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Regulatory File Cy.

August 1, 1973

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RECEIVED AUG 13 1973 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

Mr. J. F. O'Leary, Director
Directorate of Licensing
U.S. Atomic Energy Commission
Washington, D.C. 20545

Subject: Proposed Change to Appendix A of DPR-25, AEC Dkt 50-249

Dear Mr. O'Leary:

Pursuant to Section 50.59 of 10 CFR Part 50 and Paragraph 3.B of Facility License DPR-25, Commonwealth Edison Company hereby submits a proposed change to Appendix A of DPR-25 (Dresden Unit 3). The purpose of this change is to modify the current Technical Specifications concerning Administrative Controls, control rod scram reactivity, the radiological environmental monitoring program, and the chemical composition of standby liquid control system poison solution.

The page changes to the Technical Specifications are attached, and a safety evaluation for the proposed change is given below.

Administrative Controls

This proposed change is in response to a request in a letter from D. J. Skovholt to Byron Lee, Jr., dated September 28, 1972. As requested, this proposed Section 6 of the Technical Specifications conforms as nearly as possible to Regulatory Guides 1.16 and 1.21. This proposed Section 6 reflects the Quad-Cities Technical Specifications which have been reviewed and approved by your staff. The Dresden Station Section 6 differs only to the specific manning requirements in Figure 6.1.2 and Table 6.1.1. This proposed change has been reviewed and approved by the Dresden Station Review Board and the Nuclear Review Board.

Control Rod Scram Reactivity

The evaluation of this proposed change is discussed in Dresden Special Report No. 29. This change as indicated

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on the revised pages is applicable to the current fuel cycle (Cycle 2).

Radiological Environmental Monitoring Program

These proposed revisions to Table 4.8.1 are requested to modify the names assigned to certain sampling stations and relocate certain sample stations to optimize off-site monitoring. The total number of active air samplers remains unchanged by this proposal. Following is a description of the proposed changes.

(1) Changing the name of the Goose Lake air sampler back to its former name Clay Products.

(2) Deactivating the Lorenzo air sampler station (but not moving the housing) because of its proximity to the Pheasant Trail sampler (see Change No. 5 below).

(3) Moving the presently deactivated Plainfield sampler to the corner of Collins Road and Dresden Lock and Dam Road.

(4) Moving the Hansel sampler to the vicinity of the Goose Lake Prairie State Park.

(5) Moving the Breen sampler to a nearby location at the end of a street named Pheasant Trail.

(6) Moving the collection of milk and related samples to Filotto Farm from Dhuse Farm which has gone out of the dairy business.

This proposed change has been reviewed and approved by the Dresden Station Review Board and the Nuclear Review Board.

Standby Liquid Control System

This change is to clarify the chemical formula of the sodium pentaborate used in the Standby Liquid Control System. The formula given is consistent with the design of the system described in Section 6.7.1 of the Final Safety

Analysis Report. This clarification is proposed because the chemical formula shown in Table 6.7.1 of the Final Safety Analysis Report contains two typographical errors, i.e., $Na_2Br_{10}O_6 \cdot 10H_2O$ vs. $Na_2B_{10}O_{16} \cdot 10H_2O$.

Attached are three (3) signed originals and 19 copies of this proposed license change.

Very truly yours,

Byron Lee Jr.

Byron Lee, Jr.
Vice-President

SUBSCRIBED and SWORN to before me this 1st day of August, 1973.

Notary Public

Bases:

2.1 The transients expected during operation of the Dresden 3 unit have been analyzed starting at the rated thermal power condition of 2527 MWt at 100% recirculation flow. It should be noted that this power is equivalent to the designed maximum power and a higher power cannot physically be obtained under normal operating conditions unless the turbine bypass system is used. In addition, 2527 MWt is the licensed maximum steady-state power level of Dresden 3. This maximum steady-state power will never be knowingly exceeded.

Dresden 3 was not analyzed from a power level which included instrument errors. To protect against misleading conclusions from analysis not reflecting realistic instrument errors, conservatism was incorporated by conservatively estimating the controlling factors such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, axial power shapes, etc. These factors are all selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for the evaluation of reactor dynamics performance. Comparisons have been made showing results obtained from a General Electric boiling water reactor and the predictions made by the model. The comparisons and results are summarized in Topical Report APED-5698, "Summary of Results Obtained From A Typical Startup and Power Test Program for a General Electric Boiling Water Reactor."

