UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 17, 2000

Mr. Craig G. Anderson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: ARKANSAS NUCLEAR ONE, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: THE ELIMINATION OF POST ACCIDENT SAMPLING SYSTEM REQUIREMENTS (TAC NOS. MA6062 AND MA6063)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 208 to Facility Operating License No. DPR-51 and Amendment No. 218 to Facility Operating License No. NPF-6 for Arkansas Nuclear One, Units 1 and 2 (ANO-1 and ANO-2), respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated July 14, 1999 (0CAN079901), as supplemented by letters dated February 24, 2000 (0CAN020006), and July 17, 2000 (0CAN070003).

The amendments delete requirements from the TSs to maintain a Post Accident Sampling System (PASS). In the aftermath of the accident at Three Mile Island (TMI), Unit 2, the Nuclear Regulatory Commission (NRC) imposed requirements on licensees for commercial nuclear power plants to install and maintain the capability to obtain and analyze post-accident samples of the reactor coolant and containment atmosphere. The desired capabilities of the PASS were described in NUREG-0737, "Clarification of TMI Action Plan Requirements." The NRC issued orders to licensees with plants operating at the time of the TMI accident to confirm the installation of PASS capabilities (generally as they had been described in NUREG-0737). A requirement for PASS and related administrative controls was added to the TSs of the operating plants and was included in the initial TSs for plants licensed during the 1980s and 1990s. Additional expectations regarding PASS capabilities were included in Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident."

Significant improvements have been achieved since the TMI accident in the areas of understanding risks associated with nuclear plant operations and developing better strategies for managing the response to potentially severe accidents at nuclear plants. Recent insights about plant risks and alternate severe accident assessment tools have led the NRC staff to conclude that some TMI Action Plan items can be revised without reducing the ability of licensees to respond to severe accidents. The NRC's efforts to oversee the risks associated with nuclear technology more effectively and to eliminate undue regulatory costs to licensees and the public have prompted the NRC to consider eliminating the requirements for PASS in TSs and other parts of the licensing bases of operating reactors.

The staff has completed its review of the topical reports submitted by the Combustion Engineering Owners Group (CEOG) and the Westinghouse Owners Group (WOG) that proposed the elimination of PASS. In addition, the staff has completed its review of the site-specific amendment request you submitted for ANO-1 and ANO-2. Your evaluation closely followed the methodology adopted by the CEOG. The justifications for the proposed elimination of PASS requirements center on evaluations of various radiological and chemical sampling and their potential usefulness in responding to a severe reactor accident or making decisions regarding actions to protect the public from possible release of radioactive materials. As explained in more detail in the staff's Safety Evaluation for the two topical reports and the site-specific amendment for ANO-1 and ANO-2, the staff has reviewed the available sources of information for use by decision-makers in developing protective action recommendations and assessing core damage. Based on this review, the staff found that the information provided by PASS is either unnecessary or is effectively provided by other indications of process parameters or measurement of radiation levels. Therefore, as a result of these amendments deleting requirements for PASS from the ANO-1 and ANO-2 TSs, it may be appropriate to revise (as necessary) other elements of the licensing bases and pursue design changes to alter or remove existing PASS equipment.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,
/RA/
William Reckley, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-313 and 50-368

Enclosures: 1. Amendment No. 208 to DPR-51
2. Amendment No. 218 to NPF-6
3. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 208
License No. DPR-51

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated July 14, 1999, as supplemented by letters dated February 24, 2000, and June 17, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. DPR-51 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 208 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Gramm

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 17, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 208

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
126	126

6.6 DELETED

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least hot shutdown within one hour.

b. The Nuclear Regulatory Commission shall be notified pursuant to 10 CFR 50.72 and a report submitted pursuant to the requirements of 10 CFR 50.36 and Specification 6.6.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. (Deleted)

e. (Deleted)

f. Fire Protection Program Implementation.

g. New and spent fuel storage.

h. Offsite Dose Calculation Manual and Process Control Program implementation at the site.

UNITED STATES
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ENTERGY OPERATIONS, INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 218
License No. NPF-6

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated July 14, 1999, as supplemented by letters dated February 24, 2000, and July 17, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-6 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 218 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A Gramm

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 17, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 218

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
6-13	6-13

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The unit shall be placed in at least HOT STANDBY within one hour.
b. The Vice President, Operations ANO and the SRC shall be notified within 24 hours.
c. The Nuclear Regulatory Commission shall be notified pursuant to 10CFR50.72 and a report submitted pursuant to the requirements of 10CFR50.36 and Specification 6.6.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.
b. Refueling operations.
c. Surveillance and test activities of safety related equipment.
d. (Deleted)
e. (Deleted)
f. Fire Protection Program implementation.
g. Modification of Core Protection Calculator (CPC) Addressable Constants. These procedures should include provisions to assure that sufficient margin is maintained in CPC Type I addressable constants to avoid excessive operator interaction with the CPCs during reactor operation.

NOTE: Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P that has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.

h. New and spent fuel storage.
i. ODCM and PCP implementation.

