TABLE OF CONTENTS (Continued)

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (continued)
2.1.6 Pressurizer and Main Steam Safety Valves (continued)

d. With both PORVs inoperable in Modes 4 or 5, depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the next 36 hours.

(5) Two power-operated relief valves (PORVs) and their associated block valves shall be operable in Modes 1, 2, and 3.

a. With one or both PORV(s) inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to operable status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to operable status or close its associated block valve and remove power from the block valve; restore the PORV to operable status within the following 72 hours or be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

c. With both PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to operable status or close both block valves, remove power from the block valves, and be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

d. With one or both block valve(s) inoperable, within 1 hour restore the block valve(s) to operable status or place the associated PORV(s) in the closed position. Restore at least one block valve to operable status within the next hour if both block valves are inoperable; restore the remaining inoperable block valve to operable within 72 hours. Otherwise, be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

Basis

The purpose of the two spring-loaded Pressurizer Safety Valves (PSV's) is to provide Reactor Coolant System (RCS) overpressure protection and thereby ensure that the Safety Limit for RCS pressure (i.e., 2750 psia) is not exceeded for analyzed accidents. The maximum RCS pressure transient for an analyzed accident is associated with a Loss of Load event[2].

The TS 2.1.6(1) lift settings are determined during Surveillance Testing in accordance with ASME Code test methods. The ASME Code requires that valves in steam service use steam as the test medium for establishing the setpoint. The +1%/-3% tolerance range specified in TS 2.1.6(1) applies to opening pressures determined during Surveillance Testing. When the valves are installed in the system, the presence of a water-filled loop seal at the valve inlets may result in in-situ actuation at a pressure that differs from the actuation pressure with steam at the inlet. Comparative testing and analysis indicates that with a loop seal present, the opening pressure of these valves may be up to 1% lower than the opening pressure under normal test conditions. Opening pressures below the specified setpoints are not a concern with respect to the safety limit for RCS pressure. Analysis of loss of load case involving elevated PSV opening pressures indicated that RCS pressures remained below the 2750 psia Safety Limit with PSV opening pressures up to 6% above nominal setpoints. The valves are set to a tolerance of ±1% of setpoint using ASME Code test methods before being returned to service after testing. This allows for some setpoint variance over the surveillance interval.

2.0 LIMITING CONDITIONS FOR OPERATION
2.10 Reactor Core (Continued)
2.10.2 Reactivity Control Systems and Core Physics Parameters (Continued)

2. The position of each trippable CEA required shall be determined at least once per 2 hours, and

3. Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the Shutdown Margin to less than the limits of Specification 2.10.2(1).

(ii) If the shutdown margin specified in part (i) above is not available, immediately initiate and continue boration until the requirements of 2.10.2(1) are met.

c. Moderator Temperature Coefficient

(i) The moderator temperature coefficient (MTC) requirements of 2.10.2(3) may be suspended during physics tests at less than 10^{-1}% of rated power.

(ii) If power exceeds 10^{-1}% of rated power, either:

1. Reduce power to less than 10^{-1}% of rated power within 15 minutes, or

2. Be in hot shutdown in 2 hours.

Basis
Shutdown Margin

A sufficient shutdown margin ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Shutdown margin requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum shutdown margin as specified in the COLR is initially adequate to control the reactivity transient. Accordingly,

TABLE 3-13

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 300 tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 600 tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 1200 tubes in this S.G.	C-1	None
					C-2	Plug or repair defective tubes
			C-3	Perform action for C-3 result of first sample	C-3	Perform action for C-3 result of first sample
	C-3	Inspect all tubes in this S.G., plug or repair defective tubes and inspect 600 tubes in other S.G.	The second S.G. is C-1	None	N/A	N/A
			The second S.G. is C-2	Perform action for C-2 result of second sample	N/A	N/A
			The second S.G. is C-3	Inspect all tubes in the second S.G. and plug or repair defective tubes.	N/A	N/A

N/A Not applicable

5.0 ADMINISTRATIVE CONTROLS

5.8 Procedures (Continued)

c. The change is documented, reviewed by a qualified reviewer and approved by either the plant manager or the department head designated by Administrative Controls Standing Orders as the responsible department head for that procedure within 14 days of implementation.

5.8.3 Written procedures 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.

5.9 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 Not Used

5.0 ADMINISTRATIVE CONTROLS

5.9 Reporting Requirements (Continued)

5.9.2 Not Used

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 5.21.
c. DELETED
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).

