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2CAN010203

January 31, 2002

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Arkansas Nuclear One – Unit 2
Docket No. 50-368
License Amendment Request
Revision of Section 6.0, Administrative Controls For Consistency with ANO-1 Improved Technical Specifications

REFERENCES:

1. Letter Dated August 22, 2001, Proposed Emergency Plan Change (0CAN080103)
2. Letter Dated February 22, 1999, Additional Information Concerning Proposed Administrative Controls Technical Specifications Changes (0CAN029902)
3. Letter Dated March 7, 1997, Issuance of Amendment No. 180 to Facility Operating License No. NPF-6 – Arkansas Nuclear One, Unit 2 (TAC NO. M77399) (2CNA039701)
4. Letter Dated July 31, 2001, Change to the ANO-2 Reactor Coolant Pump Flywheel Inspection Interval Surveillance Requirements (2CAN070107)

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Arkansas Nuclear One, Unit 2 (ANO-2) hereby requests the following amendment: 1) reorganization of the ANO-2 Technical Specifications (TS) Section 6.0, Administrative Controls to be consistent with NUREG-1432, Revision 2, "*Standard Technical Specifications Combustion Engineering Plants*" and the ANO-1 TSs; 2) modification of the actions and surveillance requirements associated with various TSs; 3) changes to various TS bases as needed to support the above changes; and 4) the addition of a TS Bases Control Program. The changes result in consistency between the Administrative Controls Sections of the ANO-2 TSs and the approved ANO, Unit 1 improved TS, which will be implemented in the summer of 2002. The proposed change has been evaluated in accordance with 10CFR50.91(a)(1) using criteria in 10CFR50.92(c) and it has been determined that this change involves no significant hazards considerations. The bases for these determinations are included in the attached submittal.

ANO-2 requests approval of the proposed amendment by July 2002 with an implementation period of 120 days. This is consistent with the implementation date of the ANO-1 improved TSs. Although this request is neither exigent nor emergency, your prompt review is requested.

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The proposed change includes new commitments as summarized in Attachment 4. If you have any questions or require additional information, please contact Ms. Dana Millar at 601-368-5445.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 31, 2002.

Sincerely,

CGA/dm

Attachments:
1. Analysis of Proposed Technical Specification Change
2. Proposed Technical Specification Changes (mark-up)
3. Changes to Technical Specification Bases Pages (mark-up)
4. List of Regulatory Commitments
5. Proposed Technical Specification Changes (clean pages)

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ATTACHMENT 1

2CAN010203

Analysis of Proposed Technical Specification Change

1.0 DESCRIPTION

This letter is a request to amend Operating License NPF-6 for Arkansas Nuclear One, Unit 2 (ANO-2).

The proposed changes will revise the Operating License to:

1) Delete two license conditions.

2) Reorganize the ANO-2 Technical Specifications (TS) Section 6.0, Administrative Controls;

3) Move several surveillance requirements that are considered programs from their present location in the individual specifications to TS Section 6.0;

4) Modify TS actions related to the Control Room Ventilation System to make them consistent with the proposed changes to the ANO-1 TS;

5) Make appropriate changes to the TS bases in support of the above changes; and

6) Add a TS Bases Control Program.

The proposed changes will make the ANO-1 and ANO-2 common unit TSs and Administrative Controls sections similar. Additionally, the majority of the proposed change to the Administrative Controls section is consistent with Revision 2 of NUREG-1432, *"Standard Technical Specifications Combustion Engineering Plants."*

2.0 PROPOSED CHANGE

The following provides a description of the proposed changes to the ANO-2 TSs. Many of the current requirements will be deleted because they are duplicated in the Code of Federal Regulations (CFR). The applicable section of the CFR will be referenced in the description of the change. Note the majority of the Administrative Controls section of the ANO-2 TSs is being re-organized and as such the current TS amendment reference numbers listed at the bottom of each page may no longer be associated with the information contained on the new page. Thus, with this proposed change the amendment tracking for this section will now be re-baselined and will reflect only this amendment (i.e., prior amendment references will be deleted).

The table below is organized by sections of the Facility Operating License and TSs in the order of the current TSs. Where a section in the Administrative Controls section is currently not used or defined as deleted no discussion is included. These sections may be used in the proposed re-organization of that section. Due to the complexity of this change Attachment 5 is included to provide clean pages for reference only. The final pages will be supplied at a later date. For additional reference Attachment 5 includes a cross-reference index of the new Administrative Controls Section TS location with the current TS location.

Current TS #	New Location	Description of Change
Facility Operating License (FOL)		
FOL 2.C.(3) (p)	6.5.10	The Secondary Water Chemistry Monitoring program described in this section of the FOL will be moved to Administrative Controls section 6.5.10. Reference to the specific title of the procedure will be deleted. No technical changes are proposed.
FOL 2.C.(5)	6.5.2	This requirement to establish a program to reduce leakage from systems outside containment that could or would contain highly radioactive fluids during transients will be moved to Administrative Controls section 6.5.2. The licensing requirement will be deleted. No technical changes are proposed.

Index		
Index Pages	N/A	Page XVI will be changed to reflect the new order in Section 6.0. Page XVII will be deleted. No technical changes are proposed.

Section 1.0 Definitions		
1.33	N/A	The definition of Core Operating Limits Report (COLR) currently states that "cycle-specific core operating limits shall be determined for each reload cycle in accordance with Technical Specification 6.9.5." The requirements related to COLR which are contained in the Administrative Controls section will be moved to Administrative Controls section 6.6.5. Therefore, the reference contained in this definition will be changed to Specification 6.6.5. No technical changes are proposed.

Section 3 / 4. 3 Instrumentation		
3.3.3.1, Radiation Monitoring Instrumentation		
Table 3.3-6, Actions 17 & 20	N/A	Actions 17 and 20 on Table 3.3-6 apply to the Control Room Ventilation Intake Duct Monitors. ANO-1 and ANO-2 share a common control room envelope and ventilation system therefore the specifications for the two units should be the same. The proposed change will modify Actions 17 and 20 to specifically address the required actions when operating in Mode 1, 2, 3, or 4. Additionally both actions will include requirements to place in the unit in Cold Shutdown if the actions cannot be met. This is consistent with the current requirement to enter TS 3.0.3 if the action requirements are not satisfied and therefore, this is not considered a technical change.

Current TS #	New Location	Description of Change
Table 3.3-6, New Action Requirement Action 21	New	This new action is added to make the ANO-1 and ANO-2 TSs for the Control Room Ventilation Intake Duct Monitors similar. The new action is as follows: "During handling of irradiated fuel with one or two channels inoperable, immediately place one OPERABLE CREVS train in the emergency recirculation mode or immediately suspend handling of irradiated fuel." The current actions associated with the handling of irradiated fuel are included in Actions 17 and 20. Neither of these actions allows the suspension of fuel movement as an option.
Table 3.3-6, Actions 18 & 19	N/A	Action 18 applies to the Containment High Range monitors and Action 19 applies to the Main Steam Radiation Monitors. Both actions require that a special report be prepared and submitted to the commission pursuant to Specification 6.9.2. The proposed change deletes the reference to Specification 6.9.2, which will be deleted from the Administrative Controls Section. The requirement to submit a special report will be maintained. No technical changes are proposed.
Table 3.3-10, Actions 3.b and 4.b	N/A	These actions require the submittal of a special report to the NRC when the Reactor Vessel Level Monitoring System is rendered inoperable. A reference to specification 6.9.2 is currently included in these. That reference will be deleted, however the reporting requirement retained. No technical changes are proposed.

3 / 4.4 Reactor Coolant System		
3.4.5 Steam Generators		
4.4.5.0, 4.4.5.1, 4.4.5.2, 4.4.5.3, 4.4.5.4, and Tables 4.4-1 and 4.4-2	6.5.9	These surveillance requirements describe the Steam Generator Tube Surveillance Program. In keeping with NUREG-1432, *"Standard Technical Specifications Combustion Engineering Plants,"* these program requirements will be moved to the Administrative Controls section 6.5.9. The references to surveillance requirements and table numbers will be changed to reflect the relocation into section 6.0. A new SR 4.4.5 will be added to direct performance of steam generator inspections in accordance with the Steam Generator Tube Surveillance Program. No technical changes are proposed.

Current TS #	New Location	Description of Change
4.4.5.5	6.6.7	This surveillance requirement describes the reports that are submitted to the Commission following steam generator inservice and tube inspections. This requirement will be moved to the Administrative Controls section 6.6.7 with only minor changes. Presently 4.4.5.5.c references Specification 6.9.2, which will be deleted. The requirement for the Special Report will be retained in section 6.6.7 however Specification 6.9.2 will not be referenced. References to the table number will be changed to reflect the relocation into section 6.0. No technical changes are proposed.
3.4.5 Bases	N/A	The portion of the bases of TS 3.4.5 that discusses the surveillance requirements for inspection of the steam generator tubes will be modified to support the changes described above.

3 / 4.5 Emergency Core Cooling Systems (ECCS)		
3.5.2 ECCS Subsystems - Tavg ≥ 300 °F 3.5.3 ECCS Subsystems - Tavg ≤ 300 °F		
3.5.2 & 3.5.3	N/A	Action b of both of these specifications references Specification 6.9.2, which will be deleted. The requirement to prepare and submit a report to the Commission will be maintained in the individual actions. No technical changes are proposed.

3 / 4.7 Plant Systems		
3.7.6.1 Control Room Emergency Ventilation and Air Conditioning System		
3.7.6.1	New Action "d"	The Control Room Emergency Ventilation System (CREVS) and Control Room Emergency Air Conditioning System (CREACS) are shared systems between ANO-1 and ANO-2. ANO-1 ITS 3.7.9 action B describes what is required in Modes 1, 2, 3, or 4 when the control room boundary becomes inoperable. A similar action is not included in the ANO-2 TSs. For consistency a new action "d" will be added for ANO-2.
3.7.6.1, Action d	3.7.6.1, Action e	There are no technical changes proposed to this action. It will be designated as Action e.
3.7.6.1, Action e	3.7.6.1, Action f	There are no technical changes proposed to this action. It will be designated as Action f.
3.7.6.1, Action f	3.7.6.1, Action g	There are no technical changes proposed to this action. This action will be moved to the top of page 3/4 7-18 and is thus marked as a revision to that page. It will be designated as Action g.

Current TS #	New Location	Description of Change
3.7.6.1, Action g	3.7.6.1, Action h	There are no technical changes proposed to this action. This action will be moved to the top of page 3/4 7-18 and is thus marked as a revision to that page. It will be designated as Action h.
4.7.6.1.2.b, 4.7.6.1.2.c, 4.7.6.1.2.d.1, 4.7.6.1.2.e, 4.7.6.1.2.f	6.5.11	The filtration testing program described in these individual SRs will be moved to the Administrative Controls section 6.5.11 to be consistent with ANO-1 as well as NUREG-1432. The testing program (6.5.11) will include the Control Room Emergency Air Filtration System and the Fuel Handling Area Filtration System. A statement of clarity will be added stating that TS 4.0.2 and 4.0.3 are applicable. The current SR 4.7.6.1.2.c wording will be replaced with the following wording: "Perform required Control Room Emergency Ventilation filter testing in accordance with the Ventilation Filter Testing Program (VFTP)." No technical changes are proposed.
4.7.6.1.2.d.2	4.7.6.1.2.b	SR 4.7.6.1.2.d.2 verifies every 18 months that the control room isolates and switches into a recirculation mode of operation within 10 seconds after the receipt of a control room high radiation signal. This SR will be renumbered as SR 4.7.6.1.2.b with no technical changes proposed.

3.7.8 Shock Suppressors (Snubbers)		
4.7.8.h	N/A	This SR refers to Specification 6.10.2, which was deleted from the ANO-2 TSs with Amendment 209. The reference to Specification 6.10.2 was inappropriately left in SR 4.7.8.h. The proposed change will delete the reference to Specification 6.10.2. No technical changes are proposed.

3.7.12 Spent Fuel Pool Structural Integrity		
3.7.12 Action a	N/A	Action a references Specification 6.9.2 which will be deleted. The requirement to prepare and submit a special report will remain as part of Action a. No technical changes are proposed.

Current TS #	New Location	Description of Change
3 / 4.8 Electrical Power Systems		
3.8.1.1 A. C. Sources		
4.8.1.1.2.b	6.5.13	This SR requires that at least once per 92 days a sample of the diesel fuel from the fuel storage tank be obtained and checked for viscosity, water and sediment. The SR will be moved to the Administrative Controls section 6.5.13 as the Diesel Fuel Oil Testing Program. The current SR 4.8.1.1.2.b will be replaced with the following: "By testing the diesel fuel oil in accordance with the Diesel Fuel Oil Testing Program." No technical changes are proposed.

3 / 4.9 Refueling Operations		
3.9.11 Fuel Handling Area Ventilation System		
4.9.11.2	6.5.11	This SR contains the filter testing program for the fuel handling area ventilation system. It will be moved and combined with the control room emergency ventilation filter testing requirements to the Administrative Controls section 6.5.11. The present 4.9.11.2 will be modified to require that the fuel handling area ventilation system be demonstrated operable in accordance with the Ventilation Filter Testing Program. A statement of clarity will be added to the new specification 6.5.11 stating that TS 4.0.2 and 4.0.3 are applicable. No technical changes are proposed.
Bases	N/A	The bases of this TS will be modified.

6.0 Administrative Controls		
6.1 Responsibility		
6.1.1	6.1.1	The responsibilities of the plant manager will remain in the same location with only minor editorial changes. The proposed changes will result in consistent wording between the ANO-1 and ANO-2 specifications. No technical changes are proposed.
6.1.2	6.1.2	The responsibilities associated with the Senior Reactor Operator in the control room will remain in the same location. Only minor editorial changes are proposed, which will result in consistency between ANO-1 and ANO-2. No technical changes are proposed.

Current TS #	New Location	Description of Change
6.2 Organization		
6.2.1 Offsite and Onsite Organizations		
6.2.1	6.2.1	The title will change to "Onsite and Offsite Organizations" reversing the order of the current wording. No technical changes are proposed.
6.2.1.a	6.2.1.a	Only minor editorial changes are proposed, which will result in consistency between ANO-1 and ANO-2. No technical changes are proposed.
6.2.1.b	6.2.1.b	Only minor wording changes are proposed, which will result in consistency between ANO-1 and ANO-2. No technical changes are proposed.
6.2.1.c	6.2.1.c	Only minor editorial changes are proposed, which will result in consistency between ANO-1 and ANO-2. No technical changes are proposed.
6.2.1.d	6.2.1.d	Only minor editorial changes are proposed, which will result in consistency between ANO-1 and ANO-2. No technical changes are proposed.

