the FHA inside containment were calculated as a diffuse release. The distances used to determine the X/Qs were calculated from the receptor to the closest point on the on the containment perimeter. The calculations do not consider any specific release points, such as the containment equipment hatch. The shortest distance from the containment structure to the main control room intake is 39 m as measured from the closest point on outermost reactor building wall. The shortest distance from the reactor building outermost wall to the remote control room intake is approximately 95 m. Since the containment hatch is approximately 50 m from the normal control room intake and 120 m from the remote intake, the X/Qs for the Control Room used in the current FHA analysis are conservative and applicable to the potential release location through the open equipment hatch.

The locations for the main control room air intakes and other points of interest are marked on enclosed drawing number EM-001A. The locations of the intakes were derived from drawings EB-039D and EB-029E which are also enclosed with the areas of interest highlighted. The approximate coordinates for the key locations have been added for convenience. The release locations are not marked since release location is assumed to be at the reactor building perimeter.

Additional details of the FHA analysis including input assumptions and a discussion of margins and conservatisms are included in Reference 3.

Administrative Controls

The industry, through the Nuclear Energy Institute (NEI) and its predecessor, the Nuclear Management and Resources Council (NUMARC), has developed guidance to assess and manage the increase in risk that may result during outage activities. This guidance was issued as NUMARC 91-06 "Guidelines for Industry Actions to Assess Shutdown Management" and NUMARC 93-01"Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants". Section 4.5 of NUMARC 91-06 discusses the need to ensure that containment closure can be achieved in sufficient time to prevent potential fission product release. RBS has administrative controls in place to meet the recommendations of NUMARC 91-06 Section 4.5. Sections 11.3.5 and 11.3.6 of NUMARC 93-01 Rev. 3 also address risk assessments for shutdown conditions and include additional guidance for managing an open containment.

TSTF-51, "Reviewer's Note" states that licensees adding the term "recently" must make the following commitments which are consistent with draft NUMARC 93-01, Revision 3, Section 11.2.6 "Safety Assessment for Removal of Equipment from Service During Shutdown Conditions", subheading "Containment-Primary (PWR)/Secondary(BWR)":

> –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 are 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."

NUMARC 93-01 Revision 3 has been issued and section 11 has been endorsed by Regulatory Guide 1.182. Sections 11.3.5 and 11.3.6 of Rev. 3 address the scope and methods of assessing shutdown conditions. Section 11.3.6.5 in particular provides guidelines for assessing containment systems to be removed from service including the capability to achieve containment closure in sufficient time to mitigate potential fission product release. The guidelines are the same as those in Section 11.2.6 of the draft as discussed in the TSTF. Entergy will implement these guidelines following approval of the proposed amendment to the Technical Specifications.

Contingency plans for closure of openings will include the following:

–Equipment and tools needed to facilitate closure will be staged,
–Personnel responsible for closure will be knowledgeable and trained in the procedures for establishing building integrity,
–The closure response team will be accompanied by a Radiation Protection (RP) technician for radiation protection monitoring,
–Hoses and cables routed through openings will employ a means to allow rapid, safe disconnect and removal, and
–One door in each airlock will be capable of expeditious closure

The RBS containment vessel equipment hatch opening is easily accessible for prompt closure. The hatch is located at el. 103 ft. 9 in. (just above ground level) to provide access for equipment being moved from outside the reactor building into the primary containment. The hatch cover is bolted on the outside of the vessel and is provided with a hoist with two-point suspension and a sliding rail for storage (see USAR Figures 3.8-1 and 3.8-6). The shield building equipment hatch must also be opened in order to transport equipment and materials from the yard area into the primary containment. The shield building access opening is located in-line with the containment hatch and is enclosed by two rectangular steel panels. The panels are hinge mounted on the outside of the shield building wall using a swing-open arrangement. The door panels are furnished with perimeter gaskets at head, jambs, and sill to provide leaktightness. Closing either of these openings (applying a normal or contingency method) could be used as a means of isolating the containment from the outside environment, further reducing any consequences of a FHA. The closing of these openings is not credited in the FHA analysis and is not required to meet the dose release limits of the SRP.

