Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

May 15, 2001
LIC-01-0027

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Reference: Docket No. 50-285

SUBJECT: Application for Amendment of Facility Operating License No. DPR-40

Pursuant to 10 CFR 50.90, 50.91, and 50.4, the Omaha Public Power District (OPPD) is submitting this "Application for Amendment of Facility Operating License" to revise the Fort Calhoun Station Unit No. 1 Technical Specifications (TS). The significant changes of the proposed amendment substitutes generic titles for the Manager – Fort Calhoun Station and the Vice President, and relocates requirements for the Plant Review Committee (PRC), Safety Audit Review Committee (SARC), procedure approval, and records retention from the TS to the Fort Calhoun Station Quality Assurance (QA) Program.

The substitution of generic titles for the Manager – Fort Calhoun Station and the Vice President eliminates the need for a revision to the TS should the specific title of these positions change. Similarly, relocating PRC and SARC requirements to the QA Program eliminates the need for a revision to the TS should changes to these requirements be necessary. The Manager – Quality Assurance & Quality Control reviews changes to the QA Program for regulatory compliance, and final approval is by the Vice President. Any changes that could reduce the effectiveness of the QA Program must be approved by the NRC in accordance with 10 CFR 50.54(a)(4). This proposed amendment also includes updates to conform with revised regulatory requirements and Westinghouse/CENP definitions.

The proposed amendment will reduce the burden on NRC and OPPD resources by eliminating the need to process changes to these administrative TS requirements that do not impact nuclear safety.

Attachment A contains a markup reflecting the proposed changes to the Technical Specifications. Attachment B provides the "Discussion, Justification and No Significant Hazards Consideration."

A001

OPPD respectfully requests 30 days to implement the amendment following NRC approval. If you have any questions, please contact me.

Sincerely,

W. G. Gates
Vice President

WGG/rrl

Attachments

c: E. W. Merschoff, NRC Regional Administrator, Region IV
Alan Wang, NRC Project Manager
W. C. Walker, NRC Senior Resident Inspector
B. E. Casari, Director - Environmental Health Division,
State of Nebraska
Winston & Strawn

Before the United States
Nuclear Regulatory Commission

In the Matter of	)	
	)	
Omaha Public Power District	)	Docket No. 50-285
(Fort Calhoun Station	)	
Unit No. 1)	)	

APPLICATION FOR AMENDMENT OF FACILITY OPERATING LICENSE

Pursuant to Section 50.90 of the regulations of the U. S. Nuclear Regulatory Commission ("the Commission"), Omaha Public Power District, holder of Facility Operating License No. DPR-40, herewith requests that Technical Specifications set forth in Appendix A of the Facility Operating License be amended to revise TS 5.0 to utilize generic titles for the Manager – Fort Calhoun Station and the Vice President, and to relocate PRC and SARC requirements from TS 5.0 to the Quality Assurance Program. This proposed amendment also includes updates to conform with revised regulatory requirements and Westinghouse/CENP definitions.

Proposed changes to the Technical Specifications are provided in Attachment A to this Application. A Discussion, Justification, and No Significant Hazards Consideration Analysis, which demonstrates that the proposed changes do not involve significant hazards considerations, is appended in Attachment B. The proposed changes to Appendix A, Technical Specifications of the Facility Operating License, would not authorize any change in the types or any increase in the amounts of effluents or any change in the authorized power level of the facility.

WHEREFORE, Applicant respectfully requests that Appendix A of the Facility Operating License be amended hereto as Attachment A.

A copy of this Application, including its attachments, has been submitted to the Director - Nebraska State Division of Environmental Health, as required by 10 CFR 50.91.

OMAHA PUBLIC POWER DISTRICT

By

W. G. Gates

W. G. Gates
Vice President

Subscribed and sworn to before me
this 15th day of May, 2001

Patti Lounsberry

GENERAL NOTARY-State of Nebraska
PATTI LOUNSBERRY
My Comm. Exp. Nov. 8, 2004

Notary Public

United States of America
Nuclear Regulatory Commission

In the Matter of	)	
	)	
Omaha Public Power District	)	Docket No. 50-285
(Fort Calhoun Station	)	
Unit No. 1)	)	

AFFIDAVIT

W. G. Gates, being duly sworn, hereby deposes and says that he is the Vice President in charge of all nuclear activities of the Omaha Public Power District; that he is duly authorized to sign and file with the Nuclear Regulatory Commission the attached information concerning the application for Amendment of the Facility Operating License dated May 15, 2001, regarding generic titles for the Manager – Fort Calhoun Station, and the Vice President, the relocation of PRC and SARC requirements to the Fort Calhoun Station Quality Assurance Program, and updates to conform with revised regulatory requirements and Westinghouse/CENP definitions; that he is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge, information, and belief.

W. G. Gates

W. G. Gates
Vice President

STATE OF NEBRASKA)
) ss
COUNTY OF DOUGLAS)

Subscribed and sworn to me, a Notary Public in and for the State of Nebraska on this 15th day of May, 2001.

Patti Lounsberry

GENERAL NOTARY-State of Nebraska
PATTI LOUNSBERRY
My Comm. Exp. Nov. 8, 2004

Notary Public

Attachment A

Requested Changes of Technical Specifications set forth in Appendix A of Facility Operating License No. DPR-40

U.S. Nuclear Regulatory Commission
LIC-01-0027
Attachment A

Markup

Legend
Added
~~Deleted~~

TECHNICAL SPECIFICATION

DEFINITIONS

Azimuthal Power Tilt - T_q

Azimuthal Power Tilt shall be the ~~maximum difference between the power generated in any core quadrant (upper or lower) and the average power of all quadrants in that axial half (upper or lower) of the core divided by the average power of all quadrants in that axial half (upper or lower) of the core.~~ power asymmetry between azimuthally symmetric fuel assemblies.

Unrodded Integrated Radial Peaking Factor - F_R

The Unrodded Integrated Radial Peaking Factor is the ratio of the peak pin power to the average pin power in an unrodded core, excluding azimuthal tilt, T_q. The maximum F_R limit is provided in the Core Operating Limits Report.

Process Control Program (PCP)

The document(s) that contains the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 61, 71, State Regulations, burial ground requirements, and other requirements governing the disposal of solid waste.

Dose Equivalent I-131

That concentration of I-131 (μCi/gm) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. In other words,

$$
\begin{aligned}
\text{Dose Equivalent I-131 } (\mu\text{Ci/gm}) \quad = \quad & \mu\text{Ci/gm of I-131} \\
+ \quad & 0.0361 \times \mu\text{Ci/gm of I-132} \\
+ \quad & 0.270 \times \mu\text{Ci/gm of I-133} \\
+ \quad & 0.0169 \times \mu\text{Ci/gm of I-134} \\
+ \quad & 0.0838 \times \mu\text{Ci/gm of I-135}
\end{aligned}
$$

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.4 Power Distribution Limits (Continued)

In order for these objectives to be met, the reactor must be operated consistent with the operating limits specified for margin to DNB.

The parameter limits given in (5) and the F_R^T, and Core Power Limitations Figure provided in the COLR along with the parameter limits on quadrant tilt and control element assembly position (Power Dependent Insertion Limit Figure provided in the COLR) provide a high degree of assurance that the DNB overpower margin will be maintained during steady state operation.

The actions specified assure that the reactor is brought to a safe condition.

The Reactor Coolant System flow rate of 206,000 gallons per minute is the indicated value. It does not include instrumentation uncertainties.

The calorimetric methodology shall be used to measure the Reactor Coolant System flow rate.

AZIMUTHAL POWER TILT

Azimuthal Power Tilt is measured using symmetric in-core or ex-core detectors by assuming that the ratio of the power at any core location in the presence of a tilt to the untilted power at that location is of the form:

$$P_{tilt}(r, \theta) / P_{avg}(r, \theta) - 1 = T_q \bullet g(r) \bullet \cos(\theta - \theta_0)$$

where:

$P_{tilt}(r, \theta)$ is the tilted power at radius r and azimuthal angle θ
$P_{avg}(r, \theta)$ is the average or untilted power at that location
T_q is the azimuthal tilt magnitude
$g(r)$ is the radial normalizing factor, normalized to a maximum value of unity
θ is the azimuthal core location
θ_0 is the azimuthal core location of maximum tilt."

T_q represents the maximum fractional increase in power that can occur anywhere in the core because of tilt. It is the appropriate measured value of tilt to be used when ensuring the validity of the azimuthal tilt assumed by ABB-CE in establishing safety limits.

5.0 **ADMINISTRATIVE CONTROLS**

5.1 Responsibility

5.1.1 The ~~Manager - Fort Calhoun Station~~ plant manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

5.2 Organization

5.2.1 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 for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the USAR.

b. The ~~Manager - Fort Calhoun Station~~ plant manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The ~~Vice President - shall have~~ corporate officer with responsibility for overall plant 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 to ensure nuclear safety.

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

5.2.2 Plant Staff

The plant staff organization shall be as described in Chapter 12 of the USAR and shall function as follows:

a. The minimum number and type of licensed and unlicensed operating personnel required onsite for each shift shall be as shown in Table 5.2-1.