The void reactivity coefficient utilized in the analysis is conservatively estimated to be about 25% larger than the most negative value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to the scram worth of about 75% of the control rods. The scram delay time and rate of rod insertion are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The insertion of the first dollar of reactivity strongly turns the transient and the stated 5% and 20% insertion time conservatively accomplishes this desired initial effect. The time for 50% and 90% insertion are given to assure proper completion of the insertion stroke, to further assure the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

The design peaking factors at the full power conditions for Dresden 3 result in a MCHFR value of 2.04. For analysis of the thermal consequences of the transients, higher peaking factors are used, such that a MCHFR of 1.9 is conservatively assumed to exist prior to initiation of the transients.

This choice of using conservative values of controlling parameters and initiating transients at the rated power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels. As an example, consider the sensitivity analyses conduct to provide the answer to Question 4.6.4 of Amendment 7 of the Dresden Unit 2 SAR. From the results of the Case 1 transient, the turbine trip with flux scram without bypass or relief, a significant reduction in the neutron flux and heat flux peaks will be realized when the smaller void reactivity coefficient is used. For this particular transient, if it were also analyzed at a power level of 110% of rated but with the expected void reactivity coefficient, the resulting heat flux peak would be less than the peak resulting from the analysis

Bases:

1.2 The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1325 psig as measured by the vessel steam space pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressures of the reactor pressure vessel, coolant system piping and isolation condenser. The respective design pressures are 1250 psig at 575°F, 1175 psig at 560°F, and 1250 psig at 575°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III for the pressure vessel and isolation condenser and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% X 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% X 1175 = 1410 psig). The Safety Limit pressure of 1375 psig is referenced to the lowest elevation of the primary coolant system.

The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig; this is a factor of 1.5 below the yield strength of 40,100 psi at 575°F. At the pressure limit of 1375 psig, the general membrane stress will only be 29,400 psi, still safely below the yield strength.

The relationships of stress levels to yield strength are comparable for the isolation condenser and primary system piping and provide a similar margin of protection at the established safety pressure limit.

The normal operating pressure of the reactor coolant system is 1000 psig. For the turbine trip or loss of electrical load transients the turbine trip scram or generator load rejection scram, together with the turbine bypass system limit the pressure to approximately 1100 psig (4). In addition, pressure relief valves have been provided to reduce the probability of the safety valves operating in the event that the turbine bypass should fail. These valves and the neutron flux scram limit the reactor pressure to 1185 psig (5)&(6) which is 25 psi below the setting of the first safety valve. Finally, the safety valves are sized to keep the reactor coolant system pressure below 1375 psig with no credit taken for the relief valves or turbine bypass system. Credit is taken for the neutron flux scram however.

Reactor pressure is continuously monitored in the control room during operation on a 1500 psi full scale pressure recorder.

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(4) SAR Section 11.2.2.

(5) SAR Section 4.4.3.

(6) Special Report No. 29

3.3 LIMITING CONDITION FOR OPERATION	4.3 SURVEILLANCE REQUIREMENT

C. Scram Insertion Times

1. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	5.00

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	5.300

2. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

C. Scram Insertion Times

1. After each refueling outage and prior to power operation with reactor pressure above 800 psig, all control rods shall be subject to scram-time tests from the fully withdrawn position. The scram times shall be measured without reliance on the control rod drive pumps.

2. At 16 week intervals, 50% of the control rod drives shall be tested as in 4.3.C.1 so that every 32 weeks all of the control rods shall have been tested. Whenever 50% of the control rod drives have been scram tested, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

3. 25 of the operable control rods, selected to be uniformly distributed throughout the core, shall be scram-time tested at full reactor pressure at the time intervals listed below following any outage exceeding 72 hours in duration: 1 week, 2 weeks, 4 weeks, 8 weeks, 16 weeks and continuing at 16 week intervals:
   a) If the mean 90% insertion time of the tested control rod drives increases by more than 0.25 seconds or if the mean insertion time exceeds 3.5 seconds, then an additional sample of 25 control rods, selected to be uniformly distributed throughout the core, shall be scram tested. If the mean 90% insertion time of the 50 selected control rod drives exceeds 4.25 seconds, then all operable drives will be tested. Subsequent testing shall revert to the original 25 control rods at the 1 week, 2 week, etc., sequence interval; and

Minimizer is out of service, when required, a licensed operator or other qualified technical employee can manually fulfill the control rod pattern conformance functions of the Rod Worth Minimizer. In this case, the normal procedural controls are backed up by independent procedural controls to assure conformance.