6.2.2 Unit Staff		
6.2.2.a	6.2.2.a & b	The on-duty shift composition currently described in Table 6.2-1 is consistent with the requirements contained in 10 CFR 50.54 with the exception of non-licensed operators. Therefore, this specification will be modified to reflect the requirements are contained in 10 CFR 50.54 (m) (2) (i) and moved to paragraph 6.6.2.b. A new 6.2.2.a will be added that defines the number of and time at which non-licensed operators shall be on site. These changes will result in consistency between ANO-1 and ANO-2.
6.2.2.b	N/A	The current wording in 6.2.2.b requires that a licensed Operator be in the control room when fuel is in the reactor. This paragraph is part of the requirements contained in 10 CFR 50.54(m)(2)(iii). The regulation says that "In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be present at the controls at all times." The specification is redundant to the wording contained in the regulation and will therefore be deleted.
6.2.2.c	N/A	Specification 6.2.2.c requires at least two licensed operators to be in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips. This duplicates 10 CFR 50.54(m)(1) and (m)(2)(iii). Therefore, this specification will be deleted.

Current TS #	New Location	Description of Change
6.2.2.d	6.2.2.d	Specification 6.2.2.d requires an individual qualified in radiation protection procedures to be on site when fuel is in the reactor. This specification will be retained as paragraph 6.2.2.d with the addition of a second sentence with includes the allowance for the position to be vacant up to two hours to accommodate for unexpected absences. The proposed change is consistent with the ANO-1 specification.
6.2.2.e	N/A	Specification 6.2.2.e requires that either a licensed Senior Reactor Operator or a Senior Reactor Operator limited to fuel handling directly supervise all core alterations. This individual can have no collateral duties. This duplicates 10 CFR 50.54 (m) (2) (iv) and will therefore be deleted.
6.2.2.f	6.2.2.g	This specification requires in Modes 1, 2, 3, or 4 that an individual shall be available to provide technical advisory support to the unit shift supervisor. The specification will be retained as specification 6.2.2.g with minor editorial changes proposed, which will result in consistency between ANO-1 and ANO-2.
6.2.2.g	6.2.2.e	This specification requires that an administrative program be established to limit the amount of overtime worked by plant staff performing safety-related functions. This will be retained as Specification 6.2.2.e with only minor editorial changes.
6.2.2.h	6.2.2.f	This specification requires that the operations manager or the assistant operations manager hold a senior reactor operator license. This will be retained as specification 6.2.2.f with only minor editorial changes.
Table 6.2-1	N/A	The requirements reflected in this table are consistent with the requirements in 10 CFR 50.54 (m) (2) (i) for one unit, one control room. Therefore, the table will be deleted and the requirement to comply with 10 CFR 50.54 (m) (2) (i) will be included in new paragraph 6.2.2.b.
Table 6.2-1 * Note	N/A	The * note clarifies that in Mode 5 & 6 a licensed Senior Reactor Operator in addition to the one related to fuel handling must be present. This requirement is contained in 10 CFR 50.54 and will be deleted.

Current TS #	New Location	Description of Change
Table 6.2-1 # note	6.2.2.c	The # note allows for shift crew composition to be less than the minimum requirements for a time period not to exceed 2 hours. This will be retained and moved to specification 6.2.2.c. The new 6.2.2.c will be expanded to include specification 6.2.2.g (technical support).

6.3 Unit Staff Qualifications		
6.3.1	6.3.1	Currently the unit staff has to meet or exceed the minimum qualifications of ANSI N18.1-1971. The section will be revised to reference ANSI ANS 3.1-1978, which will replace ANSI N18.1-1971. ANSI ANS 3.1-1978 is more restrictive than ANSI N18.1-1971. The ANO-2 staff presently meets the requirements set forth in ANSI ANS 3.1-1978.

6.5 Programs		
6.5.8	N/A	This specification details the Inservice Testing Program requirements and will be retained in its present location with no changes proposed.

6.7 Safety Limit Violation		
6.7.1.a	N/A	This specification requires that the unit be placed in at least HOT STANDBY within one hour of violating a Safety Limit. The Safety Limits are included in Section 2.1 of the ANO-2 TSs. Specification 6.7.1.a is duplicated in the action statements of the safety limits, Departure from Nucleate Boiling Ratio, Linear Heat Rate, and Reactor Coolant System pressure, which currently require that if the limits are exceed, the unit be placed in HOT STANDY within one hour. The same action is required by each specification, therefore specification 6.7.1.a will be deleted.
6.7.1.b	N/A	This specification requires that the Vice President, Operations ANO and the SRC be notified within 24 hours of violating a safety limit. This will be deleted.
6.7.1.c	N/A	This specification requires notification of the NRC pursuant to 10 CFR 50.72 and a written report pursuant to 10 CFR 50.36 and specification 6.6 in the event of violating a safety limit. Specification 6.6 was previously deleted in Amendment 209. At that time the reference to specification 6.6 was inappropriately left in specification 6.7.1. Specification 6.7.1.c will be deleted since it is duplicated in the CFR.

Current TS #	New Location	Description of Change
6.8 Procedures and Programs		
6.8.1.a	6.4.1.a	This specification requires that written procedures covering activities referenced in Appendix "A" of Regulatory Guide (RG) 1.33, Revision 2, February 1978 shall be established, implemented, and maintained. This will be retained as specification 6.4.1.a. No technical changes are proposed.
6.8.1.b	N/A	This specification requires that written procedures be established, implemented and maintained covering activities related to refueling operations. Appendix "A" to RG 1.33 requires that procedures be written for refueling operations. This specification is redundant to the current 6.8.1.a and will be deleted.
6.8.1.c	N/A	This specification requires that written procedures be established, implemented and maintained covering surveillance and test activities of safety related equipment. Appendix "A" to RG 1.33 requires this. Therefore, this specification is redundant to the current 6.8.1.a and will be deleted.
6.8.1.f	6.4.1.c	This specification requires the implementation of a fire protection program. It will be retained as 6.4.1.c. No technical changes are proposed.
6.8.1.g	6.4.1.e	This specification describes the procedures needed for the Core Protection Calculator Addressable Constants and software. This will be retained as specification 6.4.1.e with only minor editorial changes proposed.
6.8.1.h	N/A	This specification requires that written procedures shall be established, implemented and maintained covering new and spent fuel storage. The procedures related to these activities will be maintained as required by RG 1.33. This specification is redundant to the current 6.8.1.a and will be deleted.

Current TS #	New Location	Description of Change
6.8.1.i	6.4.1.d	This specification requires that written procedures shall be established, implemented and maintained related to the Offsite Dose Calculation Manual (ODCM) and Process Control Program (PCP). These programs will be included in Section 6.5. Written procedures will be maintained for these programs as well as the other programs included in the proposed section 6.5. The requirement to maintain procedures for the programs will become the new 6.4.1.d.
6.8.4.a	6.5.4	This specification requires that a Radioactive Effluent Controls Program shall be established, implemented, and maintained. This will be retained as specification 6.5.4. The proposed changes include renumbering of the subparagraphs from numerical to alpha characters and adding clarity that TS 4.0.2 and 4.0.3 are applicable. No technical changes are proposed.
6.8.4.b	6.5.5	This specification requires that a Component Cyclic or Transient Limit Program shall be established, implemented, and maintained. This will be retained as specification 6.5.5. No technical changes are proposed.

6.9 Reporting Requirements		
6.9.1	N/A	This specification reminds the licensee that reporting requirements are contained in 10 CFR and that additional reports are required to those specified in the 10 CFR. This will be deleted. The specific reporting requirements in the subsequent section will be listed as stand alone specifications, relocated or deleted.

Current TS #	New Location	Description of Change
Startup Report		
6.9.1.1, 6.9.1.2 and 6.9.1.3	N/A	These specifications require that a summary report of plant startups and power escalation testing be submitted following four different activities. In addition the contents and timing of that submittal are defined in specifications 6.9.1.2 and 6.9.1.3, respectively. Regulatory Guide (RG) 1.16 *"Reporting of Operating Information - - Appendix A Technical Specifications"* addressed the summary report. ANO-2 is committed to this regulatory guide in Safety Analysis Report (SAR) section 1.3.3. Currently the wording in the SAR states that "conformance with this guide is address in the Technical Specifications since the guide covers only reporting requirements." These specifications will be deleted and the wording in the SAR will be changed to remove reference to the TSs. Entergy will continue to comply with the RG.

Annual Reports		
6.9.1.4	N/A	This specification describes the annual reports that will be submitted to the NRC prior to March 1 of each year for specific activities covering the previous calendar year. The specific reports are listed in section 6.9.1.5 and are addressed below. Specification 6.9.1.4 will be deleted. The schedule for submittal of annual reports is contained in RG 1.16 or will be specifically referenced in the individual specification. The appropriate changes to ANO's conformance with the RG will be made as described above.
6.9.1.5.a	6.6.1	This specification requires that a tabulation of personnel exposures greater than 100 mrem/year be reported. This will be retained as specification 6.6.1. No technical changes are proposed, however, a submittal date of no later than April 30 has been added. This change is considered administrative since no relaxation of the reporting requirements is proposed.
6.9.1.5.b	6.6.7	This specification requires that the results of the steam generator tube inservice inspection be reported. This will be retained and combined with SR 4.4.5.5, and relocated to specification 6.6.7.

Current TS #	New Location	Description of Change
6.9.1.5.c	N/A	This specification requires documentation of all challenges to pressurizer safety valves. It was added as a result of recommendations articulated in Appendix C.2 (Item C.3.3) of NUREG-0660. Volume 1, *"NRC Action Plan Developed As A Result of the TMI-2 Accident."* It will be deleted. See section 4.0 for justification.
6.9.1.5.e	6.6.8	This specification requires reporting annually the analysis results of primary coolant specific activity if the limits of Specification 3.4.8 are exceeded. It will be retained and relocated to 6.6.8 with no technical changes proposed.
6.9.1.6	6.6.4	This specification requires that a monthly operating report be submitted. The requirement will be retained as 6.6.4. No technical changes are proposed.
6.9.2.a	3.5.2 & 3.5.3	This specification requires that a special report be submitted to the NRC whenever an actuation of the emergency core cooling system (ECCS) occurs. Specification 6.9.2.a will be deleted however, the reporting requirement will be retained in the individual ECCS specification (i.e. the proposed wording of the individual specifications will describe the reporting requirement.) No technical changes are proposed.
6.9.2.i	Table 3.3-6, Actions 18 & 19	This specification requires that a special report be submitted to the NRC whenever the containment radiation monitors are rendered inoperable. Specification 6.9.2.i will be deleted and the reporting criterion maintained in the individual containment radiation monitor specification where it is presently referenced. The wording in the actions will be changed to reflect the reporting requirements instead of the reference to 6.9.2. No technical changes are proposed.
6.9.2.j	Table 6.5.9-2 and 6.6.7.c	This specification requires that a special report be submitted to the NRC whenever a steam generator inspection results in category C-3 findings. Specification 6.9.2.j will be deleted. However, this report is retained in new table 6.5.9-2 and specification 6.6.7.c. No technical changes are proposed.

Current TS #	New Location	Description of Change
6.9.2.k	3.7.1.2, Action a	This specification requires that a special report be submitted to the NRC whenever the structural integrity of the spent fuel pool does not conform to specification 3.7.12. Specification 6.9.2.k will be deleted. However, the reporting requirement is retained in action "a" of specification 3.7.12. No technical changes are proposed.
6.9.2.n	3.3.3.6 Table 3.3-10 Action 3.b and 4.b	This specification requires that a special report be submitted whenever the Reactor Vessel Level Monitoring System is inoperable. Specification 3.3.3.5, Table 3.3-10, Action 3.b and 4.b currently reference specification 6.9.2. The wording in these actions will be modified to reflect the reporting criteria. Specification 6.9.2.n will be deleted. No technical changes are proposed.
6.9.2.o	3.3.3.1, Table 3.3-6 Action 19	This specification requires submittal of a special report whenever the main steam line radiation monitor is rendered inoperable. The special reporting requirement is currently referenced in specification 3.3.3.1, Table 3.3-6, Action 19. The wording in Action 19 will be modified to reflect the requirement for the special report. No technical changes are proposed.
6.9.3	6.6.3	This specification requires a radioactive effluent release report. The reporting requirement will be retained and relocated to specification 6.6.3. Only minor editorial changes and the addition of a submittal date are proposed.
6.9.4	6.6.2	This specification requires submittal of an annual radiological environmental operating report. This report will be retained and relocated to specification 6.6.2. Only minor editorial changes are proposed.
6.9.5	6.6.5.a	This specification requires submittal of the Core Operating Limits Report (COLR). The report will be retained and relocated to specification 6.6.5. Only minor editorial changes are proposed.
6.9.5.1	6.6.5.b	This specification includes a list of the analytical methods used to determine the core operating limits. The list will be retained as part of specification 6.6.5. Only minor editorial changes are proposed.
6.9.5.2	6.6.5.c	This specification requires determination that all applicable limits of the safety analysis are met. The specification will be retained and relocated to specification 6.6.5.c. No technical changes are proposed.

Current TS #	New Location	Description of Change
6.9.5.3	6.6.5.d	This specification delineates to whom the COLR shall be distributed. The specification will be retained and relocated to specification 6.6.5.d. Only minor editorial changes are proposed.

6.11 Radiation Protection Program		
6.11	N/A	This specification requires that procedures for personnel radiation protection are prepared, approved, maintained, and adhered to. This specification will be deleted. The procedures related to these activities will be maintained as required by RG 1.33. Entergy will maintain written procedures in accordance with RG 1.33 as reflected in the current 6.8.1.a and the proposed 6.4.1.a. Specification 6.11 will be deleted.

6.13 High Radiation Area		
6.13.1	6.7.1	This specification requires that high radiation areas shall be barricaded and conspicuously posted as high radiation areas when the intensity of radiation is less than 1000 mrem/hr. The specification will be retained and relocated to 6.7.1. The current reference to paragraphs 20.203(c)(2) and 20.202(b)(3) will be changed to 20.1601 (a) and (b), Paragraphs 20.202 and 20.203 were replaced with 20.1601. More detail will be added to be consistent with the approved ANO-1 wording of the same specification. This change updates the CFR reference to the current CFR and is editorial only.
6.13.2	6.7.2	This specification requires that high radiation area in which the intensity of radiation is greater than 1000 mrem/hr be locked. This specification will be retained and relocated to 6.7.2. More detail will be added which will result in consistency with the approved ANO-1 wording.

6.14 Offsite Dose Calculation Manual (ODCM)		
6.14	6.5.1	This specification describes the ODCM. This specification will be retained and relocated to 6.5.1. Only minor editorial changes are proposed.

6.15 Containment Leakage Rate Testing Program		
6.15	6.5.16	This specification describes the program requirements related to containment leakage rate testing. The program will be relocated, without any technical changes, to specification 6.5.16

In addition to the above changes, Entergy proposes to add a TS Bases Control Program to the Administrative Controls Section as section 6.5.14. This program will provide a means for processing changes to the Bases of the TSs under appropriate administrative controls and reviews. The addition of this program is consistent with NUREG-1432 and the approved ANO-1 conversion to the improved TSs.