6.8.2 Deleted

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 208 AND 218 TO

FACILITY OPERATING LICENSE NOS. DPR-51 AND NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNITS 1 AND 2

DOCKET NOS. 50-313 AND 50-368

1.0 INTRODUCTION

By letter dated July 14, 1999, as supplemented by letters dated February 24, 2000, and July 17, 2000, Entergy Operations, Inc. (Entergy or the licensee) submitted a request for changes to the Arkansas Nuclear One, Units 1 and 2 (ANO-1 and ANO-2), Technical Specifications (TSs). The requested changes would delete requirements from the TSs to maintain a Post Accident Sampling System (PASS). Licensees were required to implement PASS upgrades as a result of NUREG-0737, "Clarification of TMI [Three Mile Island Nuclear Station] Action Plan Requirements," and Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." Implementation of these upgrades were an outcome of the Nuclear Regulatory Commission's (NRC or the Commission) lessons learned from the accident that occurred at TMI, Unit 2. Entergy has stated that the information obtained using PASS is not required for the development of protective action recommendations (PARs) or for core damage assessment.

The February 24, 2000, and July 17, 2000, supplemental letters provided clarifying information that did not change the scope of the July 14, 1999, application nor the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

The need for a PASS was one of the findings endorsed by the Commission following the accident at the TMI plant. The Commission specified that all licensed plants have the capability of obtaining and analyzing post-accident samples of the reactor coolant and containment atmosphere, within specified times, without causing radiation exposure to any individual exceeding 5 rem to the whole body or 75 rem to the extremities. Detailed criteria for the PASS are specified in Section II.B.3 of NUREG-0737, including the following:

> *The licensee and applicant shall establish an onsite radiological and chemical analysis capability to provide, within a three-hour time frame, quantification of the following:*
>
> a) *Certain radionuclides in the reactor coolant and containment atmosphere*

b) hydrogen levels in the containment atmosphere

c) dissolved gases (e.g., H_2), chloride, and boron concentration of liquids

The TMI-related recommendations specified in NUREG-0737 were subsequently incorporated into 10 CFR 50.34(f)(2)(viii). However, this rule applied only to applications pending at that time (i.e., Perkins Nuclear Station, Units 1, 2, and 3; Allens Creek Nuclear Generating Station, Unit 1; Pebble Springs Nuclear Plant, Units 1 and 2; Black Fox Station, Units 1 and 2; Skagit/Hanford Nuclear Power Project, Units 1 and 2; and Offshore Power Systems).

On March 17, 1982, the NRC issued Generic Letter (GL) 82-05, "Post-TMI Requirements," in which the NRC requested that licensees establish a firm schedule for implementing post-accident sampling. On November 1, 1983, NRC issued GL 83-36 and GL 83-37, "NUREG-0737 Technical Specifications," which provided guidance on how to address post-accident sampling in the TSs for boiling-water reactors (BWRs) and pressurized-water reactors (PWRs), respectively. In GL 83-36 and GL 83-37, the NRC indicated that all licensees should establish, implement, and maintain an administrative program that would include training of personnel, and provide procedures for sampling and analyses, and provisions for sampling and analysis equipment. The licensees could elect to reference this program in the administrative controls section of the TSs and include its detailed description in the plant operation manuals. However, the recommendations described in Section II.B.3 of NUREG-0737 were imposed as requirements for the majority of operating plants through license conditions or by orders.

Regulatory Guide 1.97 describes acceptable means for licensees to comply with the Commission's regulations (General Design Criteria 13, 19, and 64 of Appendix A to 10 CFR Part 50) to provide instrumentation to monitor plant variables and systems during and following an accident. Regulatory Guide 1.97 included a list of variables to be monitored which included the samples specified in NUREG-0737 and the following additional samples:

- pH in the RCS [reactor coolant system]
- boron, pH, chlorides, and radionuclides in the containment sump

Since these criteria for PASS have been issued, the NRC staff have performed three generic evaluations pertinent to the elimination of some or all of the requirements for PASS. These are discussed below.

In the mid-1980s, the staff sponsored a contractor to review regulatory requirements that may have marginal importance to risk. One of the issues reviewed was the NUREG-0737 criteria for PASS. The conclusion, reported in NUREG/CR-4330, "Review of Light Water Reactor Requirements: Volume 3" (May 1987), was that several of the PASS criteria could be relaxed without impacting safety. However, the staff did not take action to modify the PASS criteria based upon the contractor's conclusions.

In 1993, during its review of licensing issues pertaining to evolutionary and advance light water reactors, the staff evaluated requirements for PASS specified in 10 CFR 50.34(f)(2)(viii). The staff recommended to the Commission in SECY 93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (AWLR) Designs" (April 2,

1993), that: (1) elimination of hydrogen analysis of containment atmosphere samples is appropriate, given that the safety grade hydrogen monitoring instrumentation will be installed; (2) relaxation of dissolved gas (including dissolved hydrogen) sampling time to 24 hours is appropriate; (3) elimination of the mandatory requirement for chloride samples is appropriate; (4) relaxation of the boron sampling time to 8 hours after an accident is appropriate; and (5) relaxation of the sampling time for radionuclides (used to determine the degree of core damage) to 24 hours is appropriate.

In addition, in 1993, the staff evaluated the Combustion Engineering Owners Group (CEOG) Topical Report CEN-415, "Modifications of Post Accident Sampling System Requirements," (Revision 1, December 1991). In a letter dated April 12, 1993, the NRC approved: (1) deletion of pH measurement in the containment sump, (2) deletion of hydrogen sampling of the containment atmosphere, (3) deletion of sampling for iodine (if core damage assessment procedures are based on samples of xenon or krypton activities), and (4) deletion of oxygen analysis of reactor coolant.