In addition, the fuel pool and reactor vessel water levels are adequately controlled by TS 3.7.6, 3.9.6, and 3.9.7 to ensure the assumptions of the FHA analysis are maintained. These TS requirements remain unchanged, so the water levels will continue to be maintained during the handling of any irradiated fuel bundles, regardless of the radiation decay period.

In summary, Entergy concludes that the requirements for primary containment may be relaxed during CORE ALTERATIONS and when handling irradiated fuel in the primary containment that has undergone a natural decay period of 11 days and that license condition 2.C (17) may be deleted based on the following.

- The only accident postulated to occur during CORE ALTERATIONS that results in fuel damage and radioactive release is the FHA. The other accidents postulated to occur, such as inadvertent criticality or the inadvertent loading of and subsequent operation with a fuel assembly in an improper location, are not postulated to result in fuel cladding integrity damage during the shutdown

conditions. The proposed changes do not affect the requirements that protect or mitigate a reactor vessel draindown event.

- The current FHA analysis of record demonstrates that containment closure is not required to mitigate the consequences of a FHA once the fuel has undergone a natural decay period of 11 days (i.e., the fuel has not been part of a critical reactor core within the last 11 days). Dose consequences from the analysis remain within the guidelines of the SRP.

- Entergy will implement administrative controls in accordance with the guidelines of NUMARC 93-01 Rev. 3, Section 11.3.6.5. regarding the availability of ventilation and radiation monitor systems and contingency plans for prompt closure of containment openings.

Proposed changes to TS Table 3.3.7.1-1, TS 3.7.2, and TS 3.7.3

TS Table 3.3.7.1-1, TS 3.7.2, and TS 3.7.3 require the Control Room (CR) Fresh Air system, the CR local intake radiation monitors, and the CR air conditioning system to be OPERABLE during CORE ALTERATIONS and during the movement of irradiated fuel assemblies in the primary or secondary containment. Changes are proposed to these sections to 1) delete the requirement for OPERABILITY during CORE ALTERATIONS and 2) revise the term "secondary containment" to "fuel building". The requirements associated with the handling of irradiated fuel assemblies are not being changed.

The deletion of the OPERABILITY requirements during CORE ALTERATIONS is consistent with TSTF-51, Rev. 2 and is justified on the same basis as the similar proposed changes to TS 3.6.1.10. (i.e., the only accident postulated to occur during CORE ALTERATIONS that results in fuel damage and radioactive release is the FHA). The changes to the term "secondary containment" are proposed because the fuel building is the only building outside of the primary containment in which fuel is handled and per Amendment 113, the fuel building is no longer considered part of the secondary containment. This change makes these TSs consistent with similar TS such as 3.8.2, 3.8.5, and 3.8.8, which use the term "fuel building", rather than "secondary containment".

Proposed Change to TS 5.5.2, "Primary Coolant Sources Outside Containment"

The proposed change to TS 5.5.2 deletes the SGT system from the list of systems included in the "Primary Coolant Sources Outside Containment" leakage control program. The program is intended to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practical. This leakage control program typically includes ESF systems that circulate contaminated fluids outside the primary containment.

The function of the SGT system is to ensure that airborne radioactive materials that leak from the primary containment or from ESF systems following a design basis accident are filtered and adsorbed prior to exhausting to the environment. The SGT system is designed to maintain an annulus negative pressure of at least 0.5 inches W.G. during a DBA and the auxiliary building at 0.25 inches negative pressure during a DBA to ensure that the release of radioactive materials is limited as much as practical to the SGT system filtered leakage path. The ability to achieve these requirements is in part dependent on the leak tightness of the annulus and the auxiliary building. SR 3.6.4.1.1 and SR 3.6.4.1.4 verify the leak tightness of these boundaries.

The discharge section of the SGT system ductwork does not contain "highly radioactive fluids" because the airflow is downstream of the filters where iodine and contaminants have been removed to at least 99% efficiency. Although the inlet ducts may contain highly radioactive fluid (airborne radioactive contaminants), the inlet ducts are contained within the secondary containment boundary and are under negative pressure up to the filter train. There are no unfiltered pathways to the suction of the SGT fans. Therefore, the inclusion of the system in the TS 5.5.2 leakage control program is unnecessary and should be deleted.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Entergy Operations, Inc. is proposing that the River Bend Station Operating License be amended to relax the containment OPERABILITY requirements when handling non-recently irradiated fuel and during CORE ALTERATIONS. These changes are consistent with Technical Specification Task Force (TSTF) traveler 51, Rev. 2 and are justified based on the current fuel handling accident analysis of record. Entergy is also proposing to relax the Control Room filtration, ventilation inlet radiation detection and air conditioning system requirements during CORE ALTERATIONS in accordance with TSTF-51. In addition, a change is proposed to delete the Standby Gas Treatment (SGT) system from the "Primary Coolant Sources Outside Containment" leakage control program.

