ATTACHMENT I

INDEX

ADMINISTRATIVE CONTROLS

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

LIQUID RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD) (mCi/ml)(1)
1. Batch Waste Release Tanks(4) (Waste Monitor Tanks and Recyle Monitor Tank)	P Each Batch	P Each Batch	Principal Gamma Emitters(6)	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma emitters)	1×10^{-5}
	P Each Batch	M Composite(2)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite(2)	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
2. Continuous Releases(5) (Containment Ventilation Unit Condensate Drain Tank Discharge and Conventional Waste Water Treatment System Outlet)	Continuous(3)	W Composite(3)	Principal Gamma Emitters(6)	5×10^{-7}
			I-131	1×10^{-6}
	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	Continuous(3)	M Composite(3)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Continuous(3)	Q Composite(3)	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

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

6.1.2 The Shift Supervisor (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Vice-President Nuclear Production shall be reissued to all Nuclear Production Department station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as described in the FSAR, Chapter 13.

UNIT STAFF

6.2.2 The unit organization shall be as described in the FSAR, Chapter 13, and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. At least one licensed Operator for each unit shall be in the control room when fuel is in either reactor. In addition, while either unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;

c. A Health Physics Technician# shall be on site when fuel is in either reactor;

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

e. A site Fire Brigade of at least five members shall be maintained onsite at all times.# The Fire Brigade shall not include three members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency; and

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#The Health Physics Technician and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

(Figure Deleted)

(Figure Deleted)

6.5 REVIEW AND AUDIT

6.5.1 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.1.1 Each procedure and program required by Specification 6.8 and other procedures which affect nuclear safety, and changes thereto, shall be prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group which prepared the procedure, or changes thereto.

6.5.1.2 Proposed changes to the Appendix A Technical Specifications shall be prepared by a qualified individual/organization. The preparation of each proposed Technical Specifications change shall be reviewed by an individual/group other than the individual/group which prepared the proposed change, but who may be from the same organization as the individual/group which prepared the proposed change. Proposed changes to the Technical Specifications shall be approved by the Station Manager.

6.5.1.3 Proposed modifications to unit nuclear safety-related structures, systems and components shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to nuclear safety-related structures, systems, and components shall be approved prior to implementation by the Station Manager; or by the Operating Superintendent, the Technical Services Superintendent, the Superintendent of Integrated Scheduling, or the Maintenance Superintendent, as previously designated by the Station Manager.

6.5.1.4 Individuals responsible for reviews performed in accordance with Specifications 6.5.1.1, 6.5.1.2, and 6.5.1.3 shall be members of the station supervisory staff, previously designated by the Station Manager to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the appropriate designated station review personnel.

6.5.1.5 Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be reviewed by the Station Manager; or by the Operating Superintendent, the Technical Services Superintendent, the Maintenance Superintendent, or the Superintendent of Integrated Scheduling as previously designated by the Station Manager.

ACTIVITIES (Continued)

6.5.1.6 ALL REPORTABLE EVENTS and all violations of Technical Specifications shall be investigated and a report prepared which evaluates the occurrence and which provides recommendations to prevent recurrence. Such reports shall be approved by the Station Manager and transmitted to the Vice President, Nuclear Production, and to the Director of the Nuclear Safety Review Board.

6.5.1.7 The Station Manager shall assure the performance of special reviews and investigations, and the preparation and submittal of reports thereon, as requested by the Vice President, Nuclear Production.

6.5.1.8 The station security program, and implementing procedures, shall be reviewed at least once per 12 months. Recommended changes shall be approved by the Station Manager or Superintendent of Station Services and transmitted to the Vice President, Nuclear Production, and to the Director of the Nuclear Safety Review Board.

6.5.1.9 The station emergency plan, and implementing procedures, shall be reviewed at least once per 12 months. Recommended changes shall be approved by the Station Manager and transmitted to the Vice President, Nuclear Production, and to the Director of the Nuclear Safety Review Board.