5.0 **ADMINISTRATIVE CONTROLS**

5.2 Organization (Continued)

b. An Operator or Technician qualified in Radiation Protection Procedures shall be onsite when fuel is in the reactor.

c. 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 the operation.

d. Fire protection program responsibilities are assigned to those positions and/or groups designated by asterisks in USAR 12.1-1 through 12.1-4 according to the procedures specified in ~~Section 5.8 of the Technical Specifications~~ the Quality Assurance Program.

e. Administrative procedures shall be developed and implemented to limit the working hours of plant staff who perform safety-related functions. Administrative procedures shall reflect the personnel whose working hours will be affected. Shift coverage shall be maintained without routine heavy use of overtime.

Deviations from the guidelines shall be authorized in advance by the Department Manager, plant manager ~~Manager - Fort Calhoun Station~~, or their designated alternates, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Routine deviation from the administrative guidelines shall not be authorized.

Controls shall be included in the procedures such that individual overtime shall be reviewed ~~monthly~~ periodically by the Department Manager, plant manager ~~Manager - Fort Calhoun Station~~, or their designated alternates, or higher levels of management, to ensure that excessive hours have not been assigned.

f. The ~~Supervisor -~~ Manager - Shift Operations, the Shift Managers, and the Control Room Supervisors ~~Licensed Senior Operators~~ shall hold a senior reactor operator license. The Licensed Operators shall hold a reactor operator license.

5.0 **ADMINISTRATIVE CONTROLS**

5.3 Facility Staff Qualification

5.3.1 Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, with the exception of the Manager - Radiation Protection (MRP), ~~and~~ the Shift Technical Advisors (STA), the senior reactor operator licensees, and the reactor operator licensees, who shall meet ~~. The MRP shall meet~~ the requirements set forth in Regulatory Guide 1.8, Revision 3, dated ~~September~~ May ~~1975~~ 2000, entitled ~~"Personnel Selection and Training"~~ "Qualification and Training of Personnel for Nuclear Power Plants." ~~The MRP is considered to meet the educational and experience qualifications set forth in Regulatory Guide 1.8 with at least five years of experience in applied radiation protection and extensive formal training in radiation protection. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant for transients and accidents.~~

FIGURE 5-1

This Figure has been deleted

FIGURE 5-2

This Figure has been deleted

TABLE 5.2-1

MINIMUM SHIFT CREW COMPOSITION(ii)

License Category	Core Alteration	Cold Shutdown or Refueling Shutdown	Operating or Hot Shutdown Modes
Senior Operator License	2~~1~~(i)	1	2(iii)
Operator License	2	1	2(iv)
Non-Licensed	(As required)	1	2
Shift Technical Advisor	None	None	1

(i) This ~~Does not~~ includes the individual with Senior Operator License supervising ~~Refueling Operations~~ Core Alterations.

(ii) Shift crew composition may be one 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 5.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift ~~crewman~~ crewmember being late or absent.

(iii) At least one of these individuals must be in the control room at all times.

(iv) At least one of these individuals (or the second senior licensed operator, if both senior licensed operators are in the control room) must be present at the controls at all times.

5.0 **ADMINISTRATIVE CONTROLS**

5.4 Training

5.4.1 A retraining and replacement training program for the plant staff shall be maintained under the direction of the Manager - Training and shall meet or exceed the requirements of Section ~~5.5~~ 6 of ANSI/ANS 3.1-1993, as modified by Regulatory Guide 1.8, Revision 3, dated May 2000 ~~ANSI N18.1-1971~~ and 10 CFR Part 55.

5.5 Review and Audit

5.5.1 Plant Review Committee (PRC)

~~Function~~

~~5.5.1.1 The Plant Review Committee shall function to advise the Manager - Fort Calhoun Station on all matters related to nuclear safety.~~ A committee composed of key management personnel designated as the PRC acts in an advisory capacity on all matters related to nuclear safety to the plant manager and serves in accordance with Quality Assurance Program requirements, USAR Section 12.5, and plant Standing Orders.

~~Composition~~

~~5.5.1.2 The official Plant Review Committee shall consist of at least six but not more than eleven members and shall be composed of the:~~

~~Chairman: Manager - Fort Calhoun Station~~

~~Members: The members shall be Department Heads or supervisory staff representing operations, maintenance, engineering, chemistry, radiation protection and other technical disciplines as determined by the Chairman.~~

~~All members shall be qualified to the applicable requirements of Specification 5.3 prior to being appointed by the Chairman.~~

~~Alternates~~

~~5.5.1.3 Alternate members shall be appointed in writing by the Plant Review Committee Chairman to serve on a temporary basis.~~

~~Meeting Frequency~~

~~5.5.1.4 The Plant Review Committee shall meet at least once per calendar month and as convened by the Plant Review Committee Chairman.~~

~~Quorum~~

~~5.5.1.5 A quorum of the Plant Review Committee shall consist of the Chairman or Alternate Chairman and a majority of members including alternates. At any one time, no more than a minority of the quorum shall consist of alternate members participating as voting members in Plant Review Committee activities.~~

5.0 **ADMINISTRATIVE CONTROLS**

~~Responsibilities~~

~~5.5.1.6 The Plant Review Committee shall be responsible for:~~

~~a. Review of (1) Administrative Controls Standing Orders and changes thereto, (2) procedures required by Specification 5.8 and requiring a 10 CFR 50.59 safety evaluation, and (3) proposed changes to procedures required by Specification 5.8 and requiring a 10 CFR 50.59 safety evaluation;~~

~~b. Review of all proposed tests and experiments that affect nuclear safety.~~

~~c. Review of all proposed changes to the Technical Specifications.~~

~~d. Review of all proposed changes to the Core Operating Limits Report.~~

~~e. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.~~

~~f. Investigation of all violations of the Technical Specifications and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Vice President and to the Chairperson of the Safety Audit and Review Committee.~~

~~g. Review of facility operations to detect potential safety hazards.~~

~~h. Performance of special reviews and investigations and reports thereon as requested by the Chairperson of the Safety Audit and Review Committee.~~

~~i. DELETED~~

~~j. DELETED~~

~~k. Review of the Fire Protection Program Plan and shall submit changes to the Chairperson of the Safety Audit and Review Committee.~~

~~l. Review of all Reportable Events.~~

~~Authority~~

~~5.5.1.7 The Plant Review Committee shall:~~

~~a. Recommend in writing to the Manager - Fort Calhoun Station approval or disapproval of items considered under 5.5.1.6(a) through (e) above.~~

5.0 ADMINISTRATIVE CONTROLS

~~5.5.1.7 b. Render determinations in writing with regard to whether or not each item considered under 5.5.1.6(b) through (f) above constitutes an unreviewed safety question.~~

~~c. Provide immediate written notification to the Vice President and the Chairperson of the Safety Audit and Review Committee of disagreement between the Plant Review Committee and the Manager - Fort Calhoun Station; however, the Manager - Fort Calhoun Station shall have responsibility for resolution of such disagreements pursuant to 5.1.1 above.~~

~~Records~~

~~5.5.1.8 The Plant Review Committee shall maintain written minutes of each meeting and copies shall be provided to the Vice President and Chairperson of the Safety Audit and Review Committee.~~

5.5.2 Safety Audit and Review Committee (SARC)

~~Function~~

~~5.5.2.1 The SARC shall function to provide the independent review and audit of designated activities in the areas of:~~ The Safety Audit and Review Committee (SARC) is a committee composed of highly qualified and experienced OPPD management personnel and consultants, which functions to provide independent review and audit of activities in accordance with the Quality Assurance Program requirements, USAR Section 12.5, and the SARC Charter. The SARC reports to and advises the corporate officer responsible for overall plant nuclear safety.

~~a. nuclear power plant operation~~
~~b. nuclear engineering~~
~~c. chemistry and radiochemistry~~
~~d. metallurgy~~
~~e. instrumentation and control~~
~~f. radiological safety~~
~~g. mechanical and electrical engineering~~
~~h. quality assurance~~
~~i. fire protection~~

~~Composition~~

~~5.5.2.2 The Safety Audit and Review Committee shall be composed of:~~

~~Chairperson:~~	~~Member as appointed by the Vice President~~
~~Member:~~	~~Vice President~~
~~Member:~~	~~Division Manager - Nuclear Assessments~~
~~Member:~~	~~Division Manager - Engineering & Operations Support~~
~~Member:~~	~~Manager - Fort Calhoun Station~~
~~Member:~~	~~Other qualified OPPD personnel and/or consultants as required and as determined by the SARC Chairperson~~

5.0 ADMINISTRATIVE CONTROLS

~~Alternates~~

~~5.5.2.3 Alternate members shall be appointed in writing by the Chairperson of the Safety Audit and Review Committee to serve on a temporary basis; however, no more than two alternates may participate in the Safety Audit and Review Committee activities at any one time.~~

~~Consultants~~

~~5.5.2.4 Consultants shall be utilized as determined by the Safety Audit and Review Committee Chairperson to provide expert advice to the Safety Audit and Review Committee.~~

~~Meeting Frequency~~

~~5.5.2.5 The Safety Audit and Review Committee shall meet at least once every six months.~~