By letter dated March 30, 2000, as supplemented by letter dated April 14, 2000, the CEOG submitted Topical Report CE NPSD-1157, Revision 1, "Technical Justification for the Elimination of the Post-Accident Sampling System from the Plant Design and Licensing Basis for CEOG Utilities." CE NPSD-1157 evaluated information obtained from PASS samples to determine its contribution to plant safety and accident recovery. The report considered the progression and consequences of core damage accidents and assessed the accident progression with respect to plant abnormal and emergency operating procedures, severe accident management guidance, and emergency plans. CE NPSD-1157 concluded that all of the current PASS samples specified in NUREG-0737 may be eliminated. Specifically, CE NPSD-1157 recommended the following regarding the PASS:

1. Eliminate PASS sampling of RCS dissolved gases,
2. Eliminate PASS sampling of RCS hydrogen,
3. Eliminate PASS sampling of RCS oxygen,
4. Eliminate PASS sampling of RCS pH,
5. Eliminate PASS sampling of RCS chlorides,
6. Eliminate PASS sampling of RCS boron,
7. Eliminate PASS sampling of RCS conductivity,
8. Eliminate PASS sampling of radionuclides in the RCS,
9. Eliminate PASS sampling of containment hydrogen,
10. Eliminate PASS sampling of containment oxygen,
11. Eliminate PASS sampling of radionuclides in the containment atmosphere,

12. Eliminate PASS sampling of containment sump pH,

13. Eliminate PASS sampling of chlorides in the containment sump,

14. Eliminate PASS sampling of boron in the containment sump, and

15. Eliminate PASS sampling of radionuclides in the containment sump.

The NRC staff has completed its review of CE NPSD-1157 and concluded that PASS requirements can be eliminated. The bases for the staff's conclusions are documented in a safety evaluation dated May 16, 2000 (Accession No. ML003715250). The CEOG topical report identified the 15 specific sampling requirements that are satisfied through the incorporation of a PASS. This report provided a detailed rationale demonstrating that the information provided by the PASS is not required for identification, mitigation, or personnel protection in the event that a reactor accident were to occur. The topical report showed that there are sufficient alternate plant indications available for operators to adequately identify plant conditions and take the appropriate actions when mitigating the consequences of a postulated accident. In addition, the report demonstrated that the appropriate PARs could be made through the use of alternative instrumentation without reliance on PASS. In the majority of cases, the accident progression is such that PARs would be made in advance of the availability of information from PASS. Therefore, PASS would only provide confirmatory information that the appropriate PARs were made. Therefore, since PASS is not required for the development of PARs or required for core damage assessment methodologies, these sampling requirements can be eliminated. The staff concluded that this report is acceptable as a reference for individual, plant-specific licensing applications for applicable Combustion Engineering (CE) plants, which are subject to the limitations specified in the topical report and in the associated safety evaluation developed by the NRC staff.

Furthermore, in parallel with review of CE NPSD-1157, the staff reviewed Westinghouse Owners Group (WOG) Topical Report WCAP-14986-P, Revision 1, "Westinghouse Owners Group Post Accident Sampling System Requirements: A Technical Basis," (August 1998), which requested similar changes to PASS requirements for Westinghouse PWRs.

WCAP-14986-P evaluated the same 15 PASS sampling requirements and similarly concluded that these requirements could be eliminated. The NRC staff has completed its review of the topical report and concluded that PASS requirements can be eliminated. The basis for the staff's conclusion is documented in a safety evaluation dated June 14, 2000 (Accession No. ML003723268). The staff concluded that this report is acceptable for referencing in licensing applications for Westinghouse plants subject to the limitations specified in the report and in the associated NRC staff's safety evaluation.

Entergy has submitted an application to amend the TS Administrative Control requirements for ANO-1 and ANO-2 to remove the requirement to maintain a PASS. The requirement to maintain the capability for post-accident sampling was originated in NUREG-0737, Item II.B.3, clarified in Regulatory Guide 1.97, Revision 3, and implemented through a Confirmatory Order dated March 14, 1983.

Entergy has submitted a site-specific application requesting the elimination of these post-accident sampling requirements. The nuclear steam supply system (NSSS) for ANO-2

was designed by CE. ANO-2 is a member of the CEOG and is the lead plant in applying topical report CE NPSD-1157. Therefore, the review of the submittal for ANO-2 will ensure that the conditions specified in the topical report and the NRC staff's corresponding safety evaluation were satisfied. The NSSS for ANO-1 was designed by Babcock and Wilcox (B&W). The B&W Owners Group (BWOG) has not submitted a topical report of the elimination of post-accident sampling requirements. Therefore, ANO-1 will be reviewed as a site-specific amendment without reliance on a topical report. Entergy indicated that grouping ANO-1 and ANO-2 into a single submittal was justified because PASS is identical for both units, both units use similar methodologies for accident mitigation and control of offsite releases, and both units contain similar redundant equipment and instrumentation that may be used for accident analysis without the reliance on PASS. Therefore, as requested, the NRC staff will review the site-specific application for both of these units as a single site-specific application.