6.5.1.10 The Station Manager shall assure the performance of a review by a qualified individual/organization of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation, recommendations, and disposition of the corrective ACTION to prevent recurrence to the Vice President, Nuclear Production and to the Nuclear Safety Review Board.

6.5.1.11 The Station Manager shall assure the performance of a review by a qualified individual/organization of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and Radwaste Treatment Systems.

6.5.1.12 Reports documenting each of the activities performed under Specifications 6.5.1.1 through 6.5.1.11 shall be maintained. Copies shall be provided to the Vice President, Nuclear Production, and the Nuclear Safety Review Board.

6.5.2 NUCLEAR SAFETY REVIEW BOARD (NSRB)

FUNCTION

6.5.2.1 The NSRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

RECORDS

6.5.2.11 Records of NSRB activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each NSRB meeting shall be prepared, approved, and forwarded to the Vice President, Nuclear Production, and to the Executive Vice President, Engineering, Construction, and Production, within 14 days following each meeting;

b. Reports of reviews encompassed by Specification 6.5.2.8 above, shall be prepared, approved and forwarded to the Vice President, Nuclear Production, and to the Executive Vice President, Engineering, Construction, and Production, within 14 days following completion of the review; and

c. Audit reports encompassed by Specification 6.5.2.9 above, shall be forwarded to the Vice President, Nuclear Production, and to the Executive Vice President, Engineering, Construction, and Production, and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the Station Manager; or by: (1) the Operating Superintendent, (2) the Technical Services Superintendent, (3) the Maintenance Superintendent, or (4) the Superintendent of Integrated Scheduling, as previously designated by the Station Manager, and the results of the review shall be submitted to the NSRB and the Vice President, Nuclear Production.

6.7 SAFETY LIMIT VIOLATION

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

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President, Nuclear Production, and the NSRB shall be notified within 24 hours;

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Operating Superintendent and the Station Manager. 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;

SAFETY LIMIT VIOLATION (Continued)

c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRB and the Vice President, Nuclear Production, within 14 days of the violation; and

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

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

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

b. The applicable procedures required to implement the requirements of NUREG-0737;

c. Security Plan implementation;*

d. Emergency Plan implementation;

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation; and

g. Quality Assurance Program for effluent and environmental monitoring.

6.8.2 Each procedure of Specification 6.8.1 above, and changes thereto, shall be reviewed and approved by the Station Manager; or by: (1) the Operating Superintendent, (2) the Technical Services Superintendent, (3) the Maintenance Superintendent, or (4) the Superintendent of Integrated Scheduling, as previously designated by the Station Manager; prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of Specification 6.8.1 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 management staff, at least one of whom holds a Senior Operator license on the unit affected; and

*Review and approval may be performed by the Superintendent of Station Services.

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

c. The change is documented, reviewed, and approved by the Station Manager; or by: (1) the Operating Superintendent, (2) the Technical Services Superintendent, (3) the Maintenance Superintendent; or (4) the Superintendent of Integrated Scheduling, as previously designated by the Station Manager, within 14 days of implementation.

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

a. Reactor Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include RHR, Boron Recycle, Refueling Water, Liquid Waste, Waste Gas, Safety Injection, Chemical and Volume Control, Containment Spray, and Nuclear Sampling. The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and

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

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1) Training of personnel,

2) Procedures for monitoring, and

3) Provisions for maintenance of sampling and analysis equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

1) Identification of a sampling schedule for the critical variables and control points for these variables,

2) Identification of the procedures used to measure the values of the critical variables,

3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Radioactive Effluent Release Reports shall include the following information for each type of solid waste shipped offsite during the report period:

a. Total container volume, in cubic meters,

b. Total Curie quantity (determined by measurement or estimate),

c. Principal radionuclides (determined by measurement or estimate),

d. Type of waste (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Number of shipments, and

f. Solidification agent or absorbent (e.g., cement, or other approved agents (media)).