~~Quorum~~

~~5.5.2.6 A quorum of the Safety Audit and Review Committee shall consist of the Chairperson or his designated alternate and a majority of the Safety Audit and Review Committee members including alternates. No more than a minority of the quorum shall have line responsibility for the operation of the nuclear plant.~~

~~Review~~

~~5.5.2.7 The Safety Audit and Review Committee shall review:~~

~~a. The safety evaluations for 1) procedures, equipment or systems and 2) tests or experiments completed under the provision of section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.~~

~~b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in section 50.59, 10 CFR.~~

5.0 ADMINISTRATIVE CONTROLS

~~5.5.2.7 c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.~~

~~d. Proposed changes to Technical Specifications and Facility Operating License DPR-40.~~

~~e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.~~

~~f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.~~

~~g. All Licensee Event Reports required by 10 CFR 50.73.~~

~~h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.~~

~~i. Reports and meeting minutes of the Plant Review Committee.~~

~~The Chairperson of the Safety Audit and Review Committee (SARC) may designate subgroups, special working committees, or audit teams as he deems necessary in order to carry out the responsibilities of the SARC. These subgroups, committees, or audit teams will perform the SARC responsibilities and report on their activities for review at the next regularly scheduled SARC meeting following any group's action.~~

~~Audit~~

~~5.5.2.8 Audits of facility activities shall be performed under the cognizance of the Safety Audit and Review Committee. These audits shall encompass:~~

~~a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions.~~

~~b. The training and qualifications of the facility staff.~~

~~c. Actions taken to correct deficiencies occurring in facility equipment, structures, systems, components or method of operation that affect nuclear safety.~~

~~d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50.~~

5.0 **ADMINISTRATIVE CONTROLS**

~~5.5.2.8 e. DELETED~~

~~f. DELETED~~

~~g. DELETED~~

~~h. The Radiological Effluent Program including the Radiological Environmental Monitoring Program and the results thereof, the Offsite Dose Calculation Manual and implementing procedures, and the Process Control Program for the solidifications of radioactive waste.~~

~~i. The fire protection and loss prevention program utilizing either qualified off-site licensee personnel or an outside fire protection consultant.~~

~~j. Any other area of facility operation considered appropriate by the Safety Audit and Review Committee or the Vice President.~~

~~Authority~~

~~5.5.2.9 The Safety Audit and Review Committee shall report to and advise the Vice President on those areas of responsibility specified in Sections 5.5.2.7 and 5.5.2.8.~~

~~Records~~

~~5.5.2.10 Records of Safety Audit and Review Committee activities shall be prepared, approved and distributed as indicated below:~~

~~a. Minutes of each Safety Audit and Review Committee meeting shall be prepared, approved and forwarded to the Vice President within 30 days following each meeting.~~

~~b. Reports of reviews encompassed by Section 5.5.2.7e, f, g, h, and i above shall be prepared, approved and forwarded to the Vice President within 30 days following completion of the review.~~

~~c. Audit reports encompassed by Section 5.5.2.8 above shall be forwarded to the Vice President and to the responsible management positions designated by the Safety Audit and Review Committee within 30 days after completion of the audit.~~

DELETED

5.0 **ADMINISTRATIVE CONTROLS**

5.6 Reportable Event Action

5.6.1 The following actions shall be taken in the event of a REPORTABLE EVENT:

a. The Commission shall be notified pursuant to the requirements of 10 CFR 50.72, if applicable.

b. Each Reportable Event shall be reviewed by the Plant Review Committee and submitted to the Chairperson of the Safety Audit and Review Committee and the ~~Vice President~~ corporate officer responsible for overall plant nuclear safety.

c. Submit reports of Reportable Events pursuant to the requirements of Specification 5.9.2.

5.7 Safety Limit Violation

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

a. The unit shall be placed in at least HOT SHUTDOWN within 1 hour.

b. The Safety Limit Violations shall be reported to the ~~Vice President~~ corporate officer responsible for overall plant nuclear safety and the Chairperson of the Safety Audit and Review Committee (SARC) within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Plant Review Committee. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Chairperson of the Safety Audit and Review Committee and the ~~Vice President~~ corporate officer responsible for overall plant nuclear safety within 14 days of the violation.

5.8 Procedures

5.8.1 Written procedures and administrative policies shall be established, implemented and maintained in accordance with the Quality Assurance Program. ~~that meet or exceed the minimum requirements of sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix A of USNRC Regulatory Guide 1.33 except as provided in 5.8.2 and 5.8.3 below.~~

~~5.8.2 Each procedure of Specification 5.8.1, and changes thereto, and any other procedure or procedure change that the Manager - Fort Calhoun Station determines to affect nuclear safety, shall be reviewed and approved as described below, prior to implementation.~~

5.0 **ADMINISTRATIVE CONTROLS**

~~5.8.2.1 Each procedure, or change thereto, shall be reviewed by a Qualified Reviewer (QR) who is knowledgeable in the functional area affected but is not the individual preparer. The QR may be from the same line-organization as the preparer. The QR shall render a determination in writing of whether or not cross-disciplinary review of a procedure, or change thereto is necessary. If necessary, such review shall be performed by appropriate personnel.~~

~~5.8.2.2 Each procedure, or change thereto, shall be reviewed by the Department Head designated by Administrative Controls Standing Orders as the responsible Department Head for that procedure, and the review shall include a determination of whether or not a 10 CFR 50.59 safety evaluation is required. If a 10 CFR 50.59 safety evaluation is not required, the procedure, or change thereto, shall be approved by the responsible Department Head or the Manager-Fort Calhoun Station, prior to implementation. Administrative Controls Standing Orders, and the Fire Protection Program Plan shall be reviewed in accordance with Specification 5.5.1.6 and approved by the Manager-Fort Calhoun Station.~~

~~5.8.2.3 If the responsible Department Head determines that a procedure, or change thereto, requires a 10 CFR 50.59 safety evaluation, the responsible Department Head shall render a determination in writing of whether or not the procedure, or change thereto, involves an Unreviewed Safety Question (USQ) and shall forward the procedure, or change thereto with the associated safety evaluation to the PRC for review in accordance with Specification 5.5.1.6.a. If a USQ is involved, NRC approval is required prior to implementation of the procedure, or change.~~

~~5.8.2.4 Qualified Reviewers shall meet or exceed the respective qualifications for either Supervisors Requiring an AEC License, Professional-Technical Personnel, or Technical Support Personnel, as specified in ANSI N18.1 - 1971. Personnel recommended to be QRs shall be reviewed by the PRC and approved and designated as such by the PRC Chairman. The responsible Department Head shall ensure that a sufficient complement of QRs for their functional area is maintained in accordance with Administrative Controls Standing Orders.~~

~~5.8.2.5 Each procedure of Specification 5.8.1 shall be reviewed periodically as set forth in Administrative Controls Standing Orders.~~

~~5.8.2.6 Records documenting the activities performed under Specifications 5.8.2.1 through 5.8.2.4 shall be maintained in accordance with Specification 5.10.~~

5.0 ADMINISTRATIVE CONTROLS

~~5.8.3 Temporary changes to procedures of 5.8.1 above may be made provided:~~

~~a. The intent of the original procedure is not altered.~~

~~b. The change is approved by two members of the plant supervisory staff, at least one of whom holds a Senior Reactor Operator's License.~~

~~c. The change is documented, reviewed by a Qualified Reviewer and approved by either the Manager - Fort Calhoun Station or the Department Head designated by Administrative Controls Standing Orders as the responsible Department Head for that procedure within 14 days of implementation.~~

~~5.8.4 Written procedures approved per 5.8.2 above shall be implemented which govern the selection of fuel assemblies to be placed in Region 2 of the spent fuel racks (Technical Specification 2.8). These procedures shall require an independent verification of initial enrichment requirements and fuel burnup calculations for a fuel bundle to assure the "acceptance" criteria for placement in Region 2 are met. This independent verification shall be performed by individuals or groups other than those who performed the initial acceptance criteria assessment, but who may be from the same organization.~~

~~5.8.5 Written procedures shall be established and maintained for implementation of the Fire Protection Program.~~

5.9 Reporting Requirements

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the ~~Director of the~~ appropriate NRC Regional Office ~~of Inspection and Enforcement~~ unless otherwise noted.

5.9.1 Routine Reports

a. Startup Report. 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 manufacture by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the USAR and shall in general 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.

5.0 ADMINISTRATIVE CONTROLS

5.9.1 Continued

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.

b. Annual Occupational Exposure Report. An annual occupational exposure report shall be submitted on or before April 30 of each year. The report shall consist of a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to

5.9.1 Continued

work and job functions,[3/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling outages. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

c. Monthly Operating Report. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, ~~Mail Station Pl-137, Washington, D. C. 20555,~~ with a copy to the appropriate Regional Office, no later than the fifteenth of each month following the calendar month covered by the report. This monthly report shall also include a statement regarding any challenges or failures to the pressurizer power operated relief valves or safety valves occurring during the subject month.