Entergy also proposes to add a requirement to establish, implement and maintain emergency operating procedures to the Administrative Controls Section as section 6.4.1.b. This is consistent with NUREG-1432 and the approved ANO-1 conversion to the proposed TSs.

A proposed change (reference 4) to the ANO-2 TSs to relocate the Reactor Coolant Pump Flywheel Inspection from SR 4.4.10.1 to Section 6.0 is currently in NRC staff review. As a placeholder for this change Section 6.5.7 has been designated as "later."

Specification 6.6.6 in the new Administrative Controls section will not be used and in the proposed change is designated as "Not Used."

3.0 BACKGROUND

The proposed change relocates several surveillances from the Surveillance Requirements section of Technical Specifications (TSs) to the Administrative Controls section of TSs without any technical change to these requirements. In addition, the proposed change re-orders the Administrative Controls section of the TSs in an effort to establish consistency between the approved ANO-1 conversion to the improved TSs and the ANO-2 Administrative Controls section. The change also results in the Administrative Controls section being similar to the order contained in the *"Standard Technical Specifications Combustion Engineering Plants"* (NUREG-1432).

The proposed changes to the system related TSs, Control Room Ventilation Intake Duct Monitors and Control Room Emergency Ventilation System, provide clarity and add new action statements. These changes do not result in any modifications to the plant and continue to meet the plant design basis.

The proposed deletion of several of the specifications contained in the Administrative Controls section has no impact on the plant design. In most cases these administrative controls are redundant to current regulatory requirements and will continue to be performed as required by the associated 10 CFR requirement. The specifications included in the Administrative Controls section do not contain requirements that affect the plant design as defined by 10 CFR 50.36, *"Technical Specifications."*

4.0 TECHNICAL ANALYSIS

Section 2.0, Proposed Changes, provided a detailed listing of proposed changes to the Technical Specifications. No further discussion will be provided for the administrative changes described in Section 2.0 as having no technical changes. Only the proposed changes that are considered technical will be discussed in this section.

Process Monitors located in the Control Room Ventilation System Intake Ducts

The purpose of the Control Room Ventilation System is to provide heating, ventilation, and air conditioning to ensure a suitable environment for equipment and station operator comfort and safety. The purpose of the process monitors is to isolate the Control Room by automatically placing the ventilation in the recirculation mode of operation following an uncontrolled release of radioactivity. Reactor operation may continue indefinitely with the control room emergency ventilation system (CREVS) in the emergency recirculation mode of operation. The following changes do not modify the system in any way and thus the intended function of the system is maintained.

TS Table 3.3-6, Action 21 will be added as a new action associated with the process monitors located in control room ventilation system intake ducts. Actions associated with these process monitors are currently included in Actions 17 and 20 and are applicable during Modes 1, 2, 3, 4, and during handling of irradiated fuel. The new Action 21 will be applicable during handling of irradiated fuels and the current Actions 17 and 20 will be applicable during Modes 1, 2, 3, and 4.

Currently, Action 17 states the following:

> "With no channels OPERABLE, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation."

This action will be retained however will be applicable only during Modes 1, 2, 3 and 4. The action will be modified to direct that the unit be placed in HOT STANDBY within the 6 hours and COLD SHUTDOWN within the following 30 hours if the CREVS cannot be maintained in the recirculation mode of operation within 1 hour of both channels becoming inoperable. This is consistent with the current practice, which would require entry into TS 3.0.3 if the CREVS could not be maintained in the recirculation mode of operation. The new action will read as follows:

> "In MODE 1, 2, 3, or 4 with no channels OPERABLE, within 1 hour initiate and maintain operation of the control room emergency ventilation system (CREVS) in the recirculation mode of operation or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN in the following 30 hours."

The current Action 20 states the following:

> "With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 7 days, or within the next 6 hours initiate and maintain the control room emergency ventilation system in the recirculation mode of operation."

The wording in this action will be modified to be applicable in Modes 1, 2, 3, and 4 only. The additional six-hour allowance to place an operable CREVS in the recirculation mode of operation will be deleted to be consistent with the ANO-1 TSs. The action will be modified to direct that the unit be placed in HOT STANDBY within the 6 hours and COLD SHUTDOWN within the following 30 hours if the CREVS cannot be maintained in the recirculation mode of operation. The new action will read as follows:

> "In Mode 1, 2, 3, or 4 with the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to

OPERABLE status within 7 days, or initiate and maintain the control room emergency ventilation system in the recirculation mode of operation, or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN in the following 30 hours."

The actions applicable during handling of irradiated fuel will be contained in new action 21, which will state the following:

"During handling of irradiated fuel with one or two channels inoperable, immediately place one OPERABLE CREVS train in the emergency recirculation mode or immediately suspend handling of irradiated fuel."

The new action 21 allows suspension of fuel handling activities as a viable option. When the CREVS is placed in the recirculation mode of operation, a small quantity of filtered outside air is drawn into the control room while the normal supply and return ducts are isolated. This assures protection to the control room operator in the event of a fuel handling accident and subsequent airborne activity. This action is consistent with ANO-1 improved TS 3.3.16, Action D.

TS 3.7.6.1, Control Room Emergency Ventilation And Air Conditioning System

A new Action "d" will be added to TS 3.7.6.1. The new action describes what is required in Modes 1, 2, 3, or 4 when the control room boundary becomes inoperable. ANO-2 does not currently have an action that addresses the failure of the control room boundary. The new action would allow 24 hours to restore the boundary before a shutdown is required. This is consistent with the approved ANO-1 TS, which will be implemented in the summer of 2002.

TS Section 6.2.2, Unit Staff

Several changes are proposed for this section. 10 CFR 50.54, "*Conditions of Licenses*" includes several requirements for staffing. The current Technical Specifications are redundant to many of the requirements contained in 10 CFR 50.54 and therefore will be deleted.

The current 6.2.2.a requires that each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1. During Modes 1, 2, 3, and 4 this table requires two Senior Licensed Operators, two Reactor Licensed Operators, and three non-licensed operators. The table also specifies manning for Modes 5 and 6 which includes one Senior Licensed Operator who is not supervising Fueling Handling, one Reactor Licensed Operator, and 1 non-licensed operator. The requirements of Table 6.2-1 are consistent with the requirements contained in the table included in 10 CFR 50.54 (m)(2)(i) for one unit, one control room with the exception of the non-licensed operators. The current specification 6.2.2.a will be modified to include the requirements for non-licensed operators only and maintain the current staffing requirements. The specification to maintain the staffing requirements for licensed operators will become 6.2.2.b and reference 10 CFR 50.54 (m)(2)(i) for one unit, one control room.

The current TS 6.2.2.b requires that at least one licensed Operator shall be in the control room when fuel is in the reactor. This is consistent with the requirements of 10 CFR 50.54(m)(2)(iii) and therefore will be deleted since it is redundant to the regulations.

The current TS 6.2.2.c requires that at least two licensed Operators be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips.

10 CFR 50.54(m)(2)(iii) requires that a senior licensed operator be present in the control room during operational modes other than cold shutdown or refueling. It also requires that a licensed or senior licensed operator be at the controls at all times in addition to the senior licensed operator specified. Reactor startups, scheduled reactor shutdowns and recovery from reactor trips occur during operational modes other than cold shutdown. Therefore, the requirements contained in the TS are redundant to those contained in 10 CFR 50.54(m)(2)(iii) and will therefore be deleted.

The current TS 6.2.2.d requires that an individual qualified in radiation protection will be on site when fuel is in the reactor. This specification will remain in its present location. Table 6.2-1, which reflects Operations staffing composition, includes an allowance for unexpected absences for a time period not to exceed two hours. This allowance for unexpected absences will be added to TS 6.2.2.d. The addition of this allowance is consistent with proposed changes to the ANO Emergency Plan (reference 1).

The current TS 6.2.2.e requires that a licensed senior reactor operator or senior reactor operator limited to fuel handling who has no other concurrent responsibilities shall directly supervise core alterations. This requirement is redundant to 10 CFR 50.54 (m)(iv) and will therefore be deleted.

The * note associated with Table 6.2-1 will be deleted. This note is associated with senior licensed operators and states that the required number of senior licensed operators in Mode 5 and 6 does not include the senior licensed operator who is supervising core alterations. 10 CFR 50.54 (m)(iv) states that a senior licensed operator must be limited to fuel handling to directly supervise the activity. The minimum requirements for shift listed in the Table referenced by 10 CFR 50.54(m)(2)(i) require one senior licensed operator. Therefore, the note is included in the combination of these 10 CFR 50.54 requirements and will be deleted.

The # note associated with Table 6.2-1 will be relocated to specification 6.2.2.c and expanded to include reference to 10 CFR 50.54 (m)(2)(i). This is consistent with the change proposed to specification 6.2.2.a. In addition a reference to the new 6.2.2.g will be added. This is consistent with the changes proposed to the Emergency Plan (reference 1).

Section 6.3, Unit Staff Qualifications

The current 6.3.1 requires that the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971. ANSI ANS 3.1-1978 exceeds the qualification requirements included in ANSI N18.1-1971. ANO-2 currently meets the requirements contained in ANSI ANS 3.1-1978. Therefore this reference will be changed to ANSI ANS 3.1-1978. Entergy's Quality Assurance Program Manual (QAPM) states: "Qualification requirements for personnel will meet ANSI/ANS 3.1-1978 except where exception to ANSI N18.1 or to this Standard is identified in the applicable unit's Technical Specifications." Therefore the proposed change does not render the QAPM inaccurate.

Specification 6.7, Safety Limit Violation

This section describes the actions required if a safety limit is violated. The safety limits for ANO-2, departure from nucleate boiling ratio, peak linear heat rate, and reactor coolant system pressure, are included in Section 2.1 of the TSs. If the defined limit for any of these is

exceeded, the unit must be placed in HOT STANDBY within 1 hour. This is consistent with the requirements stated in TS 6.7.1.a. Therefore, TS 6.7.1.a will be deleted.

Specification 6.7.1.b requires that the Vice President, Operations ANO and the SRC be notified within 24 hours of violating a safety limit. This notification is administratively controlled as part of the ANO corrective action process. This notification will be deleted. This notification is not required to ensure any of the four criteria listed in 10 CFR 50.36. The administrative controls section of Technical Specifications is described in 10 CFR 50.36 as reporting what is necessary to assure operation of the facility in a safe manner. Although this notification will continue to be performed as part of the standard practices for notification, it does not assure the facility is operated in a safe manner. Actions taken in the control room by the control room operators assure the safety of the facility.

Specification 6.7.1.c requires that the Nuclear Regulatory Commission (NRC) be notified in the event of a safety limit violation. 10 CFR 50.36, 10 CFR 50.72 and 10 CFR 50.73 require verbal and written notification to the NRC if a plant shutdown is required by the unit's Technical Specifications. TS 6.7.1.c will be deleted since it is redundant to the requirements contained in these regulations.

6.8. Procedures and Programs

Specification 6.8.1.a requires that written procedures be established as recommended in Appendix "A" of Regulatory Guide (RG) 1.33, Revision 2, February 1978. Specifications 6.8.1.b, c and h list specific types of written procedures that shall be established, implemented, and maintained. The types of procedures listed in these specifications are procedures covering refueling operations, surveillance and test activities on safety related equipment, and new and spent fuel storage, respectively. RG 1.33 includes a list of the types of procedures that should be established. Each of these is included in the RG and therefore 6.8.1.b, c and h will be deleted.

6.9. Reporting Requirements

Specification 6.9.1 provides an introduction into this section of reporting requirements. There are no specific reports included in it other than a reference to the reporting requirements of Title 10, Code of Federal Regulations. Since Entergy is required to complete the reports contained in the Title 10, the wording in this specification is redundant and will be deleted.

Specification 6.9.1.1, 6.9.1.2, and 6.9.1.3 describe reporting requirements related to startup reports. These will be deleted from the Technical Specifications based on the following discussion. Regulatory Guide 1.16, *"Reporting of Operations Information – Appendix A Technical Specifications"* includes guidance related to the contents of the startup reports in section C.1.a. The guidance contained in the ANO-2 TSs is consistent with the guidance contained in RG 1.16. ANO-2 is committed to this RG as is reflected in the ANO-2 Safety Analysis Report (SAR) section 1.3.3. The wording currently states: "Conformance with this guide is addressed in the Technical Specifications since the guide covers only reporting requirements." These words will be changed based on the approval of the proposed change.

Specification 6.9.1.4 provides an introductory paragraph for the list of annual reports and specifies the submittal date of the report. This paragraph will be deleted and the submittal date included in the individual specifications related to annual reports.

Specification 6.9.1.5.a describes the information that should be included in the personnel exposure report. The reporting requirements will be retained and relocated to TS 6.6.1. The submittal date however is being revised to April 30 to be consistent with ANO-1 and the submittal date included in 10 CFR 20.2206, *"Reports of Individual Monitoring."* This change is considered administrative since no relaxation of the reporting requirement is proposed. Additionally note 2, which is referenced by TS 6.9.1.5.a, will be moved into the specification and modified to reflect that this tabulation supplements the requirements of 10 CFR 20.2206. The current reference to paragraph 20.407 of 10 CFR Part 20 is no longer accurate, as there is no such paragraph in 10 CFR Part 20.

Specification 6.9.1.5.c requires that all challenges to the pressurizer safety valves be documented and included in an annual report. In 1997 with the issuance of ANO-2 TS Amendment 180 (reference 3), which added the Low Temperature Overpressure Protection (LTOP) requirements, ANO committed include within the report of challenges to the pressurizer safety valves a report of any challenges to the LTOP valves. This commitment is being deleted as part of this proposed change. NUREG-1432 previously contained the requirement to report challenges of pressurizer safety valves as part of the monthly operating report. Technical Specification Traveler number 258 removed this monthly reporting requirement based on Generic Letter 97-02, *"Revised Content of Monthly Operating Report"* and discussions related to the NRC Performance Indicator Program. The conclusion was that this information was not needed in the assessment of NRC Performance Indicators and as such the requirement to include information related to challenges of the pressurizer safety valves in the monthly operating report was not needed. The NUREG does not require reporting pressurizer safety valve challenges annually. Although the NUREG previously required a monthly report of any pressurizer safety valve challenges, Entergy took exception to the monthly reporting requirement in a February 1999 request for additional information related to the administrative controls of the ANO-1 and ANO-2 TSs (reference 2). Entergy continued to require the annual report. It is proposed that this reporting requirement be deleted. The reason for deletion is consistent with the logic used in the above referenced traveler even though the reporting frequencies differ.

Specification 6.11 requires that procedures for personnel radiation protection shall be prepared consistent with 10 CFR 20 requirements. RG 1.33 requires that procedures for control of radioactivity be developed, including those, which limit personnel exposure. Therefore, this specification is redundant to the requirements contained in RG 1.33 and will be deleted.