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

RECORD RETENTION (Continued)

g. Records of training and qualification for current members of the unit staff;

h. Records of inservice inspections performed pursuant to these Technical Specifications;

i. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

j. Records of meetings of the NSRB and reports required by Specification 6.5.1.12;

k. Records of the service lives of all snubbers listed in Tables 3.7-4a and 3.7-4b including the date at which the service life commences and associated installation and maintenance records;

l. Records of secondary water sampling and water quality; and

m. Records of analyses required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

6.10.3 Records of quality assurance activities required by the QA Manual shall be retained for a period of time required by ANSI N45.2.9-1974.

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is equal to or less than 1000 mrem/hr at 45 CM (18 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mrem/hr provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

HIGH RADIATION AREA (Continued)

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; or

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; or

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Station Health Physicist in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mrem/hr at 45 CM (18 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

For individual areas accessible to personnel with radiation levels greater than 1000 mrem/hr that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, that area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

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*Measurement made at 18 inches from source of radioactivity.

PROCESS CONTROL PROGRAM (PCP) (Continued)

6.13.2 Licensee-initiated changes to the PCP:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   2) A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   3) Documentation of the fact that the change has been reviewed and found acceptable by the Station Manager.

b. Shall become effective upon review and acceptance by a qualified individual/organization.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee-initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered, dated and containing the revision number together with appropriate analyses or evaluations justifying the change(s);

   2) A determination that the change will not reduce the accuracy or reliability of dose calculations or Setpoint determinations; and

   3) Documentation of the fact that the change has been reviewed and found acceptable by the Station Manager.

b. Shall become effective upon review and acceptance by a qualified individual/organization.

Attachment I.a

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Attachment I.a

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Justification and Safety Analysis

The proposed changes to the McGuire Technical Specifications address a range of administrative matters. This proposal seeks to streamline, update, correct, and clarify the administrative controls at McGuire Nuclear Station, as specified in the Technical Specifications. This proposal also includes the correction of a typographical error in Table 4.11-1.

In Table 4.11-1 the word "inlet" has been used inadvertantly instead of "outlet". The proposed amendment corrects this error.

The proposed change to Specification 6.1.2 is to clarify the intent of the Specification as to include only Nuclear Production Department Station personnel, not vendor, contractor, or other personnel. This is the result of the resolution of a concern identified at Catawba Nuclear Station (Reference Inspection Report 50-413/84-29) and corresponds to the current Catawba Nuclear Station Technical Specifications, which reflect the resolution of this situation.

The proposed amendment of the Technical Specifications seeks to correct certain job titles; Manager of Nuclear Production, or Vice President, Steam Production, and Executive Vice President, Power Operations, are being changed to Vice President, Nuclear Production, and Executive Vice President, Engineering, Construction and Production, respectively. The proposed amendment also seeks to delete figures 6.2-1 and 6.2-2, Offsite Organization and Station Organization, respectively. These two figures are presently in error in some respects, and are contained in Chapter 13 of the McGuire Nuclear Station Final Safety Analysis Report, and are maintained there in accordance with 10CFR50.71. Page XXI of the Index is modified to indicate these deletions.

The inclusion of the Superintendent of Integrated Scheduling in Specifications 6.5.1.3, 6.5.1.5, 6.6.1b, 6.8.2, and 6.8.3c is an administrative matter and involves no safety questions. The proposed changes would allow the Superintendent of Integrated Scheduling to review and/or approve modifications of safety-related structures, systems or components (6.5.1.3), proposed tests and experiments which affect nuclear safety and are not addressed in the FSAR or Technical Specifications (6.5.1.5), REPORTABLE EVENTS (6.6.1b), and procedures specified under Specification 6.8.1 and changes thereto (6.8.2 and 6.8.3c), if so designated by the Station Manager.

In each of the above cases, the Operating Superintendent, the Technical Services Superintendent, and the Maintenance Superintendent, each have the same authority as described above. Since the Superintendent of Integrated Scheduling is required to meet the same qualifications as each of these Superintendents, no loss of Technical Review Capability can occur, therefore there can be no impact on safety.