In the years following the issuance of NUREG-0737, a considerable amount of knowledge and operating experience has been gathered concerning core behavior and the role that the PASS would play in various accident scenarios. Commercial nuclear sites throughout the United States have developed guidelines designed specifically to deal with severe accidents. These guidelines are commonly referred to as severe accident management guidelines (SAMGs). The addition of post-accident qualified instruments has precluded the need for PASS. Operators are trained to understand the phenomenon behind indications they might encounter during accident conditions and to take action accordingly. The development and improvement of emergency operating procedures (EOPs), abnormal operating procedures, and emergency plan implementing procedures (EPIPs) has created greater capability in identifying and mitigating accident scenarios, in addition to providing offsite PARs associated with the event in progress. Because of the normally rapid succession of accident scenarios and the redundant equipment and instrumentation provided for the operator and plant staff, the role of PASS in emergency response has become primarily confirmatory.

3.0 EVALUATION

The work performed by the CEOG and the WOG in the development of their respective topical reports provided a comprehensive foundation of information regarding the history and regulatory basis for the post-accident sampling requirements. In addition, this effort produced an acceptable basis for the elimination of these requirements. Entergy has submitted a site-specific application for the elimination of post-accident sampling requirements for both ANO-1 and ANO-2. The following evaluation will address the 15 sampling requirements identified in the topical reports and addressed in the associated NRC staff safety evaluations. This evaluation will consider the licensee's basis for the elimination of post-accident sampling requirements and summarize the staff's conclusion regarding the licensee's position.

3.1 Eliminate PASS Sampling of RCS Dissolved Gases

Dissolved gas sampling is specified in NUREG-0737 and Regulatory Guide 1.97. However, NUREG/CR-4330 suggests that it could be eliminated, provided that vessel head gas vents and a reactor vessel level instrumentation system (RVLIS) are installed.

The main purpose of sampling for dissolved gases is to identify the potential of void formation in the reactor vessel dome when depressurizing or even uncovering the core in case natural circulation needs to be used for decay heat removal. Because RVLIS (which is safety grade for

ANO-1 and ANO-2) provides an indication of water level and the reactor vessel and pressurizer head vents (which are also safety grade for both ANO-1 and ANO-2) can vent non-condensable gases, both diagnosis and remediation are available. In addition, procedures and training have been enhanced to provide for detection of voids and the guidance on methods available to eliminate them. Operators are trained to detect voids by monitoring the RCS level response to pressurizer spray and heater operations. Regardless of the type of cooldown in progress, procedures provide instructions that aggressively pursue void elimination.

In addition, ANO-1 and ANO-2 are not equipped with automated gas sampling systems. Therefore, the inherent delay between sampling and the availability of the sample result is long and of no practical significance in formulating PARs and would only be used to provide confirmatory information.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of RCS dissolved gases is acceptable.

3.2 Eliminate PASS Sampling of RCS Hydrogen

PASS sampling of the reactor coolant for measurement of dissolved hydrogen is specified in NUREG-0737 and Regulatory Guide 1.97.

As in the case of dissolved gases, the main purpose of hydrogen sampling is identification of the potential of void formation in the reactor vessel dome (and at the top of the hotleg candy canes for ANO-1 and at the top of the steam generator U tubes for ANO-2) or uncovering the core when depressurizing in case natural circulation needs to be used for decay heat removal. In addition, the amount of the dissolved hydrogen could act as a surrogate indicator for dissolved fission product and non-condensable gases. As in the case of dissolved gases, the RVLIS system and the reactor vessel and pressurizer head vents can be used to both identify and vent non-condensables from the RCS when depressurizing in order to establish natural circulation.

Entergy stated that information on dissolved hydrogen in the RCS may be used to refine core damage assessments but determined that this refinement is not needed. Entergy indicated that reliance on dissolved hydrogen information in the core damage assessment methodology (CDAM) could be eliminated without significantly altering the assessment results. This activity will be completed prior to implementation of the license amendment to eliminate PASS requirements. The condition of the core may be determined based on available dose rate data and samples taken from the normal sample system. Sampling for cases with up to 5% clad failure can be accomplished with the normal sample system. In addition, in-line activity monitors for both units may be used to assess the core condition throughout the event. For larger failed fuel events, installed radiation monitoring devices and pressure/temperature instrumentation can provide the necessary information to confirm core conditions.

The staff concludes that the RVLIS system and the reactor vessel and pressurizer head vents eliminate the safety concern of dissolved hydrogen coming out of solution and inhibiting natural circulation. The staff agrees that dose rate data, sampling, installed radiation monitoring devices, and pressure/temperature instrumentation can provide the necessary information to adequately assess core conditions.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of RCS hydrogen is acceptable.

3.3 Eliminate PASS Sampling of RCS Oxygen

PASS sampling of the reactor coolant for measurement of oxygen is only recommended in NUREG-0737, but is specified in Regulatory Guide 1.97, whenever concentration of chlorides exceeds 1.5 ppm. The NRC approved elimination of measurement of this parameter from PASS for CE plants in its 1993 safety evaluation of CEN-415. This relaxation was not requested by the WOG or the BWOG at that time.

High concentration of oxygen in the RCS can enhance stress corrosion cracking (SCC) of stainless steel components caused by the presence of chlorides. However, this type of corrosion can be mitigated by the control of reactor coolant pH without regard to the oxygen concentration. In addition, whenever needed, oxygen concentration can be estimated from the oxygen concentrations in the borated water storage tank (BWST) for ANO-1 or the refueling water storage tank (RWST) for ANO-2.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of RCS oxygen is acceptable.