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met. As has been discussed in Section 4.0 of this request, many of the proposed changes will result in current requirements being deleted, which are redundant to existing regulatory requirements. The following regulations are referenced and will remain satisfied: 10 CFR 50.54, *"Conditions of Licenses,"* 10 CFR 50.36, *"Technical Specifications,"* 10 CFR 50.72, *"Immediate Notification Requirements for Operating Nuclear Power Reactors,"* 10 CFR 50.73, *"Licensee Event Report System,"* and 10 CFR20.2206, *"Reports of Individual Monitoring."* Regulatory Guide (RG) 1.33, *"Quality Assurance Program Requirements*

(Operation)" and RG 1.16, *"Reporting Of Operating Information – Appendix A Technical Specifications"* as well as Generic Letter (GL) 97-02, *"Revised Content of Monthly Operating Report"* are also included in the discussion of the proposed change.

Entergy has determined that the proposed changes 1) do not require any exemptions or relief from the regulatory requirements, other than the TS, and 2) do not affect conformance with any General Design Criteria differently than described in the Safety Analysis Report.

5.2 No Significant Hazards Consideration

Entergy Operations, Inc. (Entergy) proposes to modify the Arkansas Nuclear One, Unit 2 (ANO-2) Technical Specifications (TSs) to re-order the Administrative Controls section to be consistent with NUREG-1432, *"Standard Technical Specifications Combustion Engineering Plants"* and the ANO, Unit 1 (ANO-1) TS Administrative Controls section. This change will result in moving several surveillance requirements currently contained in the Surveillance Requirements section of the ANO-2 TSs to the Administrative Controls section. The change will also result in the deletion of several specifications currently contained in the Administrative Controls section. For the majority of those that will be deleted, the requirements are presently contained in either the Code of Federal Regulations, referenced Generic Letters, or referenced Regulatory Guides. Entergy will continue to maintain the requirements contained in these regulatory documents. Entergy also proposes to add a Technical Specification Bases Control Program as part of the requested changes.

The actions related to the Control Room Ventilation System will be modified as part of the proposed change. The ventilation system (emergency and air conditioning system) for the control room is shared with ANO-1 and thus the specifications for this system are maintained consistent between the units where appropriate. Recently, ANO-1 received approval of a submittal that resulted in the conversion their custom TSs to the format of NUREG-1430, *"Standard Technical Specifications Babcock and Wilcox Plants."* Included in this conversion were changes to the actions related to the process radiation monitors located in the Control Room Ventilation System intake ducts as well as the specifications for the Control Room Emergency Ventilation System. It is proposed to add two new actions to the ANO-2 TSs one associated with the process monitors and the other with the Control Room Emergency Ventilation System. This results in consistency between the two units specifications.

Entergy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, *"Issuance of Amendment,"* as described below.

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

   Response: No.

   The proposed change modifies the Administrative Controls section of the ANO-2 TSs to be consistent with NUREG-1432. 10 CFR 50.36, *"Technical Specifications"* defines the Administrative Controls section as follows: "Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." Therefore, by definition the specifications contained in the Administrative Controls section are not

specifications related to systems that are used to mitigate any types of accidents. The proposed changes to the Administrative Controls section therefore do not impact the ability of a plant system to perform its intended function.

The proposed changes to the Control Room Ventilation System specifications do not result in any type of plant modification to this system. The system's intended function is to provide heating, ventilation, and air conditioning to ensure a suitable environment for equipment and station operator comfort and safety.

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

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

Response: No.

The proposed change will re-organize the ANO-2 Administrative Controls section and modify the actions related to the Control Room Ventilation System. The changes to the Administrative Controls section by definition of the type of specifications, which are included in the Administrative Controls section, will not create any new or different types of accidents.

The modifications to the Control Room Ventilation System specifications result in providing clarity to existing actions and the addition of new actions. The addition of the new actions results in consistency between the ANO-1 and ANO-2 TSs. No design changes are proposed to the Control Room Ventilation System.

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

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

Response: No.

The proposed changes result in the relocation of several surveillance requirements to the Administrative Controls section as well as the re-organization of the Administrative Controls Section of the ANO-2 TSs. In addition, clarification is added to the Control Room Ventilation System action statements that result in consistency between the ANO-1 and ANO-2 TSs. These changes do not affect the margin of safety.

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

Based on the above, Entergy concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

ATTACHMENT 2

2CAN010203

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

(i) Containment Radiation Monitor

AP&L shall, prior to July 31, 1980 submit for Commission review and approval documentation which establishes the adequacy of the qualifications of the containment radiation monitors located inside the containment and shall complete the installation and testing of these instruments to demonstrate that they meet the operability requirements of Technical Specification No. 3.3.3.6.

2.C.(3)(j) Deleted per Amendment 7, 12/1/78.

2.C.(3)(k) Deleted per Amendment 12, 6/12/79 and Amendment No. 31, 5/12/82.

2.C.(3)(l) Deleted per Amendment 24, 6/19/81.

2.C.(3)(m) Deleted per Amendment 12, 6/12/79.

2.C.(3)(n) Deleted per Amendment 7, 12/1/78.

2.C.(3)(o) Deleted per Amendment 7, 12/1/78.

2.C.(3)(p) Deleted

~~p) Secondary Water Chemistry Monitoring~~

~~EOI shall implement a secondary water chemistry monitoring program using the overall plant administrative procedure "Steam Generator Water Chemistry Monitoring, Unit II", to minimize steam generator tube degradation. The program shall be defined in specific plant procedures and shall include:~~

~~1. Identification of sampling schedule for the critical parameters and control points for these parameters;~~

~~2. Identification of the procedures used to measure the values of the critical parameters;~~

~~3. Identification of process sampling points;~~

~~4. Procedure for the recording and management of data;~~

~~5. Procedures defining corrective actions for off control point chemistry conditions; and~~

~~6. A procedure identifying the authority responsible for the interpretation of the data, and the sequence and timing of administrative events required to initiate corrective action.~~

2.C.(4) (Number has never been used.)

2.C.(5) Deleted

~~(5) EOI shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following.~~

~~1. Provisions establishing preventative maintenance and periodic visual inspection requirements, and~~

~~2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.~~

(6) EOI shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

2.C.(7) Deleted per Amendment 78, 7/22/86.

(8) Antitrust Conditions

EOI shall not market or broker power or energy from Arkansas Nuclear One, Unit 2. Entergy Arkansas, Inc. is responsible and accountable for the actions of its agents to the extent said agent's actions affect the marketing or brokering of power or energy from ANO, Unit 2.

(9) Rod Average Fuel Burnup

Entergy Operations is authorized to operate the facility with an individual rod average fuel burnup (burnup averaged over the length of a fuel rod) not to exceed 60 megawatt-days/kilogram of uranium.

D. Physical Protection

EOI 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 plan, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Arkansas Nuclear One Industrial Security Plan," with revisions submitted through August 4, 1995. The Industrial Security Plan also includes the requirements for guard training and qualification in Appendix A of the safeguards contingency events in Chapter 7. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

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EXCLUSION AREA

1.31 The EXCLUSION AREA is that area surrounding ANO within a minimum radius of .65 miles of the reactor buildings and controlled to the extent necessary by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

UNRESTRICTED AREA

1.32 An UNRESTRICTED AREA shall be any area at or beyond the exclusion area boundary.

CORE OPERATING LIMITS REPORT

1.33 The CORE OPERATING LIMITS REPORT is the ANO-2 specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Technical Specification 6.6.5~~6.9.5~~ Plant operation within these operating limits is addressed in individual specifications.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Pool Area Monitor	1	Note 1	$\leq 1.5 \times 10^{-2}$ R/hr	10^{-4} - 10^{1}R/hr	13
b. Containment High Range	2	1, 2, 3, & 4	Not Applicable	1 - 10^{7} R/hr	18
2. PROCESS MONITORS					
a. Containment Purge and Exhaust Isolation	1	5 & 6	$\leq$ 2 x background	10 - 10^{6} cpm	16
b. Control Room Ventilation Intake Duct Monitors	2	Note 2	$\leq$ 2 x background	10 - 10^{6} cpm	17, 20, 21
c. Main Steam Line Radiation Monitors	1/Steam Line	1, 2, 3, & 4	Not Applicable	10^{-1} - 10^{4}mR/hr	19

Note 1 - With fuel in the spent fuel pool or building.
Note 2 - MODES 1, 2, 3, 4, and during handling of irradiated fuel.

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 13 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 16 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, complete the following:

a. If performing CORE ALTERATIONS or moving irradiated fuel within the reactor building, secure the containment purge system or suspend CORE ALTERATIONS and movement of irradiated fuel within the reactor building.

b. If a containment PURGE is in progress, secure the containment purge system.

c. If continuously ventilating, verify the SPING monitor operable or perform the ACTIONS of 3.3.3.9, or secure the containment purge system.

ACTION 17 - In MODE 1, 2, 3, or 4 ~~W~~with no channels OPERABLE, within 1 hour initiate and maintain operation of the control room emergency ventilation system (CREVS) in the recirculation mode of operation or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN in the following 30 hours.

ACTION 18 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, (1) either restore the inoperable channel to OPERABLE status within 7 days or (2) prepare and submit a Special Report to the Commission ~~pursuant to Specification 6.9.2~~ within 30 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status. With both channels inoperable, initiate alternate methods of monitoring the containment radiation level within 72 hours in addition to the actions described above.

ACTION 19 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2) prepare and submit a Special Report to the Commission ~~pursuant to Specification 6.9.2~~ within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 20 - In MODE 1, 2, 3, or 4 ~~W~~with the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 7 days, or ~~within the next 6 hours~~ initiate and maintain the control room emergency ventilation system in the recirculation mode of operation, or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN in the following 30 hours.

ACTION 21 - During handling of irradiated fuel with one or two channels inoperable, immediately place one OPERABLE CREVS train in the emergency recirculation mode or immediately suspend handling of irradiated fuel.

TABLE 3.3-10 (Con't)
POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
13. In Core Thermocouples (Core-Exit Thermocouples)	2/core quadrant	1
14. Reactor Vessel Level Monitoring System (RVLMS)	2	3, 4

Action 1: With the number of OPERABLE post-accident monitoring channels less than required by Table 3.3-10, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

Action 2: With the number of OPERABLE post-accident monitoring channels less than required by Table 3.3-10, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

If only one channel is inoperable and containment entry is required to restore the inoperable channel, the channel need not be restored until the following refueling outage.

Action 3: With the number of OPERABLE channels one less than the minimum number of channels required to be OPERABLE:

a. If repairs are feasible, restore the inoperable channel to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. If repair is not feasible without shutting down, operations may continue and a special report ~~pursuant to specification 6.9.2~~ shall be submitted to the NRC within 30 days following the failure; describing the action taken, the cause of the inoperability, and the plans and schedule for restoring the channel to OPERABLE status during the next scheduled refueling outage.

Action 4: With the number of OPERABLE channels two less than the minimum channels required to be OPERABLE:

a. If repairs are feasible, restore at least one inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. If repair is not feasible without shutting down, operation may continue and a special report ~~pursuant to specification 6.9.2~~ shall be submitted to the NRC within 30 days following the failure; describing the action taken, the cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status during the next scheduled refueling outage.

STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1,2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing Tavg above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5 Perform steam generator inspections in accordance with the Steam Generator Tube Surveillance Program.

~~4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.~~

~~NOTE: The requirements for inservice inspection do not apply during the steam generator replacement outage (2R14).~~

~~4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.~~

~~4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:~~

~~a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.~~

~~b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:~~

SURVEILLANCE REQUIREMENTS (Continued)

~~1. All nonplugged tubes that previously had detectable wall penetrations (>20%).~~

~~2. Tubes in those areas where experience has indicated potential problems.~~

~~3. A tube inspection (pursuant to Specification 4.4.5.4.a.9) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.~~

~~c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial inspection provided:~~

~~1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.~~

~~2. The inspections include those portions of the tubes where imperfections were previously found.~~

~~The result of each sample inspection shall be classified into one to the following three categories:~~

~~Category~~	~~Inspection Results~~
~~C-1~~	~~Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.~~
~~C-2~~	~~One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.~~
~~C-3~~	~~More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.~~

~~Note: In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.~~

SURVEILLANCE REQUIREMENTS (Continued)

~~4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:~~

~~a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.~~

~~b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3.a; the interval may then be extended to a maximum of once per 40 months.~~

~~c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:~~

~~1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2.~~

~~2. A seismic occurrence greater than the Operating Basis Earthquake.~~

~~3. A loss-of coolant accident requiring actuation of the engineered safeguards.~~

~~4. A main steam line or feedwater line break.~~

~~4.4.5.4 Acceptance Criteria~~

~~a. As used in this Specification~~

~~1. Tubing or Tube means that portion of the tube which forms the primary system to secondary system pressure boundary.~~

~~2. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.~~

~~3. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.~~

~~4. Degraded Tube means a tube containing imperfections ≥20% of nominal wall thickness caused by degradation.~~

~~5. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.~~

~~6. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.~~

~~7. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging because it may become unserviceable prior to the next inspection. The plugging limit is equal to 40% of the nominal tube wall thickness.~~

~~8. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.~~

~~9. Tube Inspection means an inspection of the steam generator tube from tube end (cold leg side) to tube end (hot leg side).~~

SURVEILLANCE REQUIREMENTS (Continued)

~~10. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the hydrostatic test and prior to POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.~~

~~b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.~~

~~4.4.5.5 Reports~~

~~a. Following each inservice inspection of steam generator tubes the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.~~

~~b. The complete results of the steam generator tube inservice inspection shall be reported within 12 months following the completion of the inservice inspection. This report shall include:~~

~~1. Number and extent of tubes inspected.~~

~~2. Location and percent of wall-thickness penetration for each indication of an imperfection.~~

~~3. Identification of tubes plugged.~~

~~c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report pursuant to Specification 6.9.2 as denoted by Table 4.4-2. Notification of the Commission will be made prior to resumption of plant operation (i.e., prior to entering Mode 4). The written Special Report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.~~

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	Yes
No. of Steam Generators per Unit	Two
First Inservice Inspection	One
Second & Subsequent Inservice Inspections	One[1]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in the other S.G. Special Report to NRC per Specification 6.9.2	Other S.G. is C-1	None	N/A	N/A
			Other S.G. is C-2	Perform action for C-2 result of second sample	N/A	N/A
			Other S.G. is C-3	Inspect all tubes in the other S.G. and plug defective tubes. Special Report to NRC per Spec. 6.9.2	N/A	N/A

S = 3 2 8 Where n is the number of steam generators inspected during an inspection.
n

ECCS SUBSYSTEMS - $T_{avg} \geq 300°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection pump,

b. One OPERABLE low-pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted ~~to the Commission pursuant to Specification 6.9.2~~ to the NRC within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
2CV-5101	HPSI Hot Leg Injection Isolation	Closed
2CV-5102	HPSI Hot Leg Injection Isolation	Closed
2BS26	RWT Return Line	Open

*With pressurizer pressure ≥ 1700 psia.