5.9.2 Reportable Event

A Licensee Event Report (LER) shall be submitted to the U.S. Nuclear Regulatory Commission~~, Document Control Desk, Mail Station P1-137, Washington, D. C. 20555 with a copy to Region IV of the NRC, within 30 days after discovery of~~ for any event meeting the requirements of 10 CFR Part 50.73.

3/ This tabulation supplements the requirements of § 20.2206 of 10 CFR Part 20.

5.0 ADMINISTRATIVE CONTROLS

5.9.3 Special Reports

Special reports shall be submitted to the ~~Regional Administrator of the~~ appropriate NRC 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 where appropriate:

a. In-service inspection report, reference 3.3.
b. Tendon surveillance, reference 3.5.
c. Containment structural tests, reference 3.5.
d. ~~Special maintenance reports~~. DELETED
e. DELETED
f. DELETED
g. Materials radiation surveillance specimens reports, reference 3.3.
h. DELETED
i. Post-accident monitoring instrumentation, reference 2.21
j. Electrical systems, reference 2.7(2).

5.9.4 Unique Reporting Requirements

a. Annual Radioactive Effluent Release Report

The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year of operation shall be submitted before May 1 of each year. 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 1) consistent with the objectives outlined in the ODCM and PCP, and 2) in conformance with 10 CFR 50.36a. and Section IV.B.1 of Appendix I to 10 CFR 50.

b. Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis 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 (1) the ODCM and (2) Section IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR 50.

c. Fire Protection Program Deficiency Report

Deficiencies in the Fire Protection Program described in the Updated Safety Analysis Report which meet the reportability criteria of 10 CFR 50.73 shall be reported pursuant to Section 5.9.2 of the Technical Specifications.

5.0 **ADMINISTRATIVE CONTROLS**

5.9.5 Core Operating Limits Report

a. Core Operating Limits shall be established and documented in the Core Operating Limits Report (COLR) before each reload cycle or any remaining part of a reload cycle.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC as follows:

1. OPPD-NA-8301-P-A, "Reload Core Analysis Methodology Overview" approved version as specified in the COLR.

2. OPPD-NA-8302-P-A, "Neutronics Design Methods and Verification", approved version as specified in the COLR.

3. OPPD-NA-8303-P-A, "Transient and Accident Methods and Verification", approved version as specified in the COLR.

4. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Report," April 1995 (Westinghouse Proprietary) as approved in the Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 178 to Facility Operating License No. DPR-40, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Docket No. 50-285, dated October 25, 1996.

5. WCAP-13027-P, "Westinghouse ECCS Evaluation Model for Analysis of CE-NSSS," July 1991 (Westinghouse Proprietary) as approved in the Safety Evaluation by the Office of Nuclear Reactor Regulation dated March 26, 1992, and as applied in OPPD submittal to the NRC (LIC-96-0130) dated September 3, 1996, and as approved in the Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 178 to Facility Operating License No. DPR-40, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Docket No. 50-285, dated October 25, 1996.

6. XN-75-32(P)(A) Supplements 1, 2, 3 & 4, "Computational Procedure for Evaluating Fuel Rod Bowing," approved version as specified in the COLR.

7. XN-NF-82-06(P)(A) and Supplements 2, 4 and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.

8. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.

5.0 ADMINISTRATIVE CONTROLS

5.9.5 Core Operating Limits Report

9. ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU," approved version as specified in the COLR.

10. EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.

11. XN-NF-78-44(P)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," approved version as specified in the COLR.

12. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," approved version as specified in the COLR.

13. EMF-1961(P)(A), "Statistical Setpoint/Transient Methodology for CE Reactors, Siemens Power Corporation," approved version as specified in the COLR.

14. XN-NF-621(P)(A), "Exxon Nuclear DNB Correlation for PWR Fuel Designs," approved version as specified in the COLR.

15. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," approved version as specified in the COLR.

16. EMF-92-153(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.

17. XN-NF-82-49(P)(A), Supplement 1, "Exxon Nuclear Company Evaluation Model Revised EXEM PWR Small Break Model," approved version as specified in the COLR.

18. EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," approved version as specified in the COLR.

19. ANF-84-73 Appendix B (P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuels Corporation, approved version as specified in the COLR.

20. EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs," Siemens Power Corporation, approved version as specified in the COLR.

5.0 ADMINISTRATIVE CONTROLS

c. The core operating limits shall be determined so that all applicable limits of the safety analysis are met. The Core Operating Limits Report, 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 Region IV Administrator and Senior Resident Inspector.

5.10 Records Retention

5.10.1 Records shall be retained as described in the Quality Assurance Program. ~~The following records shall be retained for at least five years:~~

~~a. Records, and logs of facility operation covering time interval at each power level.~~

~~b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.~~

~~c. Licensee Event Reports (LER).~~

~~d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.~~

~~e. Records of reactor tests and experiments.~~

~~f. Records of changes made to Operating Procedures.~~

~~g. DELETED.~~

~~h. Records of annual physical inventory of all source material of record.~~

5.0 ADMINISTRATIVE CONTROLS

~~5.10.2 The following records shall be retained for the duration of the Facility Operating License:~~

~~a. Records of drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.~~
~~b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.~~
~~c. Records of facility radiation and contamination surveys.~~
~~d. Records of radiation exposure for all individuals entering radiation control areas.~~
~~e. Records of gaseous and liquid radioactive material released to the environs.~~
~~f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.~~
~~g. Records of training and qualification for current members of the plant staff.~~
~~h. Records of in-service inspections performed pursuant to these Technical Specifications.~~
~~i. Records of Quality Assurance activities required by the QA Manual.~~
~~j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.~~
~~k. Records of meetings of the Plant Review Committee and the Safety Audit and Review Committee.~~
~~l. Records of Environmental Qualification of Electric Equipment pursuant to 10 CFR 50.49.~~
~~m. Records of the service lives of all hydraulic and mechanical snubbers, including the date at which the service life commences and associated installation and maintenance records.~~
~~n. Records of analyses required by the Radiological Environmental Monitoring Program.~~
~~o. Records of reviews performed for changes made to the Offsite Dose Calculation Manual and the Process Control Program.~~
~~p. Records of radioactive shipments.~~

~~5.10.3 A complete record of the analysis employed in the selection of any fuel assembly to be placed in Region 2 of the spent fuel racks will be retained as long as that assembly remains in Region 2 (reference Technical Specifications 2.8 and 4.4).~~

5.11 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.

5.0 ADMINISTRATIVE CONTROLS

5.11.1 In lieu of the "control device" required by paragraph 20.1601(a) of 10 CFR Part 20, and as an alternative method allowed under § 20.1601(c), each high radiation area (as defined in § 20.1601) 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 required 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 by the Manager-Radiation Protection (MRP) in the Radiation Work Permit.

5.11.2 The requirements of 5.11.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr** but less than 500 rads/hr*** (Restricted High Radiation Area). In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Manager on duty and/or the MRP with the following exception:

a. In lieu of the above, for accessible localized Restricted High Radiation Areas located in large areas such as containment, where no lockable enclosure exists in the immediate vicinity to control access to the Restricted High Radiation Area and no such enclosure can be readily constructed, then the Restricted High Radiation Area shall be:

   i. roped off such that an individual at the rope boundary is exposed to 1000 mrem/hr or less,
   ii conspicuously posted, and
   iii a flashing light shall be activated as a warning device.

*Radiation Protection personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.
**At 30 centimeters (12 inches) from the radiation source or from any surface penetrated by the radiation.
***At 1 meter from the radiation source or from any surface penetrated by the radiation.

5.0 **ADMINISTRATIVE CONTROLS**

5.12 Environmental Qualification

Deleted

5.13 Secondary Water Chemistry

A secondary water chemistry monitoring program to inhibit steam generator tube degradation shall be implemented. This program shall be described in the station chemistry manual and shall include:

1. Identification of a 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. Procedures for the recording and management of data;
5. Procedures defining corrective actions for off control point chemistry conditions; and
6. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective actions.

5.0 ADMINISTRATIVE CONTROLS

5.14 Systems Integrity

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 shall be implemented. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and
2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

5.15 Post-Accident Radiological Sampling and Monitoring

The following programs shall be implemented and maintained to ensure the capability to accurately monitor and/or sample and analyze radiological effluents and concentrations in a post-accident condition:

1. A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. (Any space which will require occupancy to permit an operator to aid in mitigation of, or recovery from, an accident is designated as vital.)
2. A program which will ensure the capability to obtain and analyze radioactive iodines and particulates in plant gaseous effluents.
3. A program which will ensure the capability to obtain and analyze a reactor coolant liquid sample under accident conditions.
4. A program which will ensure the capability to obtain and analyze a containment atmosphere sample under accident conditions.

These programs shall include the following:

1. Training of personnel.
2. Procedures for monitoring and/or sampling and analysis.
3. Provisions for maintenance of sampling and analysis equipment.

5.0 **ADMINISTRATIVE CONTROLS**

5.16 Radiological Effluents and Environmental Monitoring Programs

The following programs shall be established, implemented, and maintained.