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. This is needed for knowledgeable and efficient reactor startup at low neutron levels. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scat-scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to a written sequence. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists. Amendments 18 and 20 present the results of an evaluation of a rod block monitor failure. These amendments show that during reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCHFR's less than 1.0. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other then limiting patterns.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCHFR from becoming less than 1.0. The limiting power transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above Specification, provide the required protection, and MCHFR remains greater than 1.0. Figure 3.5.2 of the SAR[1] shows the control rod scram reactivity used in analyzing the transients. Figure 3.5.2[1] should not be confused with the

(1) For Cycle 2, Figure I-1 of Special Report No. 29

total control rod worth, 18% Δk, as listed in some amendments to the SAR. The 18% Δk value represents the amount of reactivity available for withdrawal in the cold clean core, whereas the control rod worths shown in Figure 3.5.2 of the SAR(1) represent the amount of reactivity available for insertion (scram) in the hot operating core. The minimum amount of reactivity to be inserted during a scram is controlled by permitting no more than 10% of the operable rods to have long scram times. In the analytical treatment of the transients, 390 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 270 milliseconds. Approximately 70 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid de-energizes. Approximately 200 milliseconds later, control rod motion begins. The time to de-energize the pilot valve scram solenoids is measured during the calibration tests required by Specification 4.1. The 200 milliseconds are included in the allowable scram insertion times specified in Specification 3.3.C.

The scram times for all control rods will be determined at the time of each refueling outage. A representative sample of control rods will be scram tested at increasing intervals following a shutdown. Plugging of the internal drive filters has resulted in occasional increases in scram times at rates greater than one second per week of startup operation. Scram times of new drives are approximately 2.5 to 3.0 seconds; lower rates of change in scram times following initial plant operation at power are expected. The test schedule at increasing time intervals provides reasonable assurance of detection of slow drives before system deterioration beyond the limits of Specification 3.3.C. The program was developed on the basis of the statistical approach outlined below and judgment.

The probability that the mean 90% insertion time of a sample of 25 control rod drives will not exceed 0.25 seconds of the mean of all drives is 0.99 at a risk of 0.01. If the mean time exceeds this range or the mean 90% insertion time is greater than 3.5 seconds, an additional sample of drives will be measured to verify the mean performance. Since the differences between the expected observed mean insertion time and the limit of 3.3.C greatly exceeds the expected range, this sampling technique gives assurance that the limits of 3.3.C will not be exceeded. As further assurance that the limits of 3.3.C will not be exceeded, all operable drives will be scram tested to determine compliance to Specification 3.3.C if the enlarged sample of 50 control rods exceed 4.25 seconds. The 0.75 second margin to the limit is greater than the maximum expected deviation from the mean and therefore gives assurance the the mean will not exceed the limit of Specification 3.3.C. In addition, 50% of the control rods will be checked every 16 weeks to verify the performance and for correlation with the sampling program.

The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 seconds is greater than 0.999 for a normal distribution. The measurement of the scram performance of the drives surrounding a drive exceeding the expected range of