ECCS SUBSYSTEMS - $T_{avg} \leq 300°F$

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE high-pressure safety injection pump, and

b. An OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal.

APPLICABILITY: MODES 3* and 4.

ACTION:

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted ~~to the Commission pursuant to Specification 6.9.2~~ to the NRC within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

*With pressurizer pressure < 1700 psia.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION AND AIR CONDITIONING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 Two independent control room emergency ventilation and air conditioning systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, or during handling of irradiated fuel.

ACTION:

MODES 1, 2, 3, and 4

a. With one control room emergency air conditioning system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one control room emergency ventilation system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one control room emergency air conditioning system and one control room emergency ventilation system inoperable, restore the inoperable control room emergency ventilation system to OPERABLE status within 7 days and restore the inoperable control room emergency air conditioning system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two control room emergency ventilation systems inoperable due to an inoperable control room boundary, restore the control room boundary to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

During Handling of Irradiated Fuel

de. With one control room emergency air conditioning system inoperable, restore the inoperable system to OPERABLE status within 30 days or immediately place the OPERABLE system in operation; otherwise, suspend all activities involving the handling of irradiated fuel. The provisions of Specification 3.0.4 are not applicable.

ef. With one control room emergency ventilation system inoperable, restore the inoperable system to OPERABLE status within 7 days or immediately place the control room in the emergency recirc mode of operation; otherwise, suspend all activities involving the handling of irradiated fuel. The provisions of Specification 3.0.4 are not applicable.

~~f. With one control room emergency air conditioning system and one control room emergency ventilation system inoperable:~~

~~1. restore the inoperable control room emergency ventilation system to OPERABLE status within 7 days or immediately place the control room in the emergency recirc mode of operation, and~~

~~2. restore the inoperable control room emergency air conditioning system to OPERABLE status within 30 days or immediately place the OPERABLE system in operation;~~

~~3. otherwise, suspend all activities involving the handling of irradiated fuel.~~

~~4. The provisions of Specification 3.0.4 are not applicable.~~

~~g. With both control room emergency air conditioning systems or both control room emergency ventilation systems inoperable, immediately suspend all activities involving the handling of irradiated fuel.~~

ACTION (Continued)

g. With one control room emergency air conditioning system and one control room emergency ventilation system inoperable:

1. restore the inoperable control room emergency ventilation system to OPERABLE status within 7 days or immediately place the control room in the emergency recirc mode of operation, and

2. restore the inoperable control room emergency air conditioning system to OPERABLE status within 30 days or immediately place the OPERABLE system in operation;

3. otherwise, suspend all activities involving the handling of irradiated fuel.

4. The provisions of Specification 3.0.4 are not applicable.

h. With both control room emergency air conditioning systems or both control room emergency ventilation systems inoperable, immediately suspend all activities involving the handling of irradiated fuel.

SURVEILLANCE REQUIREMENTS ~~(Continued)~~

4.7.6.1.1 Each control room emergency air conditioning system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Starting each unit from the control room, and

2. Verifying that each unit operates for at least 1 hour and maintains the control room air temperature ≤ 84°F D.B.

b. At least once per 18 months by verifying a system flow rate of 9900 cfm ± 10%.

4.7.6.1.2 Each control room emergency air filtration system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

~~b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:~~

~~1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 2000 cfm ± 10%.~~

~~2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory~~

~~Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of:~~

~~a. ≤ 2.5% for 2 inch charcoal adsorber beds, or~~

~~b. ≤ 0.5% for 4 inch charcoal adsorber beds.~~

~~3. Verifying a system flow rate of 2000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.~~

~~c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of:~~

~~1. ≤ 2.5% for 2 inch charcoal adsorber beds, or~~

~~2. ≤ 0.5% for 4 inch charcoal adsorber beds.~~

b. At least once per 18 months by verifying that on a control room high radiation test signal, the system automatically isolates the control room within 10 seconds and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

c. By performing the required Control Room Emergency Ventilation filter testing in accordance with the Ventilation Filter Testing Program (VFTP).

~~SURVEILLANCE REQUIREMENTS (continued)~~

~~d. At least once per 18 months by:~~

~~1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches Water Gauge while operating the system at a flow rate of 2000 cfm ± 10%.~~

~~2. Verifying that on a control room high radiation test signal, the system automatically isolates the control room within 10 seconds and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.~~

~~e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm ± 10%.~~

~~f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm ±10%.~~

If any snubber selected for functional testing either fails to activate or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be evaluated in a manner to ensure their OPERABILITY. This requirement shall be independent of the requirements stated in Specification 4.7.8.d for snubbers not meeting the functional test acceptance criteria.

g. Preservice Testing of Repaired, Replacement and New Snubbers

Preservice operability testing shall be performed on repaired, replacement or new snubbers prior to installation. Testing may be at the manufacturer's facility. The testing shall verify the functional test acceptance criteria in 4.7.8.e.

In addition, a preservice inspection shall be performed on each repaired, replacement or new snubber and shall verify that:

1) There are no visible signs of damage or impaired operability as a result of storage, handling or installation;

2) The snubber load rating, location, orientation, position setting and configuration (attachment, extensions, etc.), are in accordance with design;

3) Adequate swing clearance is provided to allow snubber movement;

4) If applicable, fluid is at the recommended level and fluid is not leaking from the snubber system;

5) Structural connections such as pins, bearings, studs, fasteners and other connecting hardware such as lock nuts, tabs, wire, and cotter pins are installed correctly.

h. Snubber Seal Replacement Program

The seal service life of hydraulic snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The expected service life for the various seals, seal materials, and applications shall be determined and established based on engineering information and the seals shall be replaced so that the expected service life will not be exceeded during a period when the snubber is required to be OPERABLE. The seal replacement shall be documented ~~and the documentation shall be retained in accordance with Specification 6.10.2~~.

3/4.7.12 SPENT FUEL POOL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.7.12 The structural integrity of the spent fuel pool shall be maintained in accordance with Specification 4.7.12.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

a. With the structural integrity of the spent fuel pool not conforming to the above requirements, in lieu of any other report, prepare and submit a Special Report ~~to the Commission pursuant to Specification 6.9.2~~ to the NRC within 30 days of a determination of such non-conformity.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.12.1 Inspection Frequencies - The structural integrity of the spent fuel pool shall be determined per the acceptance criteria of Specification 4.7.12.2 at the following frequencies:

a. At least once per 92 days after the pool is filled with water. If no abnormal degradation or other indications of structural distress are detected during five consecutive inspections, the inspection interval may be extended to at least once per 5 years.

b. Within 24 hours following any seismic event which actuates or should have actuated the seismic monitoring instrumentation.

4.7.12.2 Acceptance Criteria - The structural integrity of the spent fuel pool shall be determined by a visual inspection of at least the interior and exterior surfaces of the pool, the struts in the tilt pit, the surfaces of the separation walls, and the structural slabs adjoining the pool walls. This visual inspection shall verify no changes in the concrete crack patterns, no abnormal degradation or other signs of structural distress (i.e, cracks, bulges, out of plumbness, leakage, discolorations, efflorescence, etc.).

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown be transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE: (Note 1)

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day fuel tank.

2. Verifying the fuel level in the fuel storage tank.

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

4. Verifying the diesel starts from a standby condition and accelerates to at least 900 rpm in ≤ 15 seconds. (Note 2)

5. Verifying the generator is synchronized, loaded to an indicated 2600 to 2850 Kw and operates for ≥ 60 minutes. (Notes 3 & 4)

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. By testing the diesel fuel oil in accordance with the Diesel Fuel Oil Testing Program. ~~At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank obtained in accordance with ASTM D270-65, is within the acceptable limits specified in Table 1 of ASTM D975-74 when checked for viscosity, water and sediment.~~

Note 1
All planned diesel generator starts for the purposes of these surveillances may be preceded by prelube procedures.

Note 2
This diesel generator start from a standby condition in ≤ 15 sec. shall be accomplished at least once every 184 days. All other diesel generator starts for this surveillance may be in accordance with vendor recommendations.

Note 3
Diesel generator loading may be accomplished in accordance with vendor recommendations such as gradual loading.

Note 4
Momentary transients outside this load band due to changing loads will not invalidate the test. Load ranges are allowed to preclude over- loading the diesel generators.

FUEL HANDLING AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.11 The fuel handling area ventilation system shall be operating and discharging through the HEPA filters and charcoal adsorbers.

APPLICABILITY: Whenever irradiated fuel is being moved in the storage pool and during crane operation with loads over the storage pool.

ACTION:

a. With the fuel handling area ventilation system not operating, suspend all operations involving movement of fuel within the spent fuel pool or crane operation with loads over the spent fuel pool until the fuel handling area ventilation system is restored to operation.

b. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11.1 The fuel handling area ventilation system shall be determined to be in operation and discharging through the HEPA filters and charcoal adsorbers at least once per 12 hours.

4.9.11.2 The fuel handling area ventilation system shall be demonstrated OPERABLE in accordance with the Ventilation Filter Testing Program. ~~The fuel handling area ventilation system shall be demonstrated OPERABLE at the following frequencies when irradiated fuel is in the storage pool:~~

~~a. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:~~

~~1. Verifying that the ventilation system satisfies the inplace testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 39,700 cfm ± 10%.~~

~~SURVEILLANCE REQUIREMENT (Continued)~~

~~2. Verifying within 31 days after removal that laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 5.0% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.~~

~~3. Verifying a system flow rate of 39,700 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.~~

~~b. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b. of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 5.0% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.~~

~~c. At lease once per 18 months by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches Water Gauge while operating the system at a flow rate of 39,700 cfm ± 10%.~~

~~d. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 39,700 cfm ± 10%.~~

~~e. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 39,700 cfm ± 10%.~~

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The ~~ANO-2~~ plant manager operations ~~ANO~~ shall be responsible for overall unit operations and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 An individual with an active Senior Reactor Operator (SRO) license shall be designated as responsible for the control room command function while the unit is in MODE 1, 2, 3, or 4. With the unit not in MODES 1, 2, 3, or 4, an individual with an active SRO license or Reactor Operator license shall be designated as responsible for the control room command function.

6.2 ORGANIZATION

6.2.1 ~~OFFSITE AND ONSITE ORGANIZATIONS~~ Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be ~~established and~~ defined and established ~~for the~~ throughout highest management levels, ~~through~~ intermediate levels, and ~~to and including~~ all operating organization positions. These relationships shall be documented and updated, as appropriate, in the ~~form of~~ organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the unit specific titles of those personnel fulfilling ~~its~~ the responsibilities of the positions delineated in these Technical Specifications shall be documented in the Safety Analysis Report (SAR).

b. The ~~ANO-2~~ plant manager operations shall be responsible for overall ~~unit~~ safe operation of the unit and shall have control over those onsite activities necessary for safe operation and maintenance of the ~~plant~~unit.

c. A specified corporate executive shall have corporate responsibility for overall ~~plant~~ unit nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the ~~plant~~ unit to ensure nuclear safety. The specified corporate executive shall be documented in the SAR.

d. The individuals who train the operating staff, ~~and those who~~ carry out health physics, or ~~and~~ perform quality assurance functions may report to the appropriate onsite manager; however, these~~y~~ individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 ~~UNIT STAFF~~ Unit Staff

a. A non-licensed operator shall be on site when fuel is in the reactor and two additional non-licensed operators shall be on site when the reactor is in MODES 1, 2, 3, or 4.

~~a.~~b ~~Each on-duty shift shall be composed of at least the~~The minimum shift crew composition ~~shown in Table 6.2-1.~~for licensed operators shall

meet the minimum staffing requirements of 10 CFR 50.54(m)(2)(i) for one unit, one control room.

~~b. At least one licensed Operator shall be in the control room when fuel is in the reactor.~~

c. ~~At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.~~Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) for one unit, one control room, and 6.2.2.a and 6.2.2.g for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew composition to within the minimum requirements.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

~~e. All CORE ALTERATIONS shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.~~

~~f~~g. In MODES 1, 2, 3, or 4, an individual shall provide advisory technical support for the ~~unit~~ operations shift ~~supervisor~~ crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy on Engineering Expertise on Shift.

~~g~~e. ~~Administrative control shall be established to limit t~~The amount of overtime worked by ~~plant~~ unit staff members performing safety-related functions~~.~~
~~These administrative controls~~ shall be limited and controlled in accordance with the
~~guidance provided by the~~ NRC Policy Statement on working hours (Generic Letter ~~No.~~ 82-12).

~~h~~f. The operations manager or ~~the~~ assistant operations manager shall hold a SRO ~~senior reactor operator~~ license.

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~~TABLE 6.2-1~~

~~MINIMUM SHIFT CREW COMPOSITION#~~

~~LICENSE CATEGORY~~	~~APPLICABLE MODES~~	
	~~1, 2, 3, & 4~~	~~5 & 6~~
~~SOL~~	~~2~~	~~1*~~
~~OL~~	~~2~~	~~1~~
~~Non-Licensed~~	~~3~~	~~1~~

~~*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE ALTERATIONS.~~

~~#Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.~~

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI ~~N18.1-1971~~ ANS 3.1 - 1978 for comparable positions, except for (1) the designated radiation protection manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

~~6.4 DELETED~~

~~6.5 DELETED~~

~~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 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~~4 PROCEDURES ~~AND PROGRAMS~~

~~6.8.1~~6.4.1 Written procedures shall be established, implemented and maintained covering the following activities ~~referenced below~~:

a. ~~a.~~ The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix "A", ~~of Regulatory Guide 1.33, Revision 2,~~ February 1978.

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Section 7.1 of Generic Letter 82-33.

~~b. Refueling operations.~~
~~c. Surveillance and test activities of safety related equipment.~~
~~d. (Deleted)~~
~~e. (Deleted)~~
~~f~~c. Fire Protection Program implementation.

d. All programs specified in Specification 6.5.

~~g~~e. 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~~

~~6.8.3 Deleted~~

~~6.8.4 The following programs shall be established, implemented, and maintained:~~

~~a.~~6.5.4 Radioactive Effluent Controls Program

This program conforms with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

~~1)~~a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

~~2)~~b. Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20, Appendix B, Table II, Column 2;

~~3)~~c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

~~4)~~d. Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS, conforming to 10 CFR 50, Appendix I;

~~5)~~e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

~~6)~~f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

~~7)~~g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table II, Column 1;

The provisions of TS 4.0.2 and 4.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

~~8)~~h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

~~9)~~i. Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

~~10)~~j. Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

~~b.~~6.5.5 Component Cyclic or Transient Limit Program

This program provides controls to track the SAR Section 5.2.1.5 cyclic or transient occurrences to ensure that components are maintained within the design limits.