5.16.1 Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for control of radioactive effluents and for maintaining the doses to individuals in unrestricted areas from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the operability of radioactive liquid and gaseous radiation monitoring instrumentation including operability tests and setpoint determination in accordance with the methodology in the ODCM.

b. Limitations on the concentration of radioactive material, other than dissolved or entrained noble gases, released in liquid effluents to unrestricted areas conforming to ten times 10 CFR 20.1001-20.2401, Appendix B, Table 2, Column 2. For dissolved or entrained noble gases, the concentration shall be limited to 2.0 E-04 μCi/ml total activity.

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 individuals in unrestricted areas from radioactive materials in liquid effluents released to unrestricted areas conforming to Appendix I to 10 CFR Part 50.

e. Determination of cumulative doses from radioactive effluents for the current calendar quarter and current calendar year in accordance with the ODCM on a quarterly basis.

f. Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity in plant effluents.

5.16 Radiological Effluents and Environmental Monitoring Programs (continued)

g. Limitations on the concentration resulting from radioactive material, other than noble gases, released in gaseous effluents to unrestricted areas conforming to ten times 10 CFR 20.1001-20.2401, Appendix B, Table 2, Column 1. For noble gases. the concentration shall be limited to five times 10 CFR 20.1001-20.2401, Appendix B, Table 2, Column 1.

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to unrestricted areas conforming to Appendix I to 10 CFR Part 50.

i. Limitations on the annual and quarterly doses to an individual beyond the site boundary from Iodine-131, tritium, and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released to unrestricted areas conforming to Appendix I to 10 CFR Part 50.

j. Limitations on the annual dose or dose commitment to an individual beyond the site boundary due to releases or radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

5.16.2 Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

b. A Land Use Census to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the monitoring program are made if required by the results of this census.

c. Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

5.0 **ADMINISTRATIVE CONTROLS**

5.17 Offsite Dose Calculation Manual (ODCM)

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by the Quality Assurance Program. ~~Specification 5.10.2.o.~~ This documentation shall contain:

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

2. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the Plant Review Committee and the approval of the plant manager ~~Manager - Fort Calhoun Station~~.

c. Temporary changes to the ODCM may be made in accordance with the Quality Assurance Program. ~~Technical Specification 5.8.3.~~

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

5.0 **ADMINISTRATIVE CONTROLS**

5.18 Process Control Program (PCP)

Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by the Quality Assurance Program. ~~Specification 5.10.2.o.~~ This documentation shall contain:

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

   2. A determination that the change will maintain the overall conformance of the solidified waste program to existing requirements of federal, state, or other applicable regulations.

b. Shall become effective after the review and acceptance by the Plant Review Committee and the approval of the plant manager ~~Manager - Fort Calhoun Station~~.

c. Temporary changes to the PCP may be made in accordance with the Quality Assurance Program. ~~Technical Specification 5.8.3.~~

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

5.0 ADMINISTRATIVE CONTROLS

5.19 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," as modified by the following exceptions:

(1) If the Personnel Air Lock (PAL) is opened during periods when containment integrity is not required, the PAL door seals shall be tested at the end of such periods and the entire PAL shall be tested within 14 days after RCS temperature T_{cold} > 210°F.

(2) Type A tests may be deferred for penetrations of the steel pressure retaining boundary where the nominal diameter does not exceed one inch.

(3) Elapsed time between consecutive Type A tests used to determine performance shall be at least 24 months or refueling interval.

The containment design accident pressure (P_a) is 60 psig.

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

Leakage Rate acceptance criteria are:

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

b. Personnel Air Lock testing acceptance criteria are:

   (1) Overall Personnel Air Lock leakage is $\leq$ 0.1 L_a when tested at $\geq$ P_a.

   (2) For each PAL door, seal leakage rate is $\leq$ 0.01 L_a when pressurized to $\geq$ 5.0 psig.

c. Containment Purge Valve (PCV-742A/B/C/D) testing acceptance criterion is:

   For each Containment Purge Valve, leakage rate is < 18.000 SCCM when tested at $\geq$ P_a.

d. If at any time when containment integrity is required and the total Type B and C measured leakage rate exceeds 0.60 L_a Minimum Pathway Leakage Rate (MNLPR), repairs shall be initiated immediately. If repairs and retesting fail to demonstrate conformance to this acceptance criteria within 48 hours, then containment shall be declared inoperable.

5.0 ADMINISTRATIVE CONTROLS

5.19 Containment Leakage Rate Testing Program (continued)

The provisions of Specification 3.0.1 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 3.0.4 are applicable to the Containment Leakage Rate Testing Program.

Final Draft

TECHNICAL SPECIFICATION

DEFINITIONS

Azimuthal Power Tilt - T_q

Azimuthal Power Tilt shall be the power asymmetry between azimuthally symmetric fuel assemblies.

Unrodded Integrated Radial Peaking Factor - F_R

The Unrodded Integrated Radial Peaking Factor is the ratio of the peak pin power to the average pin power in an unrodded core, excluding azimuthal tilt, T_q. The maximum F_R limit is provided in the Core Operating Limits Report.

Process Control Program (PCP)

The document(s) that contains the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 61, 71, State Regulations, burial ground requirements, and other requirements governing the disposal of solid waste.

Dose Equivalent I-131

That concentration of I-131 (μCi/gm) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. In other words,

Dose Equivalent I-131 (μCi/gm)	=	μCi/gm of I-131
	+	0.0361 x μCi/gm of I-132
	+	0.270 x μCi/gm of I-133
	+	0.0169 x μCi/gm of I-134
	+	0.0838 x μCi/gm of I-135

2.0 LIMITING CONDITIONS FOR OPERATION
2.10 Reactor Core (Continued)
2.10.4 Power Distribution Limits (Continued)

In order for these objectives to be met, the reactor must be operated consistent with the operating limits specified for margin to DNB.

The parameter limits given in (5) and the F_R^T, and Core Power Limitations Figure provided in the COLR along with the parameter limits on quadrant tilt and control element assembly position (Power Dependent Insertion Limit Figure provided in the COLR) provide a high degree of assurance that the DNB overpower margin will be maintained during steady state operation.

The actions specified assure that the reactor is brought to a safe condition.

The Reactor Coolant System flow rate of 206,000 gallons per minute is the indicated value. It does not include instrumentation uncertainties.

The calorimetric methodology shall be used to measure the Reactor Coolant System flow rate.

AZIMUTHAL POWER TILT

Azimuthal Power Tilt is measured using symmetric in-core or ex-core detectors by assuming that the ratio of the power at any core location in the presence of a tilt to the untilted power at that location is of the form:

$$P_{tilt}(r, \theta) / P_{avg}(r, \theta) - 1 = T_q \bullet g(r) \bullet \cos(\theta - \theta_0)$$

where:

$P_{tilt}(r, \theta)$	is the tilted power at radius r and azimuthal angle θ
$P_{avg}(r, \theta)$	is the average or untilted power at that location
T_q	is the azimuthal tilt magnitude
$g(r)$	is the radial normalizing factor, normalized to a maximum value of unity
θ	is the azimuthal core location
θ_0	is the azimuthal core location of maximum tilt."

T_q represents the maximum fractional increase in power that can occur anywhere in the core because of tilt. It is the appropriate measured value of tilt to be used when ensuring the validity of the azimuthal tilt assumed by ABB-CE in establishing safety limits.

(next page is 2-59)

5.0 **ADMINISTRATIVE CONTROLS**

5.1 Responsibility

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

5.2 Organization

5.2.1 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 for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the USAR.

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

c. The corporate officer with responsibility for overall plant nuclear safety shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

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

5.2.2 Plant Staff

The plant staff organization shall be as described in Chapter 12 of the USAR and shall function as follows:

a. The minimum number and type of licensed and unlicensed operating personnel required onsite for each shift shall be as shown in Table 5.2-1.

5.0 **ADMINISTRATIVE CONTROLS**

5.2 Organization (Continued)

b. An Operator or Technician qualified in Radiation Protection Procedures shall be onsite when fuel is in the reactor.

c. 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 the operation.

d. Fire protection program responsibilities are assigned to those positions and/or groups designated by asterisks in USAR 12.1-1 through 12.1-4 according to the procedures specified in the Quality Assurance Program.

e. Administrative procedures shall be developed and implemented to limit the working hours of plant staff who perform safety-related functions. Administrative procedures shall reflect the personnel whose working hours will be affected. Shift coverage shall be maintained without routine heavy use of overtime.

   Deviations from the guidelines shall be authorized in advance by the Department Manager, plant manager, or their designated alternates, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Routine deviation from the administrative guidelines shall not be authorized.

   Controls shall be included in the procedures such that individual overtime shall be reviewed periodically by the Department Manager, plant manager, or their designated alternates, or higher levels of management, to ensure that excessive hours have not been assigned.

f. The Manager - Shift Operations, the Shift Managers, and the Control Room Supervisors shall hold a senior reactor operator license. The Licensed Operators shall hold a reactor operator license.

5.0 **ADMINISTRATIVE CONTROLS**

5.3 Facility Staff Qualification

5.3.1 Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, with the exception of the Manager - Radiation Protection (MRP), the Shift Technical Advisors (STA), the senior reactor operator licensees, and the reactor operator licensees, who shall meet the requirements set forth in Regulatory Guide 1.8, Revision 3, dated May 2000, entitled "Qualification and Training of Personnel for Nuclear Power Plants."