(1) For Cycle 2, Figure I-1 of Special Report No. 29

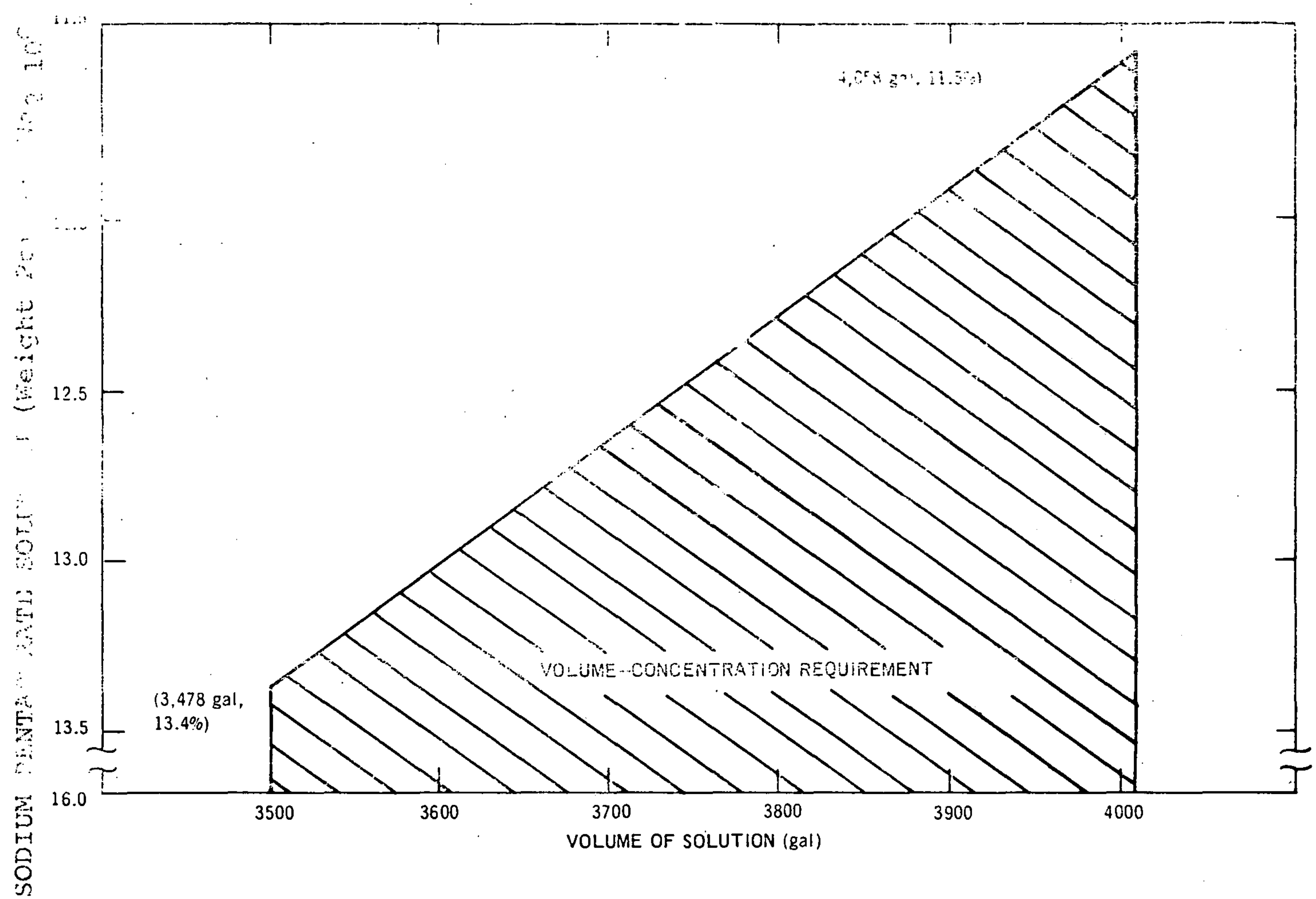

Figure 3.4.1 Standby Liquid Control Solution Requirements

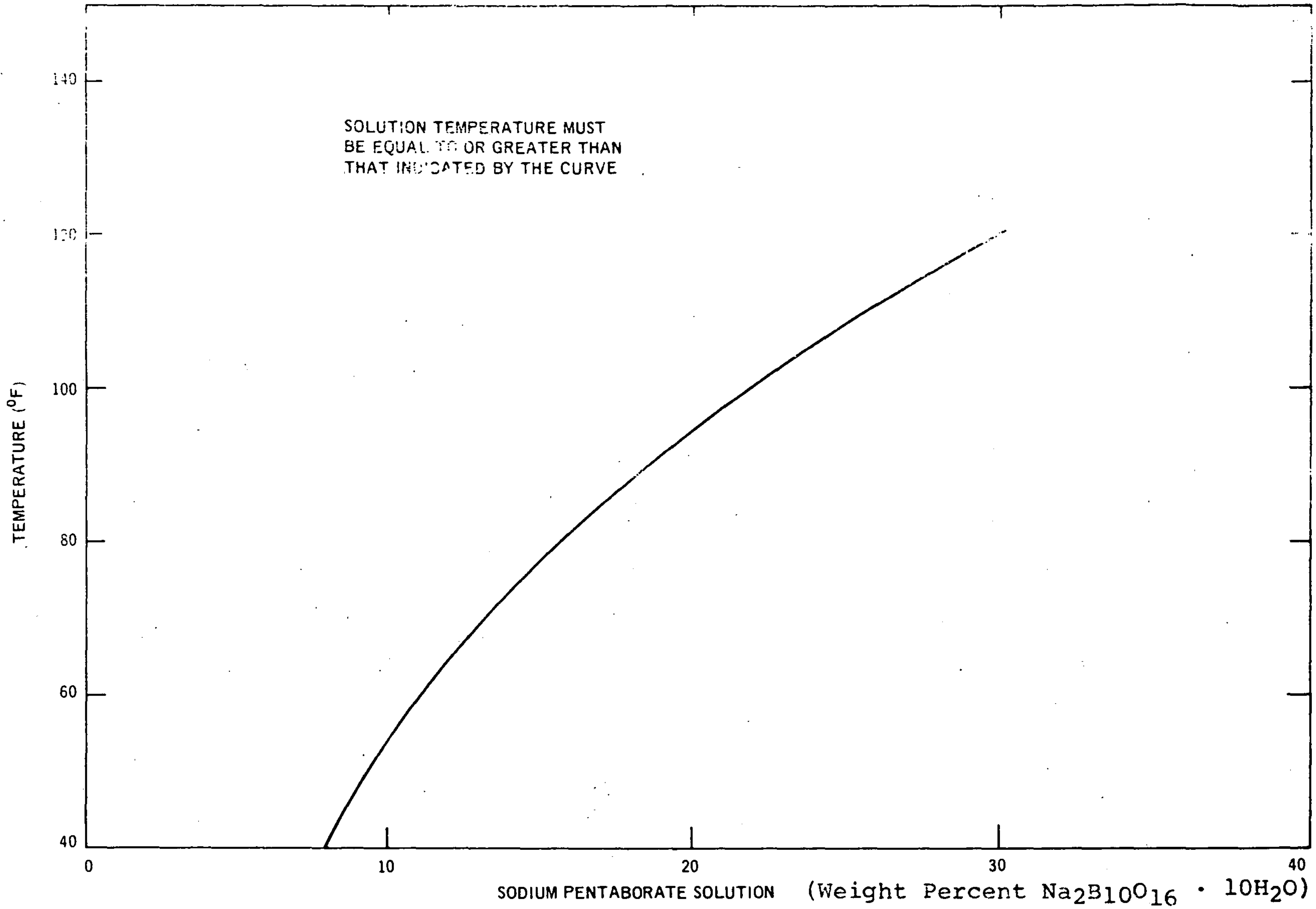

Figure 3.4.2 Sodium Pentaborate Solution Temperature Requirements

TABLE 4.8.1

SAMPLE COLLECTION AND ANALYSIS
DRESDEN NUCLEAR POWER STATION - ENVIRONMENTAL MONITORING PROGRAM

Sample Media	Type of Analysis	Collection Sites		Collection Frequency	Collection Dates
1. a. Airborne Particulate (AP)	Beta Gamma Scans (Special)	Elwood Joliet Wilmington Collins Road Morris Lisbon Coal City Bennett Farm Prairie Park Channabon Pheasant Trail Goose Lake Vil. Minooka Clay Products On -Site Stations #1, #2, #3	J-15 J-43 464 0773 016 024 J-68 BE PP CH PT 0672 J-27 J-21	Weekly	---
b. Airborne Screen (in addition to airborne particulate)	I-131	Same Locations as in 1. a.		Bi-Weekly	---
2. Gamma Background (Ion Chambers)	Gamma	Same Locations as Air Particulate Stations		Weekly	---

(Rev. w/Changes 18 and 10 dtd 12/4/72)