~~6.9~~6.6 REPORTING REQUIREMENTS

~~ROUTINE REPORTS~~

~~6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Administrator of the Regional Office unless otherwise noted.~~

~~STARTUP REPORT~~

~~6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.~~

~~6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.~~

~~6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.~~

~~ANNUAL REPORTS [1]~~

~~6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.~~

~~6.9.1.5. Reports required on an annual basis shall include:~~

6.6.1 Occupational Radiation Exposure Report (Note 1)

~~a.~~ A tabulation on an annual basis ~~for~~ of the number of station, utility and other personnel (including contractors), for whom monitoring was performed, receiving ~~exposures~~an annual deep dose equivalent >100 mrems ~~greater than 100 mrem/yr~~ and the~~ir~~ associated ~~man rem exposure~~collective deep dose equivalent (reported in person-rem) according to work and job functions,[2] (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignment to various duty functions may be estimates based on pocket ~~dosimeter~~ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures total~~l~~ing ~~less than~~ < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80 percent~~%~~ of the total ~~whole body~~deep dose equivalent received from external sources ~~shall~~ should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

Note 1 - A single submittal may be made for ANO. The submittal should combine sections common to both units.

~~B~~6.6.7. ~~The complete results of steam generator tube inservice inspections performed during the report period (reference Specification 4.4.5.5.b).~~Steam Generator Tube Surveillance Reports

a. Following each inservice inspection of steam generator tubes the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be reported within 12 months following the completion of the inservice inspection. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections, which fall into Category C-3, shall be reported in a Special Report to the Administrator of the Regional Office as denoted by Table 6.5.9-2 Notification of the Commission will be made prior to resumption of plant operation (i.e., prior to entering Mode 4). The written Special Report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

~~c. Documentation of all challenges to the pressurizer safety valves.~~

~~d. Deleted~~

~~e.~~6.6.8 The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded the results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history.

Note ~~1/~~ A single submittal may be made for ~~a multiple unit station~~ANO.. The submittal should combine ~~those~~ sections ~~that are~~ common to ~~all~~ both units ~~at the station~~.

~~2/ This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.~~

~~starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.~~

6.6.4 ~~MONTHLY OPERATING REPORT~~Monthly Operating Report

~~6.9.1.6~~ Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

~~SPECIAL REPORTS~~

~~6.9.2 Special reports shall be submitted to the Administrator of the Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:~~

~~a. ECCS Actuation, Specifications 3.5.2 and 3.5.3.~~

~~b. Deleted~~

~~c. Deleted~~

~~d. Deleted~~

~~e. Deleted~~

~~f. Deleted~~

~~g. Deleted~~

~~h. Deleted~~

~~i. Inoperable Containment Radiation Monitors, Specification 3.3.3.1.~~

~~j. Steam Generator Tubing Surveillance — Category C-3 Results, Specification 4.4.5.5.~~

~~k. Maintenance of Spent Fuel Pool Structural Integrity, Specification 3.7.12.~~

~~l. Deleted~~

~~m. Deleted~~

~~n. Inoperable Reactor Vessel Level Monitoring System (RVLMS), Specification 3.3.3.6, Table 3.3-10 Item 14.~~

~~o. Inoperable Main Steam Line Radiation Monitors, Specification 3.3.3.1, Table 3.3-6.~~

6.6.3 RADIOACTIVE EFFLUENT RELEASE REPORT *-(Note 1)

~~6.9.3~~ The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

*-Note 1 - A single submittal may be made for ANO. The submittal should combine ~~those~~ sections ~~that are~~ common to both units. The submittal shall specify the releases of radioactive material from each unit.

6.5.2 Primary Coolant Sources Outside Containment

EOI shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following.

a. Provisions establishing preventative maintenance and periodic visual inspection requirements, and

b. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

6.5.10 Secondary Water Chemistry Monitoring

EOI shall implement a secondary water chemistry monitoring program to minimize steam generator tube degradation. The program shall be defined in specific plant procedures and shall include:

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

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

c. Identification of process sampling points;

d. Procedure for the recording and management of data;

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

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

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6.6.2 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT *Note 1)

~~6.9.4~~ The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in
10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

*Note 1 - A single submittal may be made for ANO. The submittal should combine ~~those~~ sections ~~that are~~ common to both units.

6.6.5 CORE OPERATING LIMITS REPORT (COLR)

~~6.9.5~~a. ~~The core~~ Core operating limits shall be established ~~and documented in the CORE OPERATING LIMITS REPORT~~ prior to each reload cycle, or prior to any remaining ~~part~~ portion of a reload cycle, and shall be documented in the COLR.

~~6.9.5.1~~b. The analytical methods used to determine the core operating limits addressed by the individual Technical Specifications shall be those previously reviewed and approved by the NRC for use at ANO-2, specifically:

1) "The ROCS and DIT Computer Codes for Nuclear Design", CENPD-266-P-A, April 1983 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.4 for MTC, 3.1.3.6 for Regulating and Group P CEA Insertion Limits, and 3.2.4.b for DNBR Margin).

2) "CE Method for Control Element Assembly Ejection Analysis," CENPD-0190-A, January 1976 (Methodology for Specification 3.1.3.6 for Regulating and Group P CEA Insertion Limits and 3.2.3 for Azimuthal Power Tilt).

3) "Modified Statistical Combination of Uncertainties, CEN-356(V)-P-A, Revision 01-P-A, May 1988 (Methodology for Specification 3.2.4.c and 3.2.4.d for DNBR Margin and 3.2.7 for ASI).

4) "Calculative Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P, August 1974 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

5) "Calculational Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P, Supplement 1, February 1975 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

6) "Calculational Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P, Supplement 2-P, July 1975 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

7) "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CEN-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

8) "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

9) "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137, Supplement 1-P, January 1977 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

CORE OPERATING LIMITS REPORT

10) "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137, Supplement 2-P-A, dated April, 1998 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

11) "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margin, 3.1.1.4 for MTC, 3.1.3.1 for CEA Position, 3.1.3.6 for Regulating CEA and Group P Insertion Limits, and 3.2.4.b for DNBR Margin).

12) "Technical Manual for the CENTS Code," CENPD 282-P-A, February 1991 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margin, 3.1.1.4 for MTC, 3.1.3.1 for CEA Position, 3.1.3.6 for Regulating and Group P Insertion Limits, and 3.2.4.b for DNBR Margin.

13) Letter: O.D. Parr (NRC) to F.M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for 6.~~9~~6.5.~~1~~b.4, 6.~~6~~9.5.~~1~~b.5, and 6.~~9~~6.5.~~1~~b.8 methodologies.

14) Letter: O.D. Parr (NRC) to A.E. Scherer (CE), dated December 9, 1975 (NRC Staff Review of the Proposed Combustion Engineering ECCS Evaluation Model changes). NRC approval for 6.~~9~~6.5.~~1~~b.6 methodology.

15) Letter: K. Kniel (NRC) to A.E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 6.~~9~~6.5.~~1~~b.9 methodology.

16) Letter: 2CNA038403, dated March 20, 1984, J.R. Miller (NRC) to J.M. Griffin (AP&L), "CESEC Code Verification." NRC approval for 6.~~9~~6.5.~~1~~b.11 methodology.

~~6.9.5.2~~c. The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

~~6.9.5.3~~d. The ~~CORE OPERATING LIMITS REPORT~~COLR, including any mid-cycle revisions or supplements ~~thereto~~, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

~~6.10 DELETED~~

6.5.9 Steam Generator (SG) Tube Surveillance Program

Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

6.5.9.1 Steam Generator Sample Selection and Inspection

Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 6.5.9-1.

6.5.9.2 Steam Generator Tube Sample Selection and Inspection

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 6.5.9-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in specification 6.5.9.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 6.5.9.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the pre-service inspection) of each steam generator shall include:

   1. All non-plugged tubes that previously had detectable wall penetrations (>20%).

   2. Tubes in those areas where experience has indicated potential problems.

   3. A tube inspection (pursuant to Specification 6.5.9.4.a.9) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 6.5.9-2) during each inservice inspection may be subjected to a partial inspection provided:

   1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

   2. The inspections include those portions of the tubes where imperfections were previously found.

The result of each sample inspection shall be classified into one to the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

6.9.5.3 Inspection Frequencies

The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the pre-service inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 6.5.9-2 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 6.5.9.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 6.5.9-2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2.

2. A seismic occurrence greater than the Operating Basis Earthquake.

3. A loss-of coolant accident requiring actuation of the engineered safeguards.

4. A main steam line or feedwater line break.

6.5.9.4 Acceptance Criteria

a. As used in this Specification

1. Tubing or Tube means that portion of the tube, which forms the primary system to secondary system pressure boundary.

2. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

3. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

4. Degraded Tube means a tube containing imperfections ≥20% of nominal wall thickness caused by degradation.

5. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

6. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

7. Plugging Limit means the imperfection depth at or beyond which the tube hall be removed from service by plugging because it may become unserviceable prior to the next inspection. The plugging limit is equal to 40% of the nominal tube wall thickness.

8. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 6.5.9.3.c, above.

9. Tube Inspection means an inspection of the steam generator tube from tube end (cold leg side) to tube end (hot leg side).

10. Pre-service Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the hydrostatic test and prior to POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 6.5.9-2.

TABLE 6.5.9-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	Yes
No. of Steam Generators per Unit	Two
First Inservice Inspection	One
Second & Subsequent Inservice Inspections	One [1]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE 6.5.9-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first Sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in the other S.G.	Other S.G. is C-1	None	N/A	N/A
			Other S.G. is C-2	Perform action for C-2 result of second sample	N/A	N/A
		Special Report to NRC per Specification ~~6.9.2~~ 6.6.7.	Other S.G. is C-3	Inspect all tubes in the other S.G. and plug defective tubes. Special Report to NRC per Spec. ~~6.9.2~~ 6.6.7.	N/A	N/A

S = 3 2/n ~~%~~ Where n is the number of steam generators inspected during an inspection.

~~6.11 RADIATION PROTECTION PROGRAM~~

~~Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.~~

~~6.12 (DELETED)~~

6.5.11 Ventilation Filter Testing Program

Each control room emergency air (CREVS) and fuel handling area (FHAVS) filtration system shall be demonstrated OPERABLE:

a. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is:

a. 2000 cfm ± 10% for the CREVS.

b. 39,700 cfm ± 10% for the FHAVS.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of:

a. For CREVS:

i. ≤ 2.5% for 2 inch charcoal adsorber beds, or

ii. ≤ 0.5% for 4 inch charcoal adsorber beds

b. < 5.0% for FHAVS.

3. When tested in accordance with ANSI N510-1975, verify a system flow rate during system operation of:

a. 2000 cfm ± 10% for the CREVS.

b. 39,700 cfm ± 10% for the FHAVS.

b. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of:

1. For CREVS:

a. ≤ 2.5% for 2 inch charcoal adsorber beds, or

b. ≤ 0.5% for 4 inch charcoal adsorber beds

2. < 5.0% for FHAVS.

c. At least once per 18 months by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches Water Gauge while operating the system at a flow rate of:

1. 2000 cfm ± 10% for the CREVS.

2. 39,700 cfm ± 10% for the FHAVS.

d. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks when they are tested in-place in accordance with ANSI N510-1975 remove:

1. ≥ 99.95% of the DOP while operating the system at a flow rate of 2000 cfm ± 10% for the CREVS.

2. ≥ 99% of the DOP while operating the system at a flow rate of 39,700 cfm ± 10% for the FHAVS.

e. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm ±10%.

1. 2000 cfm ± 10% for the CREVS.

2. 39,700 cfm ± 10% for the FHAVS.

The provisions of TS 4.0.2 and 4.0.3 are applicable to the Ventilation Filter Testing Program surveillance frequency.

~~6.12.2 (DELETED)~~

~~6.13~~6.7 HIGH RADIATION AREA

~~6.13.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area (as defined in 20.202(b)(3) of 10 CFR 20) in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring the issuance of a radiation work permit. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:~~

~~a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.~~

~~b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.~~

~~c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the radiation work permit.~~

~~6.13.2 The requirements of 6.13.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and access to these areas shall be maintained under the administrative control of the shift supervisor on duty and/or the designated radiation protection manager.~~

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP), or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be

exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP, or equivalent, while in the area by means of closed circuit television, or personnel qualified in radiation protection procedures responsible for controlling personnel radiation exposure in the area and with the means to communicate with individuals in the area who are covered by such surveillance.

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

~~6.1~~46.5.1 OFFSITE DOSE CALCULATION MANUAL (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program.

The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release ~~and Annual Radiological Environmental Operating~~ Reports ~~required by Specifications 6.9.3 and 6.9.4~~.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s),

   2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after approval of the ANO General Manager~~, Plant Operations~~; and

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 effective. 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 ~~also~~ indicate the date (i.e., month and year) the change was implemented.

~~6.15~~6.5.16 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 58 psig.

The maximum allowable containment leakage rate, L_a, shall be 0.1% of containment air weight per day at P_a.

Leakage rate acceptance criteria are:

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

b. Air lock acceptance criteria are:

   1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2. Leakage rate for each door is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

6.5.13 Diesel Fuel Oil Testing Program

At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank obtained in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM D975-74 when checked for viscosity, water and sediment.

6.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

   1. A change in the TS incorporated in the license; or

   2. A change to the updated SAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

   Proposed changes that meet these criteria shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases

implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

c. The Bases Control Program shall contain provisions to ensure that the bases are maintained consistent with the SAR.

ATTACHMENT 3

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CHANGES TO TECHNICAL SPECIFICATION BASES PAGES

Demonstration of the safety valves' lift setting will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The steam bubble functions to relieve RCS pressure during all design transients.

The requirement that 150 KW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss-of-offsite power condition to maintain natural circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

~~The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.~~

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 150 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 150 gallons per day per steam generator can readily be detected by radiation monitors on the secondary system. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tubes examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit as defined in ~~Surveillance Requirement 4.4.5.4.a~~ Steam Generator Tube Surveillance Program. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that could affect tube wall integrity. Additionally, upgraded testing methods will be evaluated and appropriately implemented as better methods are developed and validated for commercial use.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3 certain results will be reported in a Special Report to the Commission. ~~pursuant to Specification 6.9.2 as denoted by Table 4.2-2.~~ Notification of the Commission will be made prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

GDC 30 of Appendix A to 10 CFR 50 requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems" May 1973. Likewise, the actions implemented upon inoperability of a required leak detection instrument are sufficient in maintaining the diversity and accuracy needed to effectively detect RCS leaks.

Industry practice has shown that water flow changes of 0.5 gpm to 1.0 gpm can readily be detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. In addition, the reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Instrument sensitivities of 10 - 10^6 cpm for particulate and gaseous monitoring are practical for these leakage detection systems.

12 hours is provided by a footnote to allow for plant stabilization before performance of the required reactor coolant inventory balance. This provision is necessary to ensure an accurate measurement is obtained.