3.4 Eliminate PASS Sampling of RCS pH

PASS measurement of the reactor coolant pH is specified in Regulatory Guide 1.97. The NRC approved elimination of measurement of this parameter from PASS for CE plants in its safety evaluation of CEN-415. This relaxation was not requested by the WOG or the BWOG.

Reactor coolant pH control is important for controlling SCC of stainless steel components and for iodine retention. However, PASS sampling of RCS pH is not needed since its value can be satisfactorily estimated by calculation. In the post-accident recirculation environment, the pH of the coolant is maintained alkaline through an active sodium hydroxide addition to the containment spray for ANO-1 or a passive pH control through the use of pre-staged trisodium phosphate in the containment sump for ANO-2.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of RCS pH is acceptable.

3.5 Eliminate PASS Sampling of RCS Chlorides

PASS sampling of the RCS for measurement of chlorides is specified in NUREG-0737 and Regulatory Guide 1.97.

High concentration of chlorides in the RCS can cause SCC of stainless steel components. Chlorides are introduced to the RCS by different sources which may include the containment sump. However, the pH of sump water is maintained alkaline and, therefore, the presence of chlorides will not cause corrosion damage. Also, operators are aware when contaminated water enters from other chloride containing sources and can take appropriate corrective actions to prevent corrosion damage.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of RCS chlorides is acceptable.

3.6 Eliminate PASS Sampling of RCS Boron

PASS sampling of the reactor coolant for measurement of boron is specified in NUREG-0737 and Regulatory Guide 1.97. In addition, the staff recommended to the Commission in SECY 93-087, that the capability to obtain PASS samples of RCS boron within 8 hours of accident initiation be maintained for ALWRs.

EOPs are utilized for the identification of the boron dilution level through the transient. Boron sampling is usually used for backup informational purposes and the results of these surveys are not used in the EOP progression. The licensee indicated that other corroborative evidence would suffice to prevent re-criticality. Control rod assembly position indication, startup rate, or indication of high boron concentration flow into the vessel are examples of such corroborating evidence.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of RCS boron is acceptable.

3.7 Eliminate PASS Sampling of RCS Conductivity

PASS sampling of the reactor coolant for measuring conductivity is not specified in NUREG-0737 nor Regulatory Guide 1.97.

The CEOG and WOG topical reports and the associated NRC staff safety evaluation addressed PASS sampling for RCS conductivity. The measurement of reactor coolant conductivity can be used to verify pH measurements and has no other use. The licensee did not specifically mention conductivity sampling in their application. However, the licensee did propose the elimination of pH sampling in their application as discussed in Section 3.4 of this safety evaluation. Conductivity sampling was never required by the NRC. Therefore, the staff concludes that the licensee's proposal is acceptable.

3.8 Eliminate PASS Sampling of Radionuclides in the RCS

For the purpose of this discussion, the presentation of the reactor coolant sample analysis capabilities are also applicable for the containment sump sample. PASS sampling of the reactor coolant for measurement of radionuclides is specified in NUREG-0737 and Regulatory Guide 1.97. PASS sampling of the reactor sump is specified in Regulatory Guide 1.97. NUREG-0737 specifies that the PASS have the capability to promptly (i.e., within 3 hours) quantify certain radionuclides that are indicators of the degree of core damage. Furthermore, Regulatory Guide 1.97 specifies that the isotopic analysis serves the purpose of accident release assessment.

In order to comply with emergency classifications of the current EPIPs, the licensee indicated that sampling of RCS radionuclides may be used for core damage assessment and required during low failed fuel events for the declarations of a Notification of Unusual Event and Alert. Activity levels for these classifications are equivalent to 0.1% and 1% clad failure values,

respectively. With regard to core damage assessment, the licensee states that measurement of radionuclides with PASS is not needed because there are other independent means of estimating core damage that are simpler to perform which do not utilize RCS radionuclide information from PASS. The CDAMs provide four procedures for estimating the degree of core damage, one of which is the radionuclide assessment of reactor coolant. During significant core damage events, much of the radioisotopes have left the RCS or plate out away from the sample point, resulting in a non-respresentative sample. Reliance on the three remaining CDAM protocols (core exit thermocouples (CET), containment hydrogen, and containment radiation) in conjunction with ongoing efforts provided by the SAMGs is sufficient in assessing the extent of core damage.

The letdown lines for both ANO-1 and ANO-2 contain activity monitors. If these lines are available during the progression of a postulated accident, operators would be able to trend RCS activity levels during the event. In addition, the licensee has performed radiological assessments of the normal sample system. The expected whole body dose rate at the 1% clad failure equivalent is 1.4 rem/hr. The dose rate accounts for shine from piping sources, sample piping, and other sampling activities. Sampling personnel could perform sampling activities in the normal sample room for up to 3.57 hours without exceeding the 5 rem whole body dose limit. This is adequate to obtain several samples of reactor coolant, if required. The dose rate expected for 5% clad failure conditions would be approximately 3.2 rem/hr. The licensee has stated that this dose rate evaluation can be used as an indicator of failed fuel and that the radiation levels at the normal sample system are adequate to assure sample panel access. In addition, this allows dose rate to be used as an indicator of clad failure for the declaration of a Notification of Unusual Event and Alert.