An evaluation of the proposed changes have been performed in accordance with 10CFR50.91(a)(1) regarding no significant hazards considerations using the standards in 10CFR50.92(c). A discussion of these standards as they relate to this amendment request follows:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

The relaxation of TS OPERABILITY requirements for containment and control room ventilation systems during specific shutdown conditions do not affect the probability of any accident previously evaluated and do not alter current accident analyses consequences. During plant shutdown, these systems and structures are accident mitigating features for the postulated Fuel Handling Accident (FHA) and are not considered the initiator to any previously analyzed accident. They need not be required during CORE ALTERATIONS because the only accident postulated to result in significant fuel damage and radiation release during shutdown conditions is the FHA. The control room filtration, inlet radiation detection, and the air conditioning systems will continue to be required during the handling of any irradiated fuel assembly and during operations with the potential for draining the reactor vessel (OPDRVs). The containment will only be required during OPDRVs and when moving recently irradiated fuel assemblies. The current FHA analysis of record (approved by Amendment 110) assumes the containment is open after the irradiated fuel has undergone a sufficient decay period (i.e., has not been part of a critical reactor core within the previous 11 days). The analysis demonstrates that the offsite doses remain well within the Standard Review Plan Guidelines (less than 25% of the 10CFR100 limits) and the control room doses remain less than the criteria of 10 CFR 50, Appendix A, General Design Criterion 19.

The proposed changes regarding the removal of the SGT system from the "Primary Coolant Sources Outside Containment" leakage control program does not affect the reliability or filtration efficiency of the SGT system. Current TS surveillances test filtration efficiency and secondary containment in-leakage. There are no unfiltered pathways to the suction of the fans that require leakage testing.

Therefore, these changes do <u>not</u> involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not involve any design changes or any new modes of system operation. The proposed TS changes allow certain functions to be inoperable during CORE ALTERATIONS and during the handling of irradiated fuel that has undergone a sufficient radiation decay period. However, these out-

of-service configurations are consistent with current design basis analyses. The removal of the SGT system from the "Primary Coolant Sources Outside Containment" leakage control program does not affect reliability or efficiency of the filtration system or otherwise affect the ability of the system to perform its safety function.

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

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

The proposed changes do not reduce the margin of safety, as defined by SRP 15.7.4 Rev 1. The only accident postulated to occur during shutdown that results in significant fuel damage and subsequent radiation release is the FHA. The offsite and control room doses due to a FHA with an open containment have previously been evaluated with conservative assumptions and that analysis is not affected by the proposed changes. The analysis demonstrates that due to radioactive decay following reactor shutdown, the primary containment function is only required when handling recently irradiated fuel.

The removal of the SGT system from the "Primary Coolant Sources Outside Containment" leakage control program does not affect reliability or efficiency of the filtration system or otherwise affect the ability of the system to perform its safety function.

Therefore, this change does not involve a significant reduction in the margin of safety.

Therefore, based on the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change does not involve a significant hazards consideration.

ENVIRONMENTAL IMPACT EVALUATION

Pursuant to 10CFR51.22(b), an evaluation of the proposed amendment has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10CFR 51.22 (c) (9) of the regulations. The basis for this determination is as follows:

1. The proposed license amendment does not involve a significant hazards consideration as described previously in the evaluation.

2. As discussed in the significant hazards evaluation, this change does not result in a significant change or significant increase in the radiological doses for any Design Basis Accident. The proposed license amendment does not result in a significant change in the types or a significant increase in the amounts of any effluents that may be released off-site.

3. The proposed license amendment does not result in a significant increase to the individual or cumulative occupational radiation exposure. In the event of a FHA, the open containment may help facilitate a more rapid evacuation of the containment, thus reducing potential individual exposure.