3/4.9.9 and 3/4.9.10 WATER LEVEL-REACTOR VESSEL AND SPENT FUEL POOL WATER LEVEL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 12% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.11 FUEL HANDLING AREA VENTILATION SYSTEM

The limitations on the fuel handling area ventilation system ensure that all radioactive materials released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere. The operation of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

~~Acceptable removal efficiency is shown by methyl iodide penetration of less than 5.0% when tests are performed in accordance with ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," at a temperature of 30°C and a relative humidity of 95%. The penetration acceptance criterion is determined by the following equation:~~

~~Allowable Penetration = [100% - methyl iodide efficiency for charcoal credited in accident analysis] / safety factor of 2~~

~~Applying a safety factor of 2 is acceptable because ASTM D3803-1989 is a more accurate and demanding test than older tests.~~

3/4.9.12 FUEL STORAGE

Region 1 and Region 2 of the spent fuel storage racks are designed to assure fuel assemblies of less than or equal to 5.0 w/o U-235 enrichment that are within the limits of Figure 3.9.2 will be maintained in a subcritical array with $K_{eff} \leq 0.95$ in unborated water. These conditions have been verified by criticality analyses.

The requirement for 1600 ppm boron concentration is to assure the fuel assemblies will be maintained in a subcritical array with $K_{eff} \leq 0.95$ in the event of a postulated accident. Analysis has shown that, during a postulated accident with the fuel stored within the limits of this specification, that a K_{eff} of ≤ 0.95 will be maintained when the boron concentration is at or above 1000 ppm.

Normally, fuel stored in a cross-hatch storage configuration must have all four diagonal spaces or at least two adjacent faces remain vacant to meet the criticality safety analysis mentioned above. However, the spent fuel pool walls may be credited as a neutron leakage path. Therefore, vacant spaces face adjacent to the walls of the Region I cross-hatch configured assemblies may be used to store fuel assemblies that are outside of the area of the graph enclosed by Curve A on Figure 3.9.2, excluding the most southeast and southwest corner spaces of Region 1 which must remain empty.

ATTACHMENT 4

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LIST OF REGULATORY COMMITMENTS

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Modify SAR section 1.3.3 to delete the reference to the Technical Specifications.	x		Upon Approval

ATTACHMENT 5

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PROPOSED TECHNICAL SPECIFICATION PAGES (RE-TYPED)

New TS #	Current TS #
6.1	6.1
6.1.1	6.1.1
6.1.2	6.1.2
6.2	6.2
6.2.1	6.2.1
6.2.2	6.2.2
6.2.2.a	NLO Part of Table 6.2-1
6.2.2.b	6.2.2.a & SLO & LO part of Table 6.2-1
6.2.2.c	Table 6.2-1 # Note
6.2.2.d	6.2.2.d
6.2.2.e	6.2.2.g
6.2.2.f	6.2.2.h
6.2.2.g	6.2.2.f
6.3	6.3
6.3.1	6.3.1
6.4	6.8
6.4.1	6.8.1
6.4.1.a	6.8.1.a
6.4.1.b (NEW)	N/A
6.4.1.c	6.8.1.f
6.4.1.d	6.8.1.i plus other programs
6.4.1.e	6.8.1.g
6.5.1	6.14
6.5.2	FOL 2.(c).(5)
6.5.4	6.8.4.a
6.5.5	6.8.4.b
6.5.9	SR 4.4.5.0, 4.4.5.1, 4.4.5.2, 4.4.5.3, 4.4.5.4, Tables 4.4-1 and 4.4-2
6.5.10	FOL 2.C.(3) (p)
6.5.11	SR 4.7.6.1.2 & 4.9.11.2
6.5.13	SR 4.8.1.1.2.b

New TS #	Current TS #
6.5.14 (NEW)	N/A
6.5.16	6.15
6.6	6.9
6.6.1	6.9.1.5.a
6.6.2	6.9.4
6.6.3	6.9.3
6.6.4	6.9.16
6.6.5	6.9.5
6.6.7	SR 4.4.5.5 & 6.9.1.5.b & 6.9.2.j
6.7	6.13

(i) Containment Radiation Monitor

AP&L shall, prior to July 31, 1980 submit for Commission review and approval documentation which establishes the adequacy of the qualifications of the containment radiation monitors located inside the containment and shall complete the installation and testing of these instruments to demonstrate that they meet the operability requirements of Technical Specification No. 3.3.3.6.

2.C.(3)(j) Deleted per Amendment 7, 12/1/78.

2.C.(3)(k) Deleted per Amendment 12, 6/12/79 and Amendment No. 31, 5/12/82.

2.C.(3)(l) Deleted per Amendment 24, 6/19/81.

2.C.(3)(m) Deleted per Amendment 12, 6/12/79.

2.C.(3)(n) Deleted per Amendment 7, 12/1/78.

2.C.(3)(o) Deleted per Amendment 7, 12/1/78.

2.C.(3)(p) Deleted

2.C.(4) (Number has never been used.)

2.C.(5) Deleted

(6) EOI shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

2.C.(7) Deleted per Amendment 78, 7/22/86.

(8) Antitrust Conditions

EOI shall not market or broker power or energy from Arkansas Nuclear One, Unit 2. Entergy Arkansas, Inc. is responsible and accountable for the actions of its agents to the extent said agent's actions affect the marketing or brokering of power or energy from ANO, Unit 2.

(9) Rod Average Fuel Burnup

Entergy Operations is authorized to operate the facility with an individual rod average fuel burnup (burnup averaged over the length of a fuel rod) not to exceed 60 megawatt-days/kilogram of uranium.

D. Physical Protection

EOI 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 plan, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Arkansas Nuclear One Industrial Security Plan," with revisions submitted through August 4, 1995. The Industrial Security Plan also includes the requirements for guard training and qualification in Appendix A of the safeguards contingency events in Chapter 7. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

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EXCLUSION AREA

1.31 The EXCLUSION AREA is that area surrounding ANO within a minimum radius of .65 miles of the reactor buildings and controlled to the extent necessary by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

UNRESTRICTED AREA

1.32 An UNRESTRICTED AREA shall be any area at or beyond the exclusion area boundary.

CORE OPERATING LIMITS REPORT

1.33 The CORE OPERATING LIMITS REPORT is the ANO-2 specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Technical Specification 6.6.5 Plant operation within these operating limits is addressed in individual specifications.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Pool Area Monitor	1	Note 1	$\leq 1.5 \times 10^{-2}$ R/hr	10^{-4} - 10^{1} R/hr	13
b. Containment High Range	2	1, 2, 3, & 4	Not Applicable	1 - 10^{7} R/hr	18
2. PROCESS MONITORS					
a. Containment Purge and Exhaust Isolation	1	5 & 6	$\leq$ 2 x background	10 - 10^{6} cpm	16
b. Control Room Ventilation Intake Duct Monitors	2	Note 2	$\leq$ 2 x background	10 - 10^{6} cpm	17, 20, 21
c. Main Steam Line Radiation Monitors	1/Steam Line	1, 2, 3, & 4	Not Applicable	10^{-1} - 10^{4} mR/hr	19

Note 1 - With fuel in the spent fuel pool or building.
Note 2 - MODES 1, 2, 3, 4, and during handling of irradiated fuel.

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 13 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 16 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, complete the following:

a. If performing CORE ALTERATIONS or moving irradiated fuel within the reactor building, secure the containment purge system or suspend CORE ALTERATIONS and movement of irradiated fuel within the reactor building.

b. If a containment PURGE is in progress, secure the containment purge system.

c. If continuously ventilating, verify the SPING monitor operable or perform the ACTIONS of 3.3.3.9, or secure the containment purge system.

ACTION 17 - In MODE 1, 2, 3, or 4 with no channels OPERABLE, within 1 hour initiate and maintain operation of the control room emergency ventilation system (CREVS) in the recirculation mode of operation or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN in the following 30 hours.

ACTION 18 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, (1) either restore the inoperable channel to OPERABLE status within 7 days or (2) prepare and submit a Special Report to the Commission within 30 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status. With both channels inoperable, initiate alternate methods of monitoring the containment radiation level within 72 hours in addition to the actions described above.

ACTION 19 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2) prepare and submit a Special Report to the Commission within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 20 - In MODE 1, 2, 3, or 4 with the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 7 days, or initiate and maintain the control room emergency ventilation system in the recirculation mode of operation, or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN in the following 30 hours.

ACTION 21 - During handling of irradiated fuel with one or two channels inoperable, immediately place one OPERABLE CREVS train in the emergency recirculation mode or immediately suspend handling of irradiated fuel.

TABLE 3.3-10 (Con't)
POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
13. In Core Thermocouples (Core-Exit Thermocouples)	2/core quadrant	1
14. Reactor Vessel Level Monitoring System (RVLMS)	2	3, 4

Action 1: With the number of OPERABLE post-accident monitoring channels less than required by Table 3.3-10, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

Action 2: With the number of OPERABLE post-accident monitoring channels less than required by Table 3.3-10, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

If only one channel is inoperable and containment entry is required to restore the inoperable channel, the channel need not be restored until the following refueling outage.

Action 3: With the number of OPERABLE channels one less than the minimum number of channels required to be OPERABLE:

a. If repairs are feasible, restore the inoperable channel to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. If repair is not feasible without shutting down, operations may continue and a special report shall be submitted to the NRC within 30 days following the failure; describing the action taken, the cause of the inoperability, and the plans and schedule for restoring the channel to OPERABLE status during the next scheduled refueling outage.

Action 4: With the number of OPERABLE channels two less than the minimum channels required to be OPERABLE:

a. If repairs are feasible, restore at least one inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. If repair is not feasible without shutting down, operation may continue and a special report shall be submitted to the NRC within 30 days following the failure; describing the action taken, the cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status during the next scheduled refueling outage.

STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1,2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing Tavg above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5 Perform steam generator inspections in accordance with the Steam Generator Tube Surveillance Program.

ECCS SUBSYSTEMS - $T_{avg} \geq 300°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection pump,

b. One OPERABLE low-pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the NRC within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
2CV-5101	HPSI Hot Leg Injection Isolation	Closed
2CV-5102	HPSI Hot Leg Injection Isolation	Closed
2BS26	RWT Return Line	Open

*With pressurizer pressure ≥ 1700 psia.

ECCS SUBSYSTEMS - $T_{avg} \leq 300°F$

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE high-pressure safety injection pump, and

b. An OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal.

APPLICABILITY: MODES 3* and 4.

ACTION:

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the NRC within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

*With pressurizer pressure < 1700 psia.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION AND AIR CONDITIONING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 Two independent control room emergency ventilation and air conditioning systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, or during handling of irradiated fuel.

ACTION:

MODES 1, 2, 3, and 4

a. With one control room emergency air conditioning system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one control room emergency ventilation system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With one control room emergency air conditioning system and one control room emergency ventilation system inoperable, restore the inoperable control room emergency ventilation system to OPERABLE status within 7 days and restore the inoperable control room emergency air conditioning system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two control room emergency ventilation systems inoperable due to an inoperable control room boundary, restore the control room boundary to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

During Handling of Irradiated Fuel

e. With one control room emergency air conditioning system inoperable, restore the inoperable system to OPERABLE status within 30 days or immediately place the OPERABLE system in operation; otherwise, suspend all activities involving the handling of irradiated fuel. The provisions of Specification 3.0.4 are not applicable.

f. With one control room emergency ventilation system inoperable, restore the inoperable system to OPERABLE status within 7 days or immediately place the control room in the emergency recirc mode of operation; otherwise, suspend all activities involving the handling of irradiated fuel. The provisions of Specification 3.0.4 are not applicable.

ACTION (Continued)

g. With one control room emergency air conditioning system and one control room emergency ventilation system inoperable:

1. restore the inoperable control room emergency ventilation system to OPERABLE status within 7 days or immediately place the control room in the emergency recirc mode of operation, and

2. restore the inoperable control room emergency air conditioning system to OPERABLE status within 30 days or immediately place the OPERABLE system in operation;

3. otherwise, suspend all activities involving the handling of irradiated fuel.

4. The provisions of Specification 3.0.4 are not applicable.

h. With both control room emergency air conditioning systems or both control room emergency ventilation systems inoperable, immediately suspend all activities involving the handling of irradiated fuel.

SURVEILLANCE REQUIREMENTS

4.7.6.1.1 Each control room emergency air conditioning system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Starting each unit from the control room, and

2. Verifying that each unit operates for at least 1 hour and maintains the control room air temperature ≤ 84°F D.B.

b. At least once per 18 months by verifying a system flow rate of 9900 cfm ± 10%.

4.7.6.1.2 Each control room emergency air filtration system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months by verifying that on a control room high radiation test signal, the system automatically isolates the control room within 10 seconds and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

c. By performing the required Control Room Emergency Ventilation filter testing in accordance with the Ventilation Filter Testing Program (VFTP).

If any snubber selected for functional testing either fails to activate or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be evaluated in a manner to ensure their OPERABILITY. This requirement shall be independent of the requirements stated in Specification 4.7.8.d for snubbers not meeting the functional test acceptance criteria.

g. Preservice Testing of Repaired, Replacement and New Snubbers

Preservice operability testing shall be performed on repaired, replacement or new snubbers prior to installation. Testing may be at the manufacturer's facility. The testing shall verify the functional test acceptance criteria in 4.7.8.e.

In addition, a preservice inspection shall be performed on each repaired, replacement or new snubber and shall verify that:

1) There are no visible signs of damage or impaired operability as a result of storage, handling or installation;

2) The snubber load rating, location, orientation, position setting and configuration (attachment, extensions, etc.), are in accordance with design;

3) Adequate swing clearance is provided to allow snubber movement;

4) If applicable, fluid is at the recommended level and fluid is not leaking from the snubber system;

5) Structural connections such as pins, bearings, studs, fasteners and other connecting hardware such as lock nuts, tabs, wire, and cotter pins are installed correctly.

h. Snubber Seal Replacement Program

The seal service life of hydraulic snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The expected service life for the various seals, seal materials, and applications shall be determined and established based on engineering information and the seals shall be replaced so that the expected service life will not be exceeded during a period when the snubber is required to be OPERABLE. The seal replacement shall be documented.

3/4.7.12 SPENT FUEL POOL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.7.12 The structural integrity of the spent fuel pool shall be maintained in accordance with Specification 4.7.12.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

a. With the structural integrity of the spent fuel pool not conforming to the above requirements, in lieu of any other report, prepare and submit a Special Report to the NRC within 30 days of a determination of such non-conformity.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.12.1 Inspection Frequencies - The structural integrity of the spent fuel pool shall be determined per the acceptance criteria of Specification 4.7.12.2 at the following frequencies:

a. At least once per 92 days after the pool is filled with water. If no abnormal degradation or other indications of structural distress are detected during five consecutive inspections, the inspection interval may be extended to at least once per 5 years.

b. Within 24 hours following any seismic event which actuates or should have actuated the seismic monitoring instrumentation.