FIGURE 5-1

This Figure has been deleted

FIGURE 5-2

This Figure has been deleted

TABLE 5.2-1

MINIMUM SHIFT CREW COMPOSITION(ii)

License Category	Core Alteration	Cold Shutdown or Refueling Shutdown	Operating or Hot Shutdown Modes
Senior Operator License	2(i)	1	2(iii)
Operator License	2	1	2(iv)
Non-Licensed	(As required)	1	2
Shift Technical Advisor	None	None	1

(i) This includes the individual with Senior Operator License supervising Core Alterations.

(ii) Shift crew composition may be one 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 5.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewmember being late or absent.

(iii) At least one of these individuals must be in the control room at all times.

(iv) At least one of these individuals (or the second senior licensed operator, if both senior licensed operators are in the control room) must be present at the controls at all times.

5.0 **ADMINISTRATIVE CONTROLS**

5.4 Training

5.4.1 A retraining and replacement training program for the plant staff shall be maintained under the direction of the Manager - Training and shall meet or exceed the requirements of Section 6 of ANSI/ANS 3.1-1993, as modified by Regulatory Guide 1.8, Revision 3, dated May 2000 and 10 CFR Part 55.

5.5 Review and Audit

5.5.1 Plant Review Committee (PRC)

A committee composed of key management personnel designated as the PRC acts in an advisory capacity on all matters related to nuclear safety to the plant manager and serves in accordance with Quality Assurance Program requirements, USAR Section 12.5, and plant Standing Orders.

5.0 ADMINISTRATIVE CONTROLS

DELETED

5.0 **ADMINISTRATIVE CONTROLS**

5.5.2 Safety Audit and Review Committee (SARC)

The Safety Audit and Review Committee (SARC) is a committee composed of highly qualified and experienced OPPD management personnel and consultants, which functions to provide independent review and audit of activities in accordance with the Quality Assurance Program requirements, USAR Section 12.5, and the SARC Charter. The SARC reports to and advises the corporate officer responsible for overall plant nuclear safety.

5.0 **ADMINISTRATIVE CONTROLS**

DELETED

5.0 **ADMINISTRATIVE CONTROLS**

DELETED

5.0 **ADMINISTRATIVE CONTROLS**

DELETED

5.0 ADMINISTRATIVE CONTROLS

5.6 Reportable Event Action

5.6.1 The following actions shall be taken in the event of a REPORTABLE EVENT:

a. The Commission shall be notified pursuant to the requirements of 10 CFR 50.72, if applicable.

b. Each Reportable Event shall be reviewed by the Plant Review Committee and submitted to the Chairperson of the Safety Audit and Review Committee and the corporate officer responsible for overall plant nuclear safety.

c. Submit reports of Reportable Events pursuant to the requirements of Specification 5.9.2.

5.7 Safety Limit Violation

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

a. The unit shall be placed in at least HOT SHUTDOWN within 1 hour.

b. The Safety Limit Violations shall be reported to the corporate officer responsible for overall plant nuclear safety and the Chairperson of the Safety Audit and Review Committee (SARC) within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Plant Review Committee. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Chairperson of the Safety Audit and Review Committee and the corporate officer responsible for overall plant nuclear safety within 14 days of the violation.

5.8 Procedures

5.8.1 Written procedures and administrative policies shall be established, implemented and maintained in accordance with the Quality Assurance Program.

5.0 ADMINISTRATIVE CONTROLS

5.9 Reporting Requirements

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

5.9.1 Routine Reports

a. Startup Report. 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 manufacture by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the USAR and shall in general 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.

5.0 ADMINISTRATIVE CONTROLS

5.9.1 Continued

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.

b. Annual Occupational Exposure Report. An annual occupational exposure report shall be submitted on or before April 30 of each year. The report shall consist of a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to

5.9.1 Continued

work and job functions,[3/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling outages. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

c. Monthly Operating Report. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, with a copy to the appropriate Regional Office, no later than the fifteenth of each month following the calendar month covered by the report. This monthly report shall also include a statement regarding any challenges or failures to the pressurizer power operated relief valves or safety valves occurring during the subject month.

5.9.2 Reportable Event

A Licensee Event Report (LER) shall be submitted to the U.S. Nuclear Regulatory Commission for any event meeting the requirements of 10 CFR Part 50.73.

3/ This tabulation supplements the requirements of § 20.2206 of 10 CFR Part 20.

5.0 **ADMINISTRATIVE CONTROLS**

5.9.3 Special Reports

Special reports shall be submitted to the appropriate NRC 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 where appropriate:

a. In-service inspection report, reference 3.3.
b. Tendon surveillance, reference 3.5.
c. Containment structural tests, reference 3.5.
d. DELETED
e. DELETED
f. DELETED
g. Materials radiation surveillance specimens reports, reference 3.3.
h. DELETED
i. Post-accident monitoring instrumentation, reference 2.21
j. Electrical systems, reference 2.7(2).

5.9.4 Unique Reporting Requirements

a. Annual Radioactive Effluent Release Report

The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year of operation shall be submitted before May 1 of each year. 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 1) consistent with the objectives outlined in the ODCM and PCP, and 2) in conformance with 10 CFR 50.36a. and Section IV.B.1 of Appendix I to 10 CFR 50.

b. Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis 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 (1) the ODCM and (2) Section IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR 50.

c. Fire Protection Program Deficiency Report

Deficiencies in the Fire Protection Program described in the Updated Safety Analysis Report which meet the reportability criteria of 10 CFR 50.73 shall be reported pursuant to Section 5.9.2 of the Technical Specifications.

5.0 **ADMINISTRATIVE CONTROLS**

5.9.5 Core Operating Limits Report

a. Core Operating Limits shall be established and documented in the Core Operating Limits Report (COLR) before each reload cycle or any remaining part of a reload cycle.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC as follows:

1. OPPD-NA-8301-P-A, "Reload Core Analysis Methodology Overview" approved version as specified in the COLR.

2. OPPD-NA-8302-P-A, "Neutronics Design Methods and Verification", approved version as specified in the COLR.

3. OPPD-NA-8303-P-A, "Transient and Accident Methods and Verification", approved version as specified in the COLR.

4. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Report," April 1995 (Westinghouse Proprietary) as approved in the Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 178 to Facility Operating License No. DPR-40, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Docket No. 50-285, dated October 25, 1996.

5. WCAP-13027-P, "Westinghouse ECCS Evaluation Model for Analysis of CE-NSSS," July 1991 (Westinghouse Proprietary) as approved in the Safety Evaluation by the Office of Nuclear Reactor Regulation dated March 26, 1992, and as applied in OPPD submittal to the NRC (LIC-96-0130) dated September 3, 1996, and as approved in the Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 178 to Facility Operating License No. DPR-40, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Docket No. 50-285, dated October 25, 1996.

6. XN-75-32(P)(A) Supplements 1, 2, 3 & 4, "Computational Procedure for Evaluating Fuel Rod Bowing," approved version as specified in the COLR.

7. XN-NF-82-06(P)(A) and Supplements 2, 4 and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.

8. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.

5.0 ADMINISTRATIVE CONTROLS

5.9.5 Core Operating Limits Report

9. ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU," approved version as specified in the COLR.

10. EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.

11. XN-NF-78-44(P)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," approved version as specified in the COLR.

12. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," approved version as specified in the COLR.

13. EMF-1961(P)(A), "Statistical Setpoint/Transient Methodology for CE Reactors, Siemens Power Corporation," approved version as specified in the COLR.

14. XN-NF-621(P)(A), "Exxon Nuclear DNB Correlation for PWR Fuel Designs," approved version as specified in the COLR.

15. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," approved version as specified in the COLR.

16. EMF-92-153(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.

17. XN-NF-82-49(P)(A), Supplement 1, "Exxon Nuclear Company Evaluation Model Revised EXEM PWR Small Break Model," approved version as specified in the COLR.

18. EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," approved version as specified in the COLR.

19. ANF-84-73 Appendix B (P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuels Corporation, approved version as specified in the COLR.

20. EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs," Siemens Power Corporation, approved version as specified in the COLR.

5.0 ADMINISTRATIVE CONTROLS

c. The core operating limits shall be determined so that all applicable limits of the safety analysis are met. The Core Operating Limits Report, 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 Region IV Administrator and Senior Resident Inspector.

5.10 Records Retention

5.10.1 Records shall be retained as described in the Quality Assurance Program.

5.0 **ADMINISTRATIVE CONTROLS**

5.11 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.

5.0 ADMINISTRATIVE CONTROLS

5.11.1 In lieu of the "control device" required by paragraph 20.1601(a) of 10 CFR Part 20, and as an alternative method allowed under § 20.1601(c), each high radiation area (as defined in § 20.1601) 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 required 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 by the Manager-Radiation Protection (MRP) in the Radiation Work Permit.

5.11.2 The requirements of 5.11.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr** but less than 500 rads/hr*** (Restricted High Radiation Area). In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Manager on duty and/or the MRP with the following exception:

a. In lieu of the above, for accessible localized Restricted High Radiation Areas located in large areas such as containment, where no lockable enclosure exists in the immediate vicinity to control access to the Restricted High Radiation Area and no such enclosure can be readily constructed, then the Restricted High Radiation Area shall be:

i. roped off such that an individual at the rope boundary is exposed to 1000 mrem/hr or less,
ii conspicuously posted, and
iii a flashing light shall be activated as a warning device.