References

1. River Bend Station Request for License Amendment (LAR 95-04) dated August 17, 1995.
2. USNRC Issuance of Amendment 85 (TAC No. M93306) dated January 11, 1996.
3. USNRC Issuance of Amendment 110 (TAC No. MA7701) dated March 2, 2000.
4. USNRC Issuance of Amendment 113 (TAC No. MA8916) dated September 22, 2000.

ATTACHMENT 3

TO

LETTER NO. RBG-45757

PROPOSED TECHNICAL SPECIFICATION MARK-UPS

LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

(13) Partial Feedwater Heating (Section 15.1, SER)

During power operation, the facility shall not be operated with a feedwater heating capacity which would result in a rated thermal power feedwater temperature less than 326 °F.

(14) Emergency Response Capabilities (Generic Letter 82-33, Supplement 1 to NUREG-0737, Section 7.5.2.4, SER and SSER 3, Section 18, SER, SSER 2 and SSER 3)

EOI shall complete the requirements of NUREG-0737 Supplement #1 as specified in Attachment 5. Attachment 5 is hereby incorporated into this license.

(15) Salem ATWS Events, Generic Letter 83-28 (Section 7.2.2.5, SSER 3)

EOI shall submit responses to and implement the requirements of Generic Letter 83-28 on a schedule which is consistent with that given in its letters dated August 3, 1984 and May 30, 1985.

(16) Merger Related Reports

Entergy Gulf States, Inc. shall inform the Director, NRR:

a. Sixty days prior to a transfer (excluding grants of security interests or liens) from Entergy Gulf States, Inc. to Entergy or any other entity of facilities for the production, transmission or distribution of electric energy having a depreciated book value exceeding one percent (1%) of Entergy Gulf States, Inc.'s consolidated net utility plant, as recorded on Entergy Gulf States, Inc.'s books of account.

b. Of an award of damages in litigation initiated against Entergy Gulf States, Inc. by Cajun Electric Power Cooperative regarding River Bend within 30 days of the award.

(17) Primary containment air lock doors may be open during CORE ALTERATIONS, except when moving recently irradiated fuel, (i.e., fuel that has occupied part of a critical reactor core within the previous 11 days), provided the following conditions exist:

1) One door in each air lock is capable of being closed.

2) Hoses and cables running through the air lock employ a means to allow safe, quick disconnect and are tagged at both ends with specific instructions to expedite removal.

3) There is a minimum of 23 feet of water over the core.

4) The air lock doors are not blocked open to allow expeditious closure.

DELETED.

5) A designated individual is available to expeditiously close the air lock doors.

6) Systems are available to filter and monitor releases from the containment.

D. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "River Bend Station Security Plan," with revisions submitted through May 25, 1995; "River Bend Station Guard Training and Qualification Plan," with revisions submitted through December 3, 1993; and "River Bend Station Contingency Plan," with revisions submitted through August 17, 1990. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein."

E. Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, EOI shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within 24 hours to the NRC Operations Center via the Emergency Notification System with written followup within thirty days in accordance with the procedures described in 10 CFR 50.73(b), (c), and (e).

F. The licensees shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

G. This license is effective as of the date of issuance and shall expire at midnight on August 29, 2025.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Attachments 1-5
2. Appendix A - Technical Specifications (NUREG-1172)
3. Appendix B - Environmental Protection Plan
4. Appendix C - Antitrust Conditions

Date of Issuance: November 20, 1986

Revised: December 16, 1993

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

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	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4 SR 3.3.7.1.5	≥ -47 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 3.3.7.1.4 SR 3.3.7.1.5	≤ 1.88 psid
3. Control Room Local Intake Ventilation Radiation Monitors	1,2,3 (a),(b)	1	D	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.5	≤ 0.97 x 10^{-5} μCi/cc

(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 or fuel building.

3.6 CONTAINMENT SYSTEMS

3.6.1.10. Primary Containment—Shutdown

LCO 3.6.1.10 Primary containment shall be OPERABLE.