5.0 ADMINISTRATIVE CONTROLS

5.9.6 Reactor Coolant System (RCS) Pressure - Temperature Limits Report (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature overpressure protection, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for Technical Specifications 2.1.1 and 2.1.2.

b. The analytical methods used in the PTLR shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. CE NPSD-683-A, Revision 6, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," April 2001.
2. WCAP-15443, Revision 0, "Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel," July 2000.
3. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Number 199 to Facility Operating License DPR-40 Omaha Public Power District Fort Calhoun Station, Unit Number 1, dated June 7, 2001.
4. CEN-636, Revision 2, "Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials, dated July 2000.
5. FC06876, Revision 0, "Performance of Low Temperature Overpressure Protection System Analyses Using RELAP5: Methodology Paper."
6. FC06877, "Low Temperature Overpressure Protection (LTOP) Analysis, Revision 1."
7. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Number 207 to Facility Operating License Number DPR-40 Omaha Public Power District Fort Calhoun Station, Unit Number 1, dated April 22, 2002.
8. Letter LTR-CI-01-25, Revision 0 from Westinghouse Electric Company (S.T. Byrne) to OPPD (J. Jensen), "Assessment of Extended Beltline Limit for Fort Calhoun Station Reactor Pressure Vessel," dated December 18, 2001.
9. WCAP-15741, Revision 0, "Reactor Vessel Surveillance Program Withdrawal Schedule Modifications," dated September 2001.
10. Letter from NRC (A. B. Wang) to Omaha Public Power District (R. T. Ridenoure), Fort Calhoun Station - Unit 1, Exemption from the Requirements of Appendix G to 10 CFR Part 50 (TAC No. MB8237), dated July 30, 2003.
11. Letter from Information Systems Laboratories (William Arcieri) to OPPD (J. Jensen), "WCA-09-2002: Transmittal of RELAP5/MOD3.2d," dated August 2, 2002.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period (i.e., the number of EFPY used in the P-T limit/LTOP analysis) and for any revision or supplement thereto.

5.0 ADMINISTRATIVE CONTROLS
5.10 Record Retention

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

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.

5.11.1 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:

*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

5.11 Radiation Protection Program (Continued)

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.

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.

5.14 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.

5.0 ADMINISTRATIVE CONTROLS

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.

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.

5.16 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.

5.0 **ADMINISTRATIVE CONTROLS**

5.16 Radiological Effluents and Environmental Monitoring Programs (Continued)

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.

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:

5.0 ADMINISTRATIVE CONTROLS

5.16 Radiological Effluents and Environmental Monitoring Programs (Continued)

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.

5.17 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 Technical Specification 5.8.2.

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.

5.18 Process Control Program (PCP)

Changes to the PCP:

5.0 ADMINISTRATIVE CONTROLS

5.18 Process Control Program (PCP) (Continued)

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 Technical Specification 5.8.2.

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.

5.19 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.

(4) The first Type A test performed after the November 1993 Type A test shall be no later than November 2008.

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

5.0 ADMINISTRATIVE CONTROLS

5.19 Containment Leakage Rate Testing Program (Continued)

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.

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 Specifications 3.0.4 and 3.0.5 are applicable to the Containment Leakage Rate Testing Program.

5.20 Technical Specifications (TS) Bases Control Program

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

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

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

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

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

5.0 ADMINISTRATIVE CONTROLS

5.20 Technical Specifications (TS) Bases Control Program (Continued)

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

d. Proposed changes that meet the criteria of 5.20.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.21 Containment Tendon Testing Program

This program provides controls for monitoring any tendon degradation in prestressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Containment Tendon Testing Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, Revision 3, 1990.

The provisions of TS 3.0.1 and TS 3.0.5 are applicable to the Containment Tendon Testing Program inspection frequencies.

If the acceptance criteria are not met, an immediate investigation shall be made to determine the cause(s) and extent of the non-conformance to the criteria, and the results shall be reported to the Commission within 90 days via a special report in accordance with Technical Specification 5.9.3.

5.22 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. An API gravity or an absolute specific gravity within limits,
   2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
   3. A clear and bright appearance with proper color, or a water and sediment content within limits;

b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil, and

c. Total particulate concentration of the fuel oil is $\leq$ 10 mg/l when tested every 31 days.

The provisions of TS 3.0.1 and TS 3.0.5 are applicable to the Diesel Fuel Oil Testing Program test frequencies.