*Radiation Protection personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.
**At 30 centimeters (12 inches) from the radiation source or from any surface penetrated by the radiation.
***At 1 meter from the radiation source or from any surface penetrated by the radiation.

5.0 **ADMINISTRATIVE CONTROLS**

5.12 Environmental Qualification

Deleted

5.13 Secondary Water Chemistry

A secondary water chemistry monitoring program to inhibit steam generator tube degradation shall be implemented. This program shall be described in the station chemistry manual and shall include:

1. Identification of a 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. Procedures for the recording and management of data;
5. Procedures defining corrective actions for off control point chemistry conditions; and
6. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective actions.

5.0 ADMINISTRATIVE CONTROLS

5.14 Systems Integrity

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 shall be implemented. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

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

5.15 Post-Accident Radiological Sampling and Monitoring

The following programs shall be implemented and maintained to ensure the capability to accurately monitor and/or sample and analyze radiological effluents and concentrations in a post-accident condition:

1. A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. (Any space which will require occupancy to permit an operator to aid in mitigation of, or recovery from, an accident is designated as vital.)

2. A program which will ensure the capability to obtain and analyze radioactive iodines and particulates in plant gaseous effluents.

3. A program which will ensure the capability to obtain and analyze a reactor coolant liquid sample under accident conditions.

4. A program which will ensure the capability to obtain and analyze a containment atmosphere sample under accident conditions.

These programs shall include the following:

1. Training of personnel.

2. Procedures for monitoring and/or sampling and analysis.

3. Provisions for maintenance of sampling and analysis equipment.

5.0 ADMINISTRATIVE CONTROLS

5.16 Radiological Effluents and Environmental Monitoring Programs

The following programs shall be established, implemented, and maintained.

5.16.1 Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for control of radioactive effluents and for maintaining the doses to individuals in unrestricted areas from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the operability of radioactive liquid and gaseous radiation monitoring instrumentation including operability tests and setpoint determination in accordance with the methodology in the ODCM.

b. Limitations on the concentration of radioactive material, other than dissolved or entrained noble gases, released in liquid effluents to unrestricted areas conforming to ten times 10 CFR 20.1001-20.2401, Appendix B, Table 2, Column 2. For dissolved or entrained noble gases, the concentration shall be limited to 2.0 E-04 μCi/ml total activity.

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 individuals in unrestricted areas from radioactive materials in liquid effluents released to unrestricted areas conforming to Appendix I to 10 CFR Part 50.

e. Determination of cumulative doses from radioactive effluents for the current calendar quarter and current calendar year in accordance with the ODCM on a quarterly basis.

f. Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity in plant effluents.

5.0 ADMINISTRATIVE CONTROLS

5.16 Radiological Effluents and Environmental Monitoring Programs (continued)

g. Limitations on the concentration resulting from radioactive material, other than noble gases, released in gaseous effluents to unrestricted areas conforming to ten times 10 CFR 20.1001-20.2401, Appendix B, Table 2, Column 1. For noble gases. the concentration shall be limited to five times 10 CFR 20.1001-20.2401, Appendix B, Table 2, Column 1.

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to unrestricted areas conforming to Appendix I to 10 CFR Part 50.

i. Limitations on the annual and quarterly doses to an individual beyond the site boundary from Iodine-131, tritium, and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released to unrestricted areas conforming to Appendix I to 10 CFR Part 50.

j. Limitations on the annual dose or dose commitment to an individual beyond the site boundary due to releases or radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

5.16.2 Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

b. A Land Use Census to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the monitoring program are made if required by the results of this census.

c. Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

5.0 **ADMINISTRATIVE CONTROLS**

5.17 Offsite Dose Calculation Manual (ODCM)

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by the Quality Assurance Program. This documentation shall contain:

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

   2. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the Plant Review Committee and the approval of the plant manager.

c. Temporary changes to the ODCM may be made in accordance with the Quality Assurance Program.

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

5.0 **ADMINISTRATIVE CONTROLS**

5.18 Process Control Program (PCP)

Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by the Quality Assurance Program. This documentation shall contain:

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

   2. A determination that the change will maintain the overall conformance of the solidified waste program to existing requirements of federal, state, or other applicable regulations.

b. Shall become effective after the review and acceptance by the Plant Review Committee and the approval of the plant manager.

c. Temporary changes to the PCP may be made in accordance with the Quality Assurance Program.

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

5.0 **ADMINISTRATIVE CONTROLS**

5.19 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," as modified by the following exceptions:

(1) If the Personnel Air Lock (PAL) is opened during periods when containment integrity is not required, the PAL door seals shall be tested at the end of such periods and the entire PAL shall be tested within 14 days after RCS temperature T_{cold} > 210°F.

(2) Type A tests may be deferred for penetrations of the steel pressure retaining boundary where the nominal diameter does not exceed one inch.

(3) Elapsed time between consecutive Type A tests used to determine performance shall be at least 24 months or refueling interval.

The containment design accident pressure (P_a) is 60 psig.

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

Leakage Rate acceptance criteria are:

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

b. Personnel Air Lock testing acceptance criteria are:

   (1) Overall Personnel Air Lock leakage is $\leq$ 0.1 L_a when tested at $\geq$ P_a.

   (2) For each PAL door, seal leakage rate is $\leq$ 0.01 L_a when pressurized to $\geq$ 5.0 psig.

c. Containment Purge Valve (PCV-742A/B/C/D) testing acceptance criterion is:

   For each Containment Purge Valve, leakage rate is < 18.000 SCCM when tested at $\geq$ P_a.

d. If at any time when containment integrity is required and the total Type B and C measured leakage rate exceeds 0.60 L_a Minimum Pathway Leakage Rate (MNLPR), repairs shall be initiated immediately. If repairs and retesting fail to demonstrate conformance to this acceptance criteria within 48 hours, then containment shall be declared inoperable.

5.0 **ADMINISTRATIVE CONTROLS**

5.19 Containment Leakage Rate Testing Program (continued)

The provisions of Specification 3.0.1 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 3.0.4 are applicable to the Containment Leakage Rate Testing Program.

Attachment B

Discussion, Justification and No Significant Hazards Consideration

DISCUSSION AND JUSTIFICATION

Omaha Public Power District (OPPD) is proposing to revise the Fort Calhoun Station (FCS) Unit No. 1 Technical Specifications (TS) 5.0, "Administrative Controls." The changes are administrative in nature to: 1) replace the titles of Manager – Fort Calhoun Station and Vice President with generic titles, 2) relocate the requirements for the Plant Review Committee (PRC) and the Safety Audit and Review Committee (SARC) to the Fort Calhoun Station Quality Assurance Program, 3) relocate the requirements for procedure controls and records retention to the Fort Calhoun Station Quality Assurance Program, 4) enhance and clarify the qualification and training requirements for individuals who perform licensed operator functions, 5) incorporate the Westinghouse/CENP definition of Azimuthal Power Tilt, and 6) eliminate specific mailing address and reporting requirements that are redundant to 10 CFR.

Changes

Specifications Definitions and 2.10.4 Bases

The proposed change would replace the definition of Azimuthal Power Tilt and add the bases for the definition of Azimuthal Power Tilt to the bases section of Section 2.10.4 as described in ABB Combustion Engineering Infobulletin Number 97-07, dated December 31, 1997. This will make the FCS definition and bases agree with the improved Standard Technical Specifications.

Specifications 5.2.1, 5.5.1, 5.5.2, 5.6.1, & 5.7.1

The proposed change would replace the specific title, Vice President, with the generic personnel title as provided in the improved Standard Technical Specifications (STS). This same title change will be reflected in those TS requirements being relocated to the FCS Quality Assurance Program upon approval of this amendment request.

Specifications 5.1.1, 5.2.1, 5.2.2, 5.5.1, 5.5.2, 5.8, 5.17, & 5.18

The proposed change would replace the specific title, Manager – Fort Calhoun Station, with the generic personnel title "plant manager" as provided in the improved Standard Technical Specifications (STS). This same title change will be reflected in those TS requirements relocated to the FCS Quality Assurance Program upon approval of this amendment request.

Specification 5.2.e

The proposed change revises the requirement to review overtime from monthly to periodically. Normally, overtime is only permitted during planned or forced outages. Thus, there is no need to review overtime on a monthly bases and an increased need to review during an outage. This allows us more effective use of management oversight.

Specification 5.2.f

The proposed change reflects a change to the title for the Supervisor - Operations and Licensed Senior Operator. The Manager - Shift Operations has the responsibility, authority, and qualification requirements previously conferred upon the Supervisor - Operations. The Control Room Supervisor has the responsibility, authority, and qualification requirements previously conferred upon the Licensed Senior Operator.