APPLICABILITY: During movement of recently irradiated fuel assemblies in the primary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Primary containment inoperable.	A.1 Suspend movement of recently irradiated fuel assemblies in the primary containment.	Immediately
	~~AND~~	
	~~A.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	AND	
	A.2 Initiate action to suspend OPDRVs.	Immediately

3.7 PLANT SYSTEM

3.7.2 Control Room Fresh Air (CRFA) System

LCO 3.7.2 Two CRFA subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRFA subsystem inoperable.	A.1 Restore CRFA subsystem to OPERABLE status.	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)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the primary ~~or secondary~~ containment, ~~during CORE ALTERATIONS~~, or during OPDRVs. or fuel building	----------NOTE---------- LCO 3.0.3 is not applicable. ----------	
	C.1 Place OPERABLE CRFA subsystem in emergency mode.	Immediately
	OR	
	C.2.1 Suspend movement of irradiated fuel assemblies in the primary ~~and secondary~~ containment~~s~~ and fuel building.	Immediately
	~~AND~~	
	~~C.2.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	AND	
	C.2.~~3~~2 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

(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. or fuel building	E.1 Suspend movement of irradiated fuel assemblies in the primary ~~and secondary~~ containment, and fuel building.	Immediately
	~~AND~~ ~~E.2~~ ~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	AND E.~~3~~2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	Operate each CRFA subsystem for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.2.2	Perform required CRFA filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.2.3	Verify each CRFA subsystem actuates on an actual or simulated initiation signal.	18 months

(continued)

3.7 PLANT SYSTEMS

3.7.3 Control Room Air Conditioning (AC) System

LCO 3.7.3 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, or fuel building.
~~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 ≤ 104°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 movement of irradiated fuel assemblies in the primary ~~or secondary~~ containment ~~during CORE ALTERATIONS~~ or during OPDRVs. or fuel building	----------NOTE---------- LCO 3.0.3 is not applicable. ----------	
	D.1 Place OPERABLE control room AC subsystem in operation.	Immediately
	OR	
	D.2.1 Suspend movement of irradiated fuel assemblies in the primary ~~and secondary~~ containment and fuel building	Immediately
	~~AND~~	
	~~D.2.2 Suspend CORE ALTERATIONS.~~	Immediately
	AND	
	D.2.~~3~~2 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 movement of irradiated fuel assemblies in the primary ~~or secondary~~ containment ~~during CORE ALTERATIONS~~ or during OPDRVs. or fuel building	E.1 Suspend movement of irradiated fuel assemblies in the primary ~~and secondary~~ containment ~~and~~ fuel building.	Immediately
	~~AND~~	
	~~E.2~~ ~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	AND	
	E.~~3~~ 2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify each control room AC subsystem has the capability to remove the assumed heat load.	18 months

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of, or concurrent with, the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include the Low Pressure Core Spray, High Pressure Core Spray, Residual Heat Removal, Reactor Core Isolation Cooling, and process sampling. ~~and Standby Gas Treatment Systems.~~ The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

(continued)

ATTACHMENT 4

TO

LETTER NO. RBG-45757

PROPOSED TECHNICAL SPECIFICATION BASES MARK-UPS

(FOR INFORMATION ONLY)

LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2. Drywell Pressure-High (continued)

Drywell Pressure-High signals are initiated from four pressure transmitters that sense drywell pressure. Four channels of Drywell Pressure-High Function are available (two channels per trip system) and are required to be OPERABLE to ensure that no single instrument failure can preclude CRFA System initiation.

The Drywell Pressure-High Allowable Value was chosen to be the same as the Secondary Containment Isolation Drywell Pressure-High Allowable Value (LCO 3.3.6.2).

The Drywell Pressure-High Function is required to be OPERABLE in MODES 1, 2, and 3 to ensure that control room personnel are protected during a LOCA. In MODES 4 and 5, the Drywell Pressure-High Function is not required since there is insufficient energy in the reactor to pressurize the drywell to the Drywell Pressure-High setpoint.

3. Control Room Local Intake Ventilation Radiation Monitors

The Control Room Local Intake Ventilation Radiation Monitors measure radiation levels exterior to the inlet ducting of the MCR. A high radiation level may pose a threat to MCR personnel; thus, a detector indicating this condition automatically signals initiation of the CRFA System.

The Control Room Local Intake Ventilation Radiation Monitors Function consists of two independent monitors. Two channels of Control Room Local Intake Ventilation Radiation Monitors are available and are required to be OPERABLE to ensure that no single instrument failure can preclude CRFA System initiation. The Allowable Value was selected to ensure protection of the control room personnel.