4.7.12.2 Acceptance Criteria - The structural integrity of the spent fuel pool shall be determined by a visual inspection of at least the interior and exterior surfaces of the pool, the struts in the tilt pit, the surfaces of the separation walls, and the structural slabs adjoining the pool walls. This visual inspection shall verify no changes in the concrete crack patterns, no abnormal degradation or other signs of structural distress (i.e, cracks, bulges, out of plumbness, leakage, discolorations, efflorescence, etc.).

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown be transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE: (Note 1)

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day fuel tank.

2. Verifying the fuel level in the fuel storage tank.

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

4. Verifying the diesel starts from a standby condition and accelerates to at least 900 rpm in ≤ 15 seconds. (Note 2)

5. Verifying the generator is synchronized, loaded to an indicated 2600 to 2850 Kw and operates for ≥ 60 minutes. (Notes 3 & 4)

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. By testing the diesel fuel oil in accordance with the Diesel Fuel Oil Testing Program.

Note 1
All planned diesel generator starts for the purposes of these surveillances may be preceded by prelube procedures.

Note 2
This diesel generator start from a standby condition in ≤ 15 sec. shall be accomplished at least once every 184 days. All other diesel generator starts for this surveillance may be in accordance with vendor recommendations.

Note 3
Diesel generator loading may be accomplished in accordance with vendor recommendations such as gradual loading.

Note 4
Momentary transients outside this load band due to changing loads will not invalidate the test. Load ranges are allowed to preclude overloading the diesel generators.

FUEL HANDLING AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.11 The fuel handling area ventilation system shall be operating and discharging through the HEPA filters and charcoal adsorbers.

APPLICABILITY: Whenever irradiated fuel is being moved in the storage pool and during crane operation with loads over the storage pool.

ACTION:

a. With the fuel handling area ventilation system not operating, suspend all operations involving movement of fuel within the spent fuel pool or crane operation with loads over the spent fuel pool until the fuel handling area ventilation system is restored to operation.

b. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11.1 The fuel handling area ventilation system shall be determined to be in operation and discharging through the HEPA filters and charcoal adsorbers at least once per 12 hours.

4.9.11.2 The fuel handling area ventilation system shall be demonstrated OPERABLE in accordance with the Ventilation Filter Testing Program.

6.1 RESPONSIBILITY

6.1.1 The plant manager operations shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 An individual with an active Senior Reactor Operator (SRO) license shall be designated as responsible for the control room command function while the unit is in MODE 1, 2, 3, or 4. With the unit not in MODES 1, 2, 3, or 4, an individual with an active SRO license or Reactor Operator license shall be designated as responsible for the control room command function.

6.2 ORGANIZATION

6.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in the organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the unit specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the Safety Analysis Report (SAR).

b. The plant manager operations shall be responsible for overall safe operation of the unit and shall have control over those onsite activities necessary for safe operation and maintenance of the unit.

c. A specified corporate executive shall have corporate responsibility for overall unit nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the unit to ensure nuclear safety. The specified corporate executive shall be documented in the SAR.

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 Unit Staff

a. A non-licensed operator shall be on site when fuel is in the reactor and two additional non-licensed operators shall be on site when the reactor is in MODES 1, 2, 3, or 4.

b. The minimum shift crew composition for licensed operators shall meet the minimum staffing requirements of 10 CFR 50.54(m)(2)(i) for one unit, one control room.

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) for one unit, one control room, and 6.2.2.a and 6.2.2.g for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew composition to within the minimum requirements.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

e. The amount of overtime worked by unit staff members performing safety-related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).

f. The operations manager or assistant operations manager shall hold a SRO license.

g. In MODES 1, 2, 3, or 4, an individual shall provide advisory technical support for the operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy on Engineering Expertise on Shift.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI ANS 3.1 - 1978 for comparable positions, except for (1) the designated radiation protection manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 PROCEDURES

6.4.1 Written procedures shall be established, implemented and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978.

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Section 7.1 of Generic Letter 82-33.

c. Fire Protection Program implementation.

d. All programs specified in Specification 6.5.

e. 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.

   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.

6.5 PROGRAMS AND MANUALS

6.5.1 Offsite Dose Calculation Manual (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program.

The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s),

   2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after approval of the ANO General Manager; and

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 effective. 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.

6.5.2 Primary Coolant Sources Outside Containment

EOI shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

a. Provisions establishing preventative maintenance and periodic visual inspection requirements, and

b. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

6.5.3 Not Used

6.5 PROGRAMS AND MANUALS

6.5.4 Radioactive Effluent Controls Program

This program conforms with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS, conforming to 10 CFR Part 20, Appendix B, Table II Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table II, Column 1;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of TS 4.0.2 and 4.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

6.5 PROGRAMS AND MANUALS

6.5.5 Component Cyclic or Transient Limit Program

This program provides controls to track the SAR Section 5.2.1.5 cyclic or transient occurrences to ensure that components are maintained within the design limits.

6.5.6 Not Used

6.5.7 Reactor Coolant Pump Flywheel Inspection Program

Later

6.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Every 6 weeks	At least once per 42 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice testing activities.

c. The provisions of Specification 4.0.3 are applicable to inservice testing activities, and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

6.5 PROGRAMS AND MANUALS

6.5.9 Steam Generator (SG) Tube Surveillance Program

Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

6.5.9.1 Steam Generator Sample Selection and Inspection

Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 6.5.9-1.

6.5.9.2 Steam Generator Tube Sample Selection and Inspection

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 6.5.9-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in specification 6.5.9.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 6.5.9.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the pre-service inspection) of each steam generator shall include:

   1. All non-plugged tubes that previously had detectable wall penetrations (>20%).

   2. Tubes in those areas where experience has indicated potential problems.

   3. A tube inspection (pursuant to Specification 6.5.9.4.a.9) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 6.5.9-2) during each inservice inspection may be subjected to a partial inspection provided:

   1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

   2. The inspections include those portions of the tubes where imperfections were previously found.

6.5 PROGRAMS AND MANUALS

The result of each sample inspection shall be classified into one to the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

6.5.9.3 Inspection Frequencies

The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the pre-service inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 6.5.9-2 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 6.5.9.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 6.5.9-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

   3. A loss-of coolant accident requiring actuation of the engineered safeguards.

   4. A main steam line or feedwater line break.

6.5 PROGRAMS AND MANUALS

6.5.9.4 Acceptance Criteria

a. As used in this Specification

1. Tubing or Tube means that portion of the tube, which forms the primary system to secondary system pressure boundary.

2. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

3. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

4. Degraded Tube means a tube containing imperfections ≥20% of nominal wall thickness caused by degradation.

5. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

6. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

7. Plugging Limit means the imperfection depth at or beyond which the tube hall be removed from service by plugging because it may become unserviceable prior to the next inspection. The plugging limit is equal to 40% of the nominal tube wall thickness.

8. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 6.5.9.3.c, above.

9. Tube Inspection means an inspection of the steam generator tube from tube end (cold leg side) to tube end (hot leg side).

10. Pre-service Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the hydrostatic test and prior to POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 6.5.9-2.

6.5 PROGRAMS AND MANUALS

TABLE 6.5.9-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Pre-service Inspection	Yes
No. of Steam Generators per Unit	Two
First Inservice Inspection	One
Second & Subsequent Inservice Inspections	One [1]

Table Notations:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE 6.5.9-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first Sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in the other S.G.	Other S.G. is C-1	None	N/A	N/A
			Other S.G. is C-2	Perform action for C-2 result of second sample	N/A	N/A
		Special Report to NRC per Specification 6.6.7	Other S.G. is C-3	Inspect all tubes in the other S.G. and plug defective tubes. Special Report to NRC per Spec. 6.6.7	N/A	N/A

S = 3 2/n % Where n is the number of steam generators inspected during an inspection.

6.5 PROGRAMS AND MANUALS

6.5.10 Secondary Water Chemistry

EOI shall implement a secondary water chemistry monitoring program to minimize steam generator tube degradation. The program shall be defined in specific plant procedures and shall include:

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

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

c. Identification of process sampling points;

d. Procedure for the recording and management of data;

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

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

6.5 PROGRAMS AND MANUALS

6.5.11 Ventilation Filter Testing Program (VFTP)

Each control room emergency air (CREVS) and fuel handling area (FHAVS) filtration system shall be demonstrated OPERABLE:

a. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is:

   a. 2000 cfm ± 10% for the CREVS.

   b. 39,700 cfm ± 10% for the FHAVS.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of:

   a. For CREVS:

      i. ≤ 2.5% for 2 inch charcoal adsorber beds, or

      ii. ≤ 0.5% for 4 inch charcoal adsorber beds

   b. < 5.0% for FHAVS.

3. When tested in accordance with ANSI N510-1975, verify a system flow rate during system operation of:

   a. 2000 cfm ± 10% for the CREVS.

   b. 39,700 cfm ± 10% for the FHAVS.

6.5 PROGRAMS AND MANUALS

b. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of:

1. For CREVS:

a. ≤ 2.5% for 2 inch charcoal adsorber beds, or

b. ≤ 0.5% for 4 inch charcoal adsorber beds

2. < 5.0% for FHAVS.

c. At least once per 18 months by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches Water Gauge while operating the system at a flow rate of:

1. 2000 cfm ± 10% for the CREVS.

2. 39,700 cfm ± 10% for the FHAVS.

d. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks when they are tested in-place in accordance with ANSI N510-1975 remove:

1. ≥ 99.95% of the DOP while operating the system at a flow rate of 2000 cfm ± 10% for the CREVS.

2. ≥ 99% of the DOP while operating the system at a flow rate of 39,700 cfm ± 10% for the FHAVS.

e. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate:

1. 2000 cfm ± 10% for the CREVS.

2. 39,700 cfm ± 10% for the FHAVS.

The provisions of TS 4.0.2 and 4.0.3 are applicable to the Ventilation Filter Testing Program surveillance frequency.

6.5 PROGRAMS AND MANUALS

6.5.12 Not Used

6.5.13 Diesel Fuel Oil Testing Program

At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank obtained in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM D975-74 when checked for viscosity, water and sediment.

6.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

1. A change in the TS incorporated in the license; or

2. A change to the updated SAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

Proposed changes that meet these criteria shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

c. The Bases Control Program shall contain provisions to ensure that the bases are maintained consistent with the SAR.

6.5.15 Not Used

6.5 PROGRAMS AND MANUALS

6.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 58 psig.

The maximum allowable containment leakage rate, L_a, shall be 0.1% of containment air weight per day at P_a.

Leakage rate acceptance criteria are:

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

b. Air lock acceptance criteria are:

   1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2. Leakage rate for each door is $\leq 0.01\ L_a$ when pressurized to $\geq$ 10 psig.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

6.6 REPORTING REQUIREMENTS

6.6.1 Occupational Radiation Exposure Report (Note 1)

A tabulation on an annual basis of the number of station, utility and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent >100 mrems and the associated collective deep dose equivalent (reported in person-rem) according to work and job functions, (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignment to various duty functions may be estimates based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

6.6.2 Annual Radiological Environmental Operating Report (Note 1)

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

Note 1 - A single submittal may be made for ANO. The submittal should combine sections common to both units.

6.6 REPORTING REQUIREMENTS

6.6.3 Radioactive Effluent Release Report (Note 1)

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

Note 1 - A single submittal may be made for ANO. The submittal should combine sections common to both units. The submittal shall specify the releases of radioactive material from each unit.

6.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

6.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR.

b. The analytical methods used to determine the core operating limits addressed by the individual Technical Specifications shall be those previously reviewed and approved by the NRC for use at ANO-2, specifically:

1) "The ROCS and DIT Computer Codes for Nuclear Design", CENPD-266-P-A, April 1983 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.4 for MTC, 3.1.3.6 for Regulating and Group P CEA Insertion Limits, and 3.2.4.b for DNBR Margin).

2) "CE Method for Control Element Assembly Ejection Analysis," CENPD-0190-A, January 1976 (Methodology for Specification 3.1.3.6 for Regulating and Group P CEA Insertion Limits and 3.2.3 for Azimuthal Power Tilt).

3) "Modified Statistical Combination of Uncertainties, CEN-356(V)-P-A, Revision 01-P-A, May 1988 (Methodology for Specification 3.2.4.c and 3.2.4.d for DNBR Margin and 3.2.7 for ASI).

4) "Calculative Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P, August 1974 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

5) "Calculational Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P, Supplement 1, February 1975 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

6.6 REPORTING REQUIREMENTS

6.6.5 CORE OPERATING LIMITS REPORT (COLR)

6) "Calculational Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P, Supplement 2-P, July 1975 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

7) "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CEN-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

8) "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

9) "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137, Supplement 1-P, January 1977 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

10) "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137, Supplement 2-P-A, dated April, 1998 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

11) "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margin, 3.1.1.4 for MTC, 3.1.3.1 for CEA Position, 3.1.3.6 for Regulating CEA and Group P Insertion Limits, and 3.2.4.b for DNBR Margin).

12) "Technical Manual for the CENTS Code," CENPD 282-P-A, February 1991 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margin, 3.1.1.4 for MTC, 3.1.3.1 for CEA Position, 3.1.3.6 for Regulating and Group P Insertion Limits, and 3.2.4.b for DNBR Margin.

13) Letter: O.D. Parr (NRC) to F.M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for 6.6.5.b.4, 6.6.5.b.5, and 6.6.5.b.8 methodologies.

14) Letter: O.D. Parr (NRC) to A.E. Scherer (CE), dated December 9, 1975 (NRC Staff Review of the Proposed Combustion Engineering ECCS Evaluation Model changes). NRC approval for 6.6.5.b.6 methodology.

15) Letter: K. Kniel (NRC) to A.E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 6.6.5.b.9 methodology.

16) Letter: 2CNA038403, dated March 20, 1984, J.R. Miller (NRC) to J.M. Griffin (AP&L), "CESEC Code Verification." NRC approval for 6.6.5.b.11 methodology.

6.6 REPORTING REQUIREMENTS

6.6.5 CORE OPERATING LIMITS REPORT (COLR)

c. The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.6.6 Not Used

6.6.7 Steam Generator Tube Surveillance Reports

a. Following each inservice inspection of steam generator tubes the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be reported within 12 months following the completion of the inservice inspection. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report as denoted by Table 6.5.9-2. Notification of the Commission will be made prior to resumption of plant operation (i.e., prior to entering Mode 4). The written Special Report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

6.6.8 Specific Activity Analysis

The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded the results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history.

6.7 HIGH RADIATION

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

6.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP), or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

6.7 HIGH RADIATION

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

6.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

6.7 HIGH RADIATION

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP, or equivalent, while in the area by means of closed circuit television, or personnel qualified in radiation protection procedures responsible for controlling personnel radiation exposure in the area and with the means to communicate with individuals in the area who are covered by such surveillance.

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

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f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.