For larger failed fuel events, installed radiation monitoring devices and pressure/temperature instruments are used to monitor core conditions. For loss-of-coolant accidents internal to the containment building, containment area and high range monitors are an effective input in determining the state of the reactor core. For other failed fuel events that exceed Alert limits, RCS pressure and temperature data is effective in determining the existing core condition, along with data from radiation monitoring of support systems and buildings. The EOPs assume the core has been uncovered and fuel damage has resulted if temperatures indicate superheated conditions from the CETs. Containment hydrogen concentration can also be used as a indicator of core damage.

The licensee states that there is little expectation that the RCS sample will provide sufficiently accurate information to improve upon assessments performed by the simpler methods. The licensee states that the core damage assessment procedure should be changed to eliminate reliance on radionuclide measurements.

With regard to the use of radionuclide sample information for classifying events involving failed fuel, the licensee states that the event can be classified based upon the recognition of the initiating condition which caused the fuel failure rather than measurement of the degree of fuel failure. Furthermore, the licensee states that other indications of failed fuel such as normal sample room radiation levels, can be correlated to the degree of failed fuel.

The staff considers radionuclide sampling information to be useful in estimating the degree of core damage, but recognizes that there are limitations associated with its use, in particular regarding the time needed to obtain the sample. Therefore, the staff considers it more

appropriate for emergency response purposes to estimate the degree of core damage based upon real-time indications.

In addition, the staff considers radionuclide sampling information to be useful in classifying certain types of events (such as reactivity excursion or mechanical damage) which could cause fuel damage without having an indication of overheating on CETs. However, the staff agrees with the licensee's contention that other indicators of failed fuel such as letdown radiation monitors or normal sampling system, can be correlated to the degree of failed fuel. The licensee has made a regulatory commitment to develop, implement, and maintain a methodology in which emergency classifications for low failed fuel events for the declarations of a Notification of Unusual Event and Alert can be made through a measurement of the whole body dose rate at the normal sample sink. This methodology will allow for the estimate of core damage for conditions of up to and including 5% clad failure. (See Section 4.1, Regulatory Commitments, Item 4.1.1).

Notwithstanding the licensee's justification that RCS sampling is not necessary to support emergency response decision making, the staff considers that, if core damage were to occur, sampling of the RCS to ascertain the radionuclide content would be beneficial. This information would provide the public additional confidence that the licensee understood the condition of the core and the magnitude of any remaining threat that the accident may pose. The staff considers that, in light of the lack of need of RCS sampling to support emergency response decision making during the initial phases of an accident, it is not necessary to have dedicated equipment to obtain this sample in a prompt manner. However, the staff does conclude that contingency plans should be developed to support taking post-accident RCS samples. These plans should detail the plant's existing sampling capabilities and what actions (e.g., assembling temporary shielding) may be necessary to obtain and analyze highly radioactive samples. The licensee has made a regulatory commitment to develop, implement, and maintain contingency plans to obtain and analyze highly radioactive liquid samples from the RCS and the containment sump. (See Section 4.1, Regulatory Commitments, Item 4.1.2).

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of RCS radionuclides is acceptable.

3.9 Eliminate PASS Sampling of Containment Hydrogen

PASS sampling of the containment atmosphere for hydrogen measurement is specified in NUREG-0737 and Regulatory Guide 1.97.

Separate from the above requirements for grab sampling, continuous hydrogen monitors are required by 10 CFR 50.44(b)(1), NUREG-0737 Item II.F.1, and Regulatory Guide 1.97, and are relied upon to meet the data reporting requirements of 10 CFR Part 50, Appendix E, Section VI.2.a.(i)(4). By order dated September 28, 1998, the hydrogen monitors are required to be functional in a sufficiently timely manner to support the emergency plan and related activities such as guidance for severe accident management. Regulatory Guide 1.97 specifies that redundant, safety-grade monitors have a range of 0 to 10 volume percent. The quantity of hydrogen released to containment in most severe accidents would result in concentrations within this range. However, in the event that random or spontaneous ignition does not occur, continued hydrogen production from such mechanisms as core concrete interactions and radiolysis of reactor coolant could result in the concentration exceeding the range of the

monitors late in an event. The EPIP considers the containment buildings for ANO-1 and ANO-2 to be challenged when hydrogen concentrations above 3.5% are detected. In cases for hydrogen concentrations above 10 volume percent, severe accident management decision-making would rely on default hydrogen production assumptions contained in the SAMG. Since grab sample analysis provides the only viable means of determining the actual hydrogen concentration, once the hydrogen concentration exceeds the range of the monitors, there is some value to retaining the capability for long term hydrogen concentration analysis of containment atmosphere grab samples.

The staff concludes that during the early phases of an accident, the safety-grade hydrogen monitors provide an adequate capability for monitoring containment hydrogen concentration and are an acceptable alternative to maintaining the capability to obtain and analyze containment atmosphere samples for hydrogen within 3 hours. In view of the value of grab samples for complementing the information from the hydrogen monitors in the long term (i.e., by confirming the indications from the monitors and providing hydrogen measurements for concentrations outside the range of the monitors), the licensee has indicated that containment air samples may be obtained from the hydrogen analyzer/containment air monitoring system flow paths and that a sample could be obtained and analyzed.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of containment atmosphere hydrogen concentration is acceptable.

3.10 Eliminate PASS Sampling of Containment Oxygen

PASS sampling of the containment atmosphere for oxygen measurement is specified in Regulatory Guide 1.97.