The Control Room Local Intake Ventilation Radiation Monitors Function is required to be OPERABLE in MODES 1, 2, and 3, and during ~~CORE ALTERATIONS,~~ operations with a potential for draining the reactor vessel (OPDRVs), and movement of irradiated fuel in the primary ~~or secondary~~ containment or fuel building to ensure that control room personnel are protected during a LOCA, fuel handling event, or a vessel draindown event. During MODES 4 and 5, when these specified conditions are not in progress (e.g., ~~CORE ALTERATIONS~~ OPDRVs), the probability of a LOCA or fuel damage is low; thus, the Function is not required.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.10 Primary Containment-Shutdown

BASES

BACKGROUND

The function of the primary containment is to isolate and contain fission products released from the Reactor Primary System following a Design Basis Accident (DBA) and to confine the postulated release of radioactive material to within limits. The primary containment consists of a steel lined, reinforced concrete vessel, which surrounds the Reactor Primary System and provides an essentially leak tight barrier against an uncontrolled release of radioactive material to the environment. Additionally, this structure provides shielding from the fission products that may be present in the primary containment atmosphere following accident conditions.

The isolation devices for the penetrations in the primary containment boundary are a part of the primary containment leak tight barrier. To maintain this leak tight barrier for accidents during shutdown conditions:

a. All penetrations required to be closed during accident conditions are closed by manual valves, blind flanges, or de-activated power operated or automatic valves secured in their closed positions, or the equivalent, except as provided in LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)";

b. Primary containment air locks are OPERABLE, except as provided in LCO 3.6.1.2, "Primary Containment Air Locks"; and

c. All equipment hatches are closed.

This Specification ensures that the performance of the primary containment, in the event of a fuel handling accident (FHA) involving handling recently irradiated fuel, inadvertent criticality, or reactor vessel draindown, provides an acceptable leakage barrier to contain fission products, thereby minimizing offsite doses.

APPLICABLE SAFETY ANALYSES

The safety design basis for the primary containment is that it contain the fission products from a FHA inside the

(continued)

BASES (continued)

APPLICABILITY

Due to radioactive decay, the primary containment is only required to be OPERABLE during fuel handling involving recently irradiated fuel (ie, fuel that has occupied part of a critical reactor core within the previous 11 days)

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 an OPERABLE primary containment in MODE 4 or 5 to ensure a control volume, is only required during situations for which significant releases of radioactive material can be postulated; such as during operations with a potential for draining the reactor vessel (OPDRVs), ~~during CORE ALTERATIONS~~ or during movement of recently irradiated fuel assemblies in the primary containment.

Requirements for ECCS OPERABILITY during MODES 1, 2, and 3 are discussed in the Applicability section of the Bases for LCO 3.5.1.

ACTIONS

A.1, A.2 ~~and A.3~~

In the event that primary containment is inoperable, action is required to immediately suspend activities that represent a potential for releasing radioactive material, thus placing the unit in a Condition that minimizes risk. If applicable, ~~CORE ALTERATIONS~~ and movement of recently irradiated fuel assemblies must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.10.1

This SR verifies that each primary containment penetration that could communicate gaseous fission products to the environment during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive gases outside of the primary containment boundary is within design limits. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Single isolation barriers that meet this criterion are a closed and de-activated power operated or automatic valve, a closed manual valve, a blind flange, or equivalent. This does not preclude the use of two active (ie, power operated and/or automatic) valves in the closed position for a given penetration. This SR does not require any testing or valve manipulation.

(continued)

BASES

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);

~~b. During CORE ALTERATIONS~~; and

b. During movement of irradiated fuel assemblies in the primary ~~or secondary~~ containment, or Fuel Building.

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 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, and 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

(continued)

BASES

ACTIONS

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

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 fuel building ~~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 emergency mode. 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 and fuel building 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.

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.

(continued)

BASES

ACTIONS (continued)

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

During movement of irradiated fuel assemblies in the primary ~~or secondary~~ containment or fuel building ~~during CORE ALTERATIONS,~~ or during 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 and fuel building 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 3.7.2.1

This SR verifies that a subsystem in a standby mode starts on demand from the control room and continues to operate with flow through the HEPA filters and charcoal adsorbers. 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 ≥ 10 continuous hours with the heaters energized to demonstrate the function of the system. Furthermore, the 31 day Frequency is based on the known reliability of the equipment and the two subsystem redundancy available.