The purpose of the proposed change to the Technical Specification 6.5.1.8 and 6.8.1 is to allow the Station Services Superintendent to review and approve modifications relating to the Station Security Program and associated procedures. Currently the responsibility is discharged by the Station Manager. The proposed change is purely administrative in nature and would facilitate efficient resolution of security related matters. The modifications approved by the Station Services Superintendent will be transmitted to the Vice President, Nuclear Production and the Director of the Nuclear Safety Review Board. Also, two typographical errors (one each in 6.5.1.8 and 6.5.1.9) are corrected.

The Semiannual Radioactive Effluent Release requirements listed in the Technical Specification 6.9.1.7 (pg. 6-20) require minor changes of words to make these requirements consistent with 10CFR Part 61. A footnote on this page is outdated and is also being deleted by the proposed amendment. This error and the error in Table 4.11-1 were detected by an in-house QA audit of the Technical Specifications governing the radwaste systems.

The purpose of the proposed changes to Specification 6.10.2 is to eliminate an inconsistency between the Technical Specification 6.10.2 and the Duke Power Company record retaining procedures based upon ANSI N45.2.9-1974. The proposed change would require that the records of the quality assurance activities as described in the QA Manual may be retained in accordance with the ANSI N45.2.9-1974 and not for the duration of the unit Operating License as currently required by the Technical Specifications.

McGuire Nuclear Station is committed to the directives in the Regulatory Guide 1.88. The regulatory guide affirms the use of ANSI N45.29-1974 for administrative controls governing the QA records.

The proposed change to Specification 6.12 provides consistency between McGuire and Catawba High Radiation Area Technical Specifications, and reflects the wording of the current draft of Revision 5 to the Standard Technical Specifications. The change will not significantly affect current radiation protection practices at McGuire, but is intended to clarify the requirements relative to definition of and access to high radiation areas. The change is largely semantic and does not affect plant safety.

The issuance of Amendments 32 (Unit 1) and 13 (Unit 2) resulted in the renumbering of several pages in Section 6. This resulted in duplication when pages 6-27 and 6-28 were not deleted. Since page 6-27 duplicates 6-25, and 6-28 duplicates 6-26, we propose to delete pages 6-27 and 6-28 from the Technical Specifications.

These proposed changes to the McGuire Technical Specifications are intended to promote a more efficient resolution of administrative matters at McGuire Nuclear Station. Wording changes are proposed to clarify requirements and procedure review is expanded to improve efficiency with no loss of review capability. These changes are administrative in nature and will not impact plant safety.

Analysis of Significant Hazards Consideration

As required by 10CFR50.91 this analysis provides a determination that the proposed change to the Technical Specifications does not involve any significant hazards consideration, as defined by 10CFR50.92.

The proposed changes to the Technical Specifications are purely administrative in nature and do not affect plant safety. The changes simply update, clarify, and correct the Technical Specifications. The proposed changes would not affect plant equipment or plant operational procedures and have no safety implications.

The proposed amendment would not:

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

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

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

Based upon the preceding analysis. Duke Power Company concludes that the proposed amendments do not involve a significant hazards consideration.

ATTACHMENT II

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or, in lieu of a Licensee Event Report, explain in the next Semiannual Radioactive Effluent Release Report why this inoperability was not corrected within the time specified.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-8.