Containment oxygen measurement is used to assess the combustibility of the containment atmosphere. For PWRs with large, dry containments, there is sufficient oxygen inside to support combustion. Risk studies of this containment type have shown that hydrogen combustion is most likely but containment failure is unlikely for degraded core accident sequences. This conclusion is based on the strength of large, dry containments. When needed, the oxygen concentration in containment can be sufficiently estimated by subtracting the partial pressure of steam, hydrogen, and air from the containment pressure. The partial pressure of steam is based on saturated conditions and the containment temperature. The partial pressure of hydrogen is based on the concentration of hydrogen, as indicated by the continuous hydrogen monitors, and the ideal gas law. The partial pressure of air is based on the initial conditions of containment.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of containment oxygen is acceptable.

3.11 Eliminate PASS Sampling of Radionuclides in the Containment Atmosphere

PASS sampling of the containment atmosphere for radionuclide measurement is specified in NUREG-0737 and Regulatory Guide 1.97. NUREG-0737 specifies that the PASS have the capability to promptly quantify certain radionuclides that are indicators of the degree of core damage. Furthermore, Regulatory Guide 1.97 specifies that the isotopic analysis serves the purpose of accident release assessment.

PASS measurements of the containment atmosphere radionuclide concentration are used to estimate the degree of core damage and to refine the source term used in dose assessments. In turn, core damage estimates and dose assessments are used in evaluating the type and extent of public protective actions which may be warranted. The licensee states that PASS sampling of containment atmosphere radionuclides can be eliminated because these samples are not representative of the concentration of radionuclides which may be released to the environment. The basis for this conclusion is that the concentration of the radionuclides at the sample point may not be representative of the concentration in containment due to the potential for revolitization of fission products upon containment depressurization, the plate out of aerosols (e.g., CsI) in the sample lines, and time delays associated with obtaining, processing, and interpreting the sample during non-stable phases of the accident. In addition, the licensee stated that samples of the containment atmosphere could be obtained and analyzed without reliance on the PASS.

The staff recognizes that, as described in Supplement 3 to NUREG-0654/FEMA-REP-1, Revision 2, "Criteria for Preparation and Evaluation of Emergency Response Plans and Preparedness in Support of Nuclear Plants," initial PARs should be based upon plant indications of actual or projected severe core damage. Following this initial PAR, the licensee should continue assessment of the accident to determine whether the PAR should be modified (relaxation of the PAR should not occur until the source of the threat is clearly under control). In NUREG-0654, the NRC indicated that the licensees' capability to perform this assessment should include the post-accident sampling capability. Therefore, the staff's evaluation of the licensee's recommendation for elimination of sampling the containment atmosphere for radionuclides focused on the need for this information to support whether initial PARs should be modified.

The staff agrees with the licensee's assessment regarding the limitations associated with obtaining representative samples of the containment atmosphere. The staff considers that these limitations should be taken into account when determining how to utilize the containment atmosphere sample information during an event. However, the staff position is that, due to these limitations, information obtained from PASS samples would not be a primary factor in licensee and offsite emergency response decision making regarding PARs during the early phases of an accident. However, the staff considers that containment atmosphere sample information would provide the public additional confidence that the licensee understood the magnitude of any remaining threat that the accident may pose after the accident had stabilized. Therefore, the staff also concludes that a plan should be developed for sampling the containment atmosphere. The staff does not consider it necessary to have dedicated equipment to obtain this sample in a prompt manner. These plans should detail the plant's existing sampling capabilities and what actions (e.g., assembling temporary shielding) may be necessary to obtain and analyze highly radioactive samples. The licensee has made a regulatory commitment to develop, implement, and maintain contingency plans to obtain and analyze highly radioactive samples from the containment atmosphere. (See Section 4.1, Regulatory Commitments, Item 4.1.2).

In addition, the licensee has made a regulatory commitment to develop, implement, and maintain contingency plans to demonstrate the offsite capability for monitoring radioactive iodines. (See Section 4.1, Regulatory Commitments, Item 4.1.3). Licensees currently have plans in place for the offsite monitoring of radioactive releases to ensure that appropriate,

immediate actions are taken to protect the health and safety of the public in the event of an accident and offsite release. This commitment is a compensatory action to ensure that the licensee has the capability to monitor the iodine component of the radioactive release to provide additional information about the long term ingestion pathways associated with the release, such that appropriate PARs can be made. This sampling activity provides indirect information about the containment source term and supports the elimination of sampling requirements for containment atmosphere radionuclides.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling and measurement of containment atmosphere radionuclides is acceptable.

3.12 Eliminate PASS Sampling of Containment Sump pH

PASS sampling of the containment sump for measurement of pH is specified in Regulatory Guide 1.97.

The containment sump pH plays an important role in controlling the post-accident chemistry of the containment sump water. If it becomes acidic, it can significantly affect chloride-induced SCC of stainless steel components and retention of iodine in sump water. In the post-accident, environment, the pH of the coolant is maintained alkaline through an active sodium hydroxide addition to containment spray for ANO-1 or a passive pH control through the use of pre-staged trisodium phosphate in the containment sump for ANO-2. However, for ANO-1, there may be some accident sequences when the containment spray is not activated, and sump pH may then become acidic. In addressing sump pH control concerns, the existing ANO-1 EOPs provide necessary guidance to ensure sodium hydroxide injection occurs, should the above scenario, or one similar to it, take place. Upon actuation of sump recirculation, operators are required to verify sodium hydroxide tank level. If the level in the tank indicates that full sodium hydroxide injection has not taken place, the operators are instructed to inject the remainder of the tank contents until a pre-established level is achieved. The containment sump pH value can be estimated with a sufficient degree of accuracy from the volumes and chemistries of the water incoming from different external sources that represent the major sources of acid.