(continued)

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.

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

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

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)

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 once per 4 hours 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. These Completion Times are reasonable considering that the control room temperature is being maintained within limits, the low probability of an event occuring 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 ≤ 104°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.

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

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

(continued)

BASES

ACTIONS

D.1, 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 or fuel building ~~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 and fuel building 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~~

During movement of irradiated fuel assemblies in the primary ~~or secondary~~ containment or fuel building ~~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 ~~or secondary~~ containment or fuel building must be suspended immediately. Suspension of these activities shall

(continued)

BASES

ACTIONS

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

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 3.7.3.1

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. USAR, Section 6.4.

2. USAR, Section 9.4.1.

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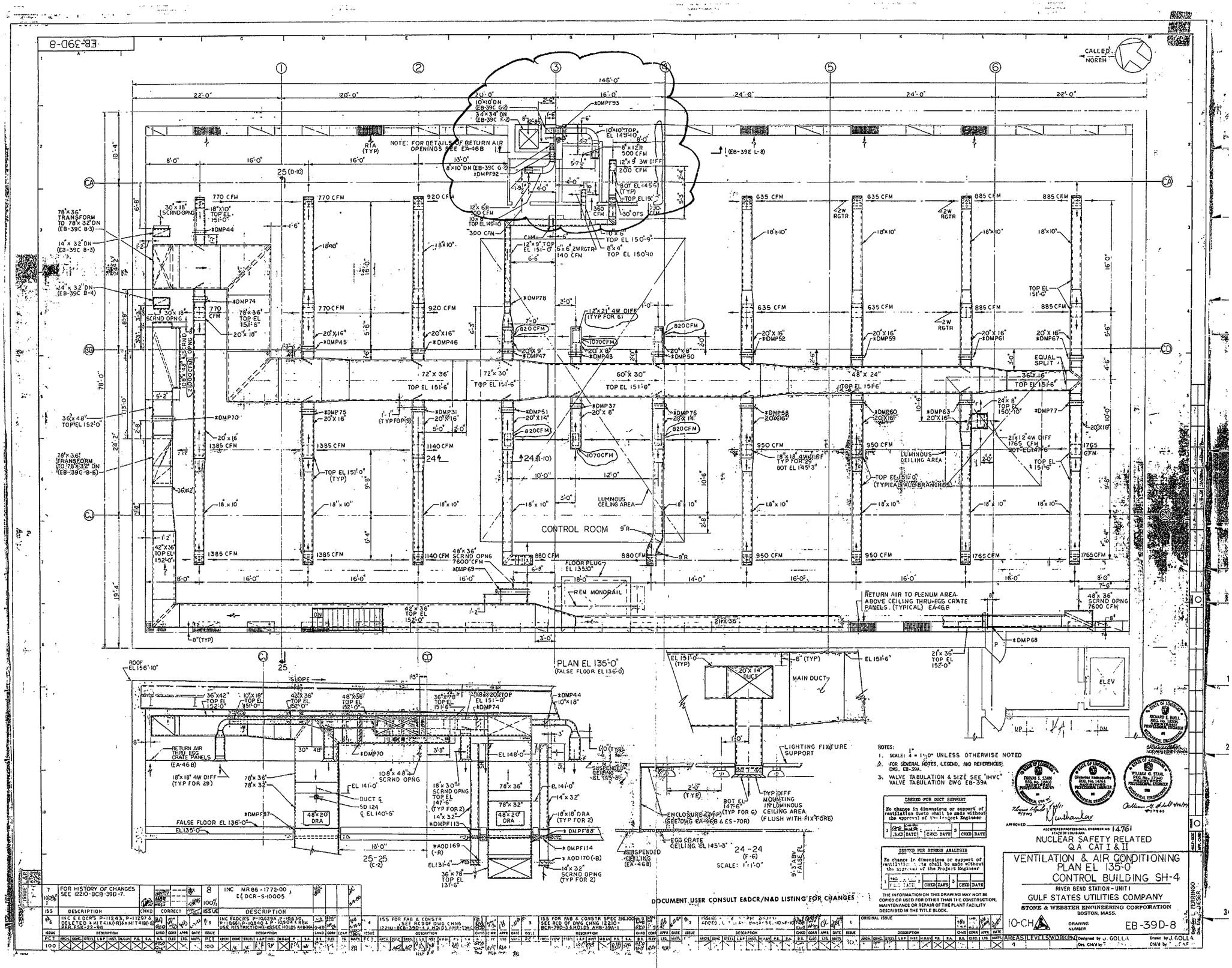