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-13.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or, in lieu of a Licensee Event Report, explain in the next Semiannual Radioactive Effluent Release Report why this inoperability was not corrected within the time specified.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-9.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

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

3) Identification of tubes plugged.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G.	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes.	N/A	N/A

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

ACTION: (Continued)

MODES 1, 2, 3, 4, and 5:

With the specific activity of the reactor coolant greater than 1.0 microCurie per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microCuries per gram of gross specific activity, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

In lieu of a Licensee Event Report, for this ACTION statement within 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 with a copy to the Director, Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, and Chief, Accident Evaluation Branch, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555. This report shall contain the results of the specific activity analyses together with the following information:

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded;

2. Results of the last isotopic analysis for radioiodines performed prior to exceeding the limit, while limit was exceeded, and one analysis after the radioiodine activity was reduced to less than the limit, including for each isotopic analysis, the date and time of sampling and the radioiodine concentrations;

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded;

4. History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded; and

5. The time duration when the specific activity of the reactor coolant exceeded 1.0 microCurie per gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

CONTAINMENT SYSTEMS

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of the exposed accessible interior and exterior surfaces of the vessel. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance of the surfaces or other abnormal degradation. Any abnormal degradation of the containment vessel detected during the above required inspections shall be reported to the Commission pursuant to 10CFR Sections 50.72 and 50.73.

CONTAINMENT SYSTEMS

REACTOR BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.7 The structural integrity of the reactor building shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the structural integrity of the reactor building not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The structural integrity of the reactor building shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of the exposed accessible interior and exterior surfaces of the reactor building and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation. Any abnormal degradation of the reactor building detected during the above required inspections shall be reported to the Commission pursuant to 10CFR Sections 50.72 and 50.73.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

1) Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution, and

2) Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code at a test pressure equal to 110% of the system design pressure.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.2 within 30 days. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

4.8.1.1.4 Diesel Generator Batteries - Each diesel generator 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

   1) The electrolyte level of each battery is above the plates, and

   2) The overall battery voltage is greater than or equal to 125 volts under a float charge.

b. At least once per 18 months by verifying that:

   1) The batteries, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration;

   2) The battery-to-battery and terminal connections are clear, tight, free of corrosion and coated with anti-corrosion material; and

   3) The battery capacity is adequate to supply and maintain in OPERABLE status its emergency loads when subjected to a battery service test.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1-4) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microCurie/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrem/yr to the whole body and less than or equal to 3000 mrem/yr to the skin, and

b. For Iodine-131 and 133, for tritium, and for all radioactive materials in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The dose rate due to radioactive materials, other than noble gases, in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

STEAM GENERATORS (Continued)

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to 10CFR Sections 50.72 and 50.73 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the Safety Injection flow will not be less than assumed in the accident analyses.

The total steam generator tube leakage limit of 1 gpm for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

Attachment II.b

Justification and Safety Analysis

The proposed changes to the Technical Specifications involve the reportability requirements of Section 6.9 of the McGuire Technical Specifications.

As presented in Generic Letter No. 83-43 (dated December 19, 1983), the regulations regarding reporting of events (10CFR sections 50.72 and 50.73) were changed. Subsequently, section 6.9 of McGuire's Technical Specifications was amended to reflect the new reporting requirements. When this section was amended, however, the references in other parts of the Specifications were not updated. This proposal is to appropriately update these references. References to Specification 6.9.1.11b have been deleted because Specification 6.9.1.11b has been deleted. Technical Specification 6.9.1 has been so changed that references to it are virtually meaningless. These references to Specification 6.9.1 have been modified on an individual basis to preserve present reporting requirements, but to clarify these requirements and remove redundancies in the requirements.

These proposed changes are administrative in nature and will have no impact upon plant safety.

ANALYSIS OF SIGNIFICANT HAZARDS CONSIDERATION

As required by 10CFR50.91 this analysis provides a determination that the proposed changes to the Technical Specifications do not involve any significant hazards consideration, as defined by 10CFR50.92.

The proposed changes in the Technical Specifications are purely administrative in nature to correct inconsistencies arising from the modification of Section 6.0 of the Technical Specifications following the change of the reporting requirements of 10 CFR Sections 50.72 and 50.73. The proposed change would not affect any equipment or plant operational procedures and has no safety implications.

The proposed amendment would not:

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

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

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

Based upon the preceding analysis, Duke Power Company concludes that the proposed amendments do not involve a significant hazards consideration.