The staff considers containment sump sampling to be useful for confirming sump pH calculations and to verify that unaccounted for acid sources have been successfully neutralized. The licensee has made a regulatory commitment to maintain the capability to obtain a liquid sample from the containment sump and analyze it for radionuclides as part of this amendment (See Section 3.8 and Section 4.1, Regulatory Commitments, Item 4.1.2). The use of the contingency plans for obtaining samples would depend on the plant conditions and the need for information by the decision-makers responsible for responding to the accident conditions.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling and measurement of containment sump pH is acceptable.

3.13 Eliminate PASS Sampling of Chlorides in the Containment Sump

PASS sampling and measurement of the containment sump for chlorides is specified in Regulatory Guide 1.97.

High concentration of chlorides in the containment sump can cause SCC of stainless steel components and affect retention of iodine. For plants with fresh water cooling systems, the problem is minimal. The pH of sump water is maintained alkaline and, therefore, the presence of chlorides will not cause corrosion damage or interfere with the retention of iodine. The volumes and chloride concentrations of the incoming water from different sources are known and the resulting concentration of chloride in the sump water can be estimated with a sufficient degree of accuracy.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of containment sump chlorides is acceptable.

3.14 Eliminate PASS Sampling of Boron in the Containment Sump

PASS sampling of the containment sump for measurement of boron concentration is specified in Regulatory Guide 1.97.

The purpose of measuring boron concentration in the containment sump is to assure the reactor would remain subcritical should sump water be used in the recirculation mode to cool the core. The water in the BWST and core flood tanks for ANO-1 and the RWST and safety injection tanks for ANO-2 have sufficient boron concentration to assure subcriticality at any time in the fuel cycle. Should unborated water be introduced in the containment sump for emergency core cooling, the sump boron concentration will be lower. However, the licensee stated that the sump level (and the corresponding amount of water) and sump water temperature are known, which allow an estimate to be made for the boron concentration.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of containment sump boron is acceptable.

3.15 Eliminate PASS Sampling of Radionuclides in the Containment Sump

This is discussed in Section 3.8.

4.0 <u>REGULATORY COMMITMENT SUMMARY</u>

The staff concludes that the licensee's application provides a sufficient technical basis to allow for elimination of requirements made to obtain the following PASS samples specified in NUREG-0737 and Regulatory Guide 1.97:

1. RCS dissolved gases
2. RCS hydrogen
3. RCS oxygen
4 RCS pH
5. RCS chlorides
6. RCS boron
7. RCS conductivity
8. RCS radionuclides
9. Containment atmosphere hydrogen
10. Containment atmosphere oxygen
11. Containment atmosphere radionuclides

12. Containment sump pH
13. Containment sump chlorides
14. Containment sump boron
15. Containment sump radionuclides

This amendment supercedes the associated requirements to obtain the above mentioned PASS sample requirements of NUREG-0737 and Regulatory Guide 1.97, and the implementing Confirmatory Order dated March 14, 1983.

4.1 Regulatory Commitments

The licensee has made the following commitments in support of their application which is requesting the elimination of the requirements for a PASS. All commitments shall be implemented prior to implementation of the requested amendment.

4.1.1 Develop, implement, and maintain the capability for classifying fuel damage events at the Alert level threshold. This capability may utilize the normal sampling system or correlate normal sample system dose rates to coolant concentrations.

4.1.2 Develop, implement, and maintain contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, the containment sump, and containment atmosphere. The contingency plans do not have to be demonstrated. Because these are contingency plans, the staff concludes that, in accordance with 10 CFR 50.47 and Appendix E to 10 CFR Part 50 for emergency plans, these contingency plans must be available to be used by the licensee during an accident; however, these contingency plans do not have to be carried out in emergency plan drills or exercises.

4.1.3 Develop, implement, and maintain the offsite capability to monitor radioactive iodines.

The staff concludes, based upon the justification provided in the licensee's application, combined with the above mentioned commitments, that there is reasonable assurance that the health and safety of the public will not be endangered by operation of ANO-1 and ANO-2 without PASS.

The NRC staff finds that reasonable controls for implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitments are best provided by the licensee's administrative control process, including its commitment management program. The above regulatory commitments do not warrant the creation of regulatory requirements.

5.0 CHANGES TO THE TECHNICAL SPECIFICATIONS

The licensee has proposed to delete the following item from ANO-1, TS 6.8, "Procedures and Programs":

i. Post accident sampling (includes sampling of reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and the containment atmosphere).

The licensee has proposed to delete the following item from ANO-2, TS 6.8, "Procedures and Programs":

j. Post accident sampling (includes sampling of reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and the containment atmosphere).

Based on the review described in Section 3.0 and the regulatory commitments discussed in Section 4.0, the staff finds this change to be acceptable.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State official was notified of the proposed issuance of the amendment. The State official had no comment.

7.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (64 FR 43773, August 11, 1999). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). These amendments also change administrative requirements. Therefore, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

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Date: August 17, 2000