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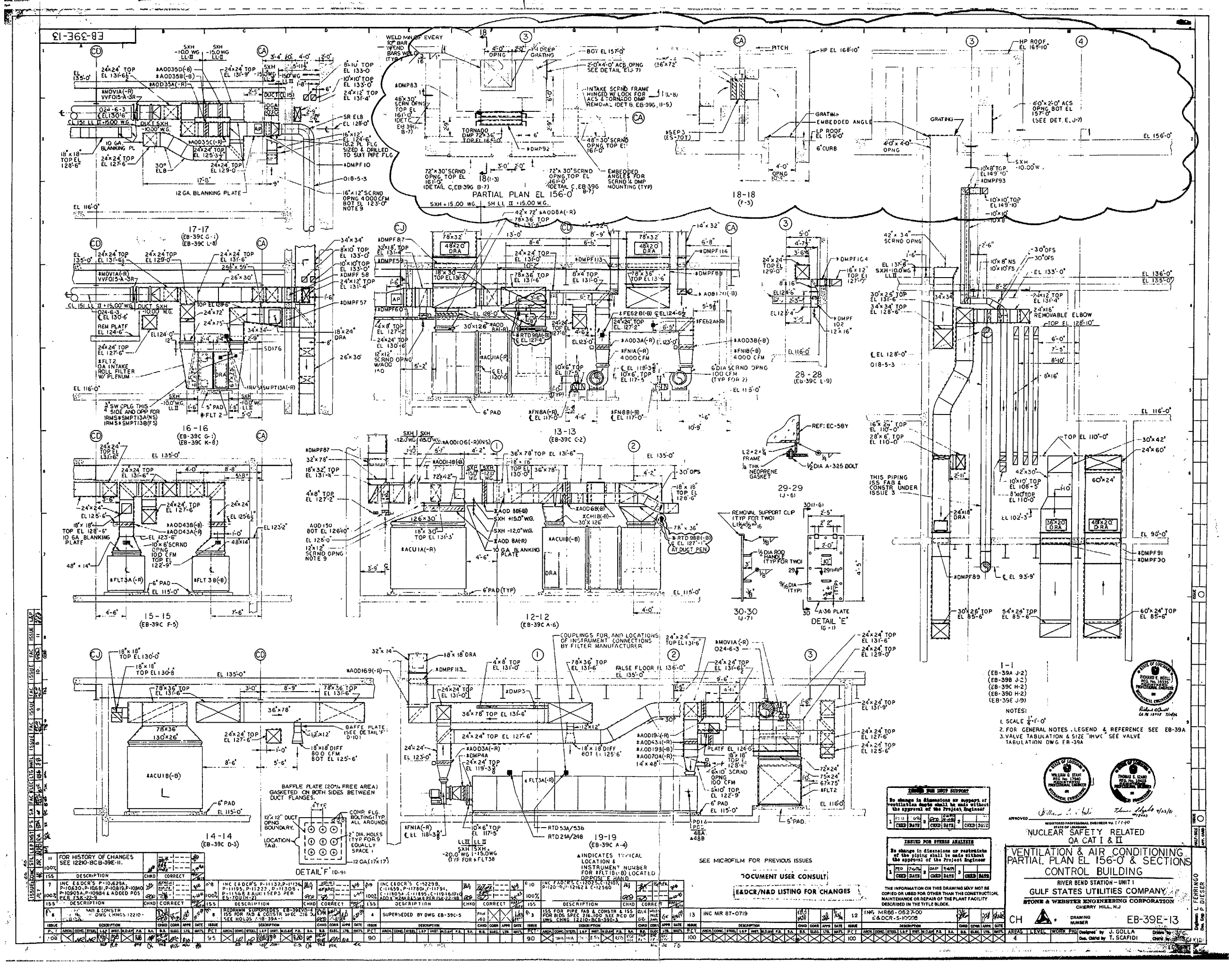

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