TABLE 4.8.1

SAMPLE COLLECTION AND ANALYSIS
DRESDEN NUCLEAR POWER STATION - ENVIRONMENTAL MONITORING PROGRAM

Sample Media	Type of Analysis	Collection Sites	Collection Frequency	Collection Dates
	Beta, Tritium, Gamma Scans (Special)	Dresden Well #1 (W1)	Quarterly	Jan.-Apr.-Jul.-Oct.
		Dresden Well #2 (W2)		Feb.-May-Aug.-Nov.
	Beta Gamma Scans (Special)	Drinking Fountain - Unit #1 (DF-1)		
11. Milk (M)	a.131-I b.89Sr, 90Sr, 137Cs	Davidson (DA) and Filotto (F)	a.Weekly b.Monthly Composite	---
	Elemental Calcium			
12. a. Fish b. Sediment c. Water	Beta, 89Sr, 90Sr Gamma Scan	Dresden Lock and Dam Pool (Routine) Brandon Lock and Dam Pool and County Line Bridge (Special)	Semi-Annual	---
d. Aquatic Plants	Beta, 89Sr, 90Sr, Gamma Scan	Dresden Inlet and Discharge Canal (General Area)	Semi-Annual	---

TABLE II

SAMPLE CODING SYSTEM
DRESDEN NUCLEAR POWER STATION - ENVIRONS PROGRAM

Sample Types

AP	Air Particulate
SW	Surface Water
WW	Well Water
WF	Fallout Water
SI	Silt
SL	Slime
M	Milk
GF	Grass
VF	Vegetation
CF	Cattle Feed
FF	Foodstuffs
SO	Soil
FP	Fish Program

Sample Location

J15	Elwood
J48	Joliet, Brandon Road
464	Wilmington
0773	Collins Road
016	Morris
024	Lisbon
J68	Coal City
BE	Bennett Farm

Sample Location

CH	Channahon
PT	Pheasant Trail
PP	Prairie Park
0672	Goose Lake Village
J27	Minooka
J21	Clay Products
A	On-Site Monitor Station #1
B	On-Site Monitor Station #2
C	On-Site Monitor Station #3
M	Morris (On Illinois River)
K	Kankakee River (At Inlet Canal)
D	DesPlaines River (At Discharge Canal)
RR	E. J. & E. Railroad Bridge (Ill. River)
MS	Morris (Illinois River - State)
DL	Dresden Locks
W1	Dresden Well #1
W2	Dresden Well #2
DF	In-Plant Drinking Fountain - Unit #1
TH	Thorsen Farm
AN	Anderson Farm
OL	Olson
DA	Davidson Farm
F	Filotto

(Rev. w/Changes 18 & 10 dtd 12/4/72)

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization, Review, Investigation and Audit

A. The Station Superintendent shall have overall full-time responsibility for safe operation of the facility. During periods when the Station Superintendent is unavailable, he shall designate this responsibility to the Assistant Station Superintendent.

B. The portion of the corporate management which relates to the operation of this station is shown in Figure 6.1.1.

C. The normal functional organization for operation of the station shall be as shown in Figure 6.1.2. The shift manning for the station shall be as shown in Figure 6.1.3.

5 D. Qualifications of the Dresden plant management and operating staff shall meet minimum acceptable levels as described in ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel", dated March 8, 1971.

E. Retraining and replacement training of Station personnel shall be in accordance with ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel", dated March 8, 1971.

F. Retraining shall be conducted at intervals not exceeding two years.

G. The Review and Investigative Function and the Audit Function for facility operations shall be constituted and have the responsibilities and authorities outlined below:

1. The Offsite Review and Investigative Function and Audit Function shall be supervised by the Superintendent of Nuclear and Fossil Systems.

a. Offsite Review and Investigative Function

5 The Superintendent of Nuclear and Fossil Systems shall: (i) provide direction for the review and investigative function and appoint a senior participant to provide appropriate direction, (ii) select each participant for this function, (iii) review and approve the findings and recommendations developed by personnel performing the review and investigative function, and (iv) report all findings of violations and provide recommendation to the Station Superintendent, Superintendent of Production Division "A", Manager of Production and that position of corporate management that has responsibility for nuclear activities.

The responsibilities of the personnel performing this function are stated below:

CORPORATE ORGANIZATION