Table 5.2-1

The proposed change reflects a change from 1 to 2 Senior Operator Licenses present during "Core Alterations" and a change in note (i) indicating that this number "includes the individual with the Senior Operator License supervising Core Alterations" replacing the existing requirement that this number "does not include the individual with Senior Operator License supervising Refueling Operations." This is not a change in intent of the requirement and is purely a clarification of the requirements to assure compliance with 10 CFR 50.54. The term "crewman" is being changed to "crewmember" in note (ii), which is gender neutral and clarifying punctuation is added to note (iv), which has no affect on interpretation or implementation.

Specifications 5.3.1, 5.4.1

The proposed change would replace the reference to ANSI N18.1-1971 and Regulatory Guide 1.8 dated September 1975, with ANSI/ANS 3.1-1993 and Regulatory Guide 1.8, Revision 3, for the Manager - Radiation Protection, the Shift Technical Advisors, and those individuals subject to 10 CFR Part 55 who perform the functions described in 10 CFR 50.54(m). OPPD has complied with the intent of these newer requirements through commitments made in the implementation of NUREG-0737. The recommendations of this later standard are clearer and do not constitute a change in existing commitments. This change is in conformance with NRC Regulatory Issues Summary 2001-01.

Specifications 5.5.1

A brief description of the Plant Review Committee (PRC) with a reference to Quality Assurance

Program and USAR requirements is being added. The specific Technical Specifications requirements for the PRC will be relocated to the FCS Quality Assurance Program as described in NRC Administrative Letter 95-06. Upon approval of this amendment request, the specific titles will be replaced with the generic personnel titles in those requirements that are relocated to the Quality Assurance Program.

Specification 5.5.2

A brief description of the Safety Audit and Review Committee (SARC) with a reference to Quality Assurance Program and USAR requirements is being added. The specific Technical Specifications requirements for the SARC will be relocated to the FCS Quality Assurance Program as described in NRC Administrative Letter 95-06. Upon approval of this amendment request, the specific titles will be replaced with the generic personnel titles in those requirements that are relocated to the Quality Assurance Program.

Specifications 5.8 & 5.2.d

A brief statement that procedures and administrative policies will be controlled as described in the FCS Quality Assurance Program is being added. The remainder of the specific Technical Specifications requirements for procedures will be relocated to the FCS Quality Assurance Program as described in NRC Administrative Letter 95-06. Any changes that could reduce the effectiveness of the QA Program must be approved by the NRC in accordance with 10 CFR 50.54(a)(4).

Specification 5.9, 5.9.1.c, 5.9.2, 5.9.3, & 5.9.5.c

The specific Technical Specifications requirements prescribing the submittal address, mail station, titles, and reporting periods are being removed as they are redundant to the requirements contained in 10 CFR and to minimize the need for future changes to this TS resulting from changes to 10 CFR.

Specification 5.9.3.d

The specific Technical Specifications requirement prescribing the submittal of a Special Maintenance Report is being removed, as this is redundant to requirements contained in 10 CFR 50.73. The initial Technical Specifications required the submittal of this "special maintenance report" in the event a redundant component (or system) covered by the Technical Specifications was determined to be out of service for a period longer than those specified in other sections of the Technical Specifications. The section detailing the specific requirements for this report were removed in 1975 with the implementation of Regulatory Guide (RG) 1.16, Revision 4, "Reporting of Operating Information – Appendix A Technical Specifications." Section 2.a of RG 1.16, "Prompt Notification With Written Followup," later to be known as the Licensee Event Report (LER), duplicated or paraphrased these requirements.

Specifications 5.10, 5.17, & 5.18

A brief statement that records shall be retained as described in the FCS Quality Assurance Program is being added. The remainder of the specific Technical Specifications requirements for records retention will be relocated to the FCS Quality Assurance Program as described in NRC Administrative Letter 95-06. References for relocated requirements will now reference the QA Program. Any changes that could reduce the effectiveness of the QA Program must be approved by the NRC in accordance with 10 CFR 50.54(a)(4).

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION

The proposed changes to the Fort Calhoun Station (FCS) Unit No. 1 Technical Specifications do not involve significant hazards consideration because operation of FCS in accordance with the changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes: revise the FCS definition of Azimuthal Power Tilt, remove specific titles from the Technical Specifications, provide minor clarifications of the training requirements for plant staff, and indicate the change in title of the Licensed Senior Operator. This change also relocates the requirements for the Plant Review Committee (PRC) and the Safety Audit and Review Committee (SARC), procedure control, and records retention to the Fort Calhoun Station Quality Assurance Program as described in NRC Administrative Letter 95-06.

The proposed change includes an update to the definition of Azimuthal Power Tilt and adds the bases for the definition of Azimuthal Power Tilt to the bases section of Section 2.10.4 as recommended in ABB Combustion Engineering (CE) Infobulletin Number 97-07, dated December 31, 1997. As noted in the infobulletin, CE discovered a discrepancy in the definition for CE analog plants that use Combustion Engineering Core Operating Report (CECOR) for monitoring and surveillance purposes. Plants that use CECOR should use the same definition as the CE digital plants. This change will make the FCS definition and bases agree with the improved Standard Technical Specifications for CE digital plants, which have previously been approved by the NRC.

The proposed change would allow the use of generic personnel titles as provided in ANSI/ANS 3.1 and NUREG-1432, "Standard Technical Specifications Combustion Engineering Plants," in lieu of plant-specific personnel titles. This change does not eliminate any of the qualifications, responsibilities or requirements for these positions, since the plant-specific personnel titles are currently identified in licensee controlled documents such as the Updated Safety Analysis Report (USAR) or the Quality Assurance Program. For example, Section 12 of the Updated Safety Analysis Report describes the management structure and reporting responsibilities of OPPD and provides an organizational chart to determine the corporate officer with responsibility for overall plant nuclear safety from other corporate officers within OPPD. Therefore, changing the terminology within the Technical Specifications, indicating this reporting responsibility does not involve a significant increase in the probability or consequences of an accident previously evaluated. Changing the periodicity of review for staff overtime is also considered an administrative change. This includes a change of the title of the Supervisor - Operations to Manager - Shift Operations, Licensed Senior Operator to Control Room Supervisor, and crewman to crewmember. The change to the number of Senior Operator License present during Core Alterations and the associated note is also considered clarifying in nature and not a change of intent.

The proposed change would update the qualification requirements for the Manager - Radiation Protection, the Shift Technical Advisors, and those individuals that perform the functions described in 10 CFR 50.54(m) to Regulatory Guide 1.8, Revision 3, and ANSI/ANS 3.1-1993. In the March 1987 revision to 10 CFR Part 55, the NRC included the requirement that those facility licensees that have made a commitment that is less than that required by the new rules must conform to the new rules automatically. OPPD had previously considered that commitments made to comply with the requirements of NUREG-0737 and the standards applied through the Institute of Nuclear Power Operations (INPO) accreditation process were equivalent to the guidance provided in Regulatory Guide 1.8, Revision 3. The proposed change provides enhancement to the current requirements and clarifies the qualifications and training requirements for licensed personnel. This provides additional assurance that these personnel are properly trained and qualified for their positions and conforms with the guidance of NRC Regulatory Issues Summary 2001-01. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change would relocate specific requirements for SARC, PRC, procedure control, and records retention to the Fort Calhoun Station Quality Assurance Program (Appendix A, of the FCS USAR). This proposed revision does not change or eliminate responsibilities or requirements for these programs. The management level and expertise of personnel who are PRC or SARC members is not being changed. The review of plant operations, procedures control, and record retention is still required to be in compliance with the Fort Calhoun Station Quality Assurance (QA) Program. Any changes in the QA Program which reduce the effectiveness of the program must be approved by the NRC in accordance with 10 CFR 50.54(a)(4). These changes meet the criteria as described in NRC Administrative Letter 95-06. Therefore, the proposed relocation of these programs to the QA Program does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change would also remove the requirements prescribing specific submittal addresses, titles, and reporting periods. For example, the requirement to submit License Event Reports within 30 days is replaced with a citation referencing 10 CFR 50.73. This is in agreement with 10 CFR 50.73 and 10 CFR 50.4(f). Additionally, an administrative requirement prescribing the submittal of a Special Maintenance Report is being deleted, as it is redundant to the requirements of 10 CFR 50.73. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes revise organizational and administrative requirements contained within the Administrative Controls section of the TS. The proposed change to the definition of Azimuthal Power Tilt is as recommended in CE Infobulletin 97-07 for CE analog plants that use CECOR for monitoring and surveillance purposes and will have no affect on accidents previously evaluated. The proposed changes do not revise any equipment setpoints, change the manner in which any plant equipment is operated, or propose any new operating modes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Involve a significant reduction in a margin of safety.

The proposed changes revise organizational and administrative requirements contained within the Administrative Controls section of the TS. The proposed change to the definition of Azimuthal Power Tilt has no affect on the margin of safety. The proposed changes do not revise any equipment setpoints, change the manner in which any plant equipment is operated, or propose any new operating modes. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above considerations, OPPD concludes that the proposed amendment to the FCS Technical Specifications do not involve significant hazards considerations as defined by 10 CFR 50.92, and the proposed amendment will not result in a condition which significantly alters the impact of the station on the environment. Thus, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and, pursuant to 10 CFR 51.22(b), no environmental assessment need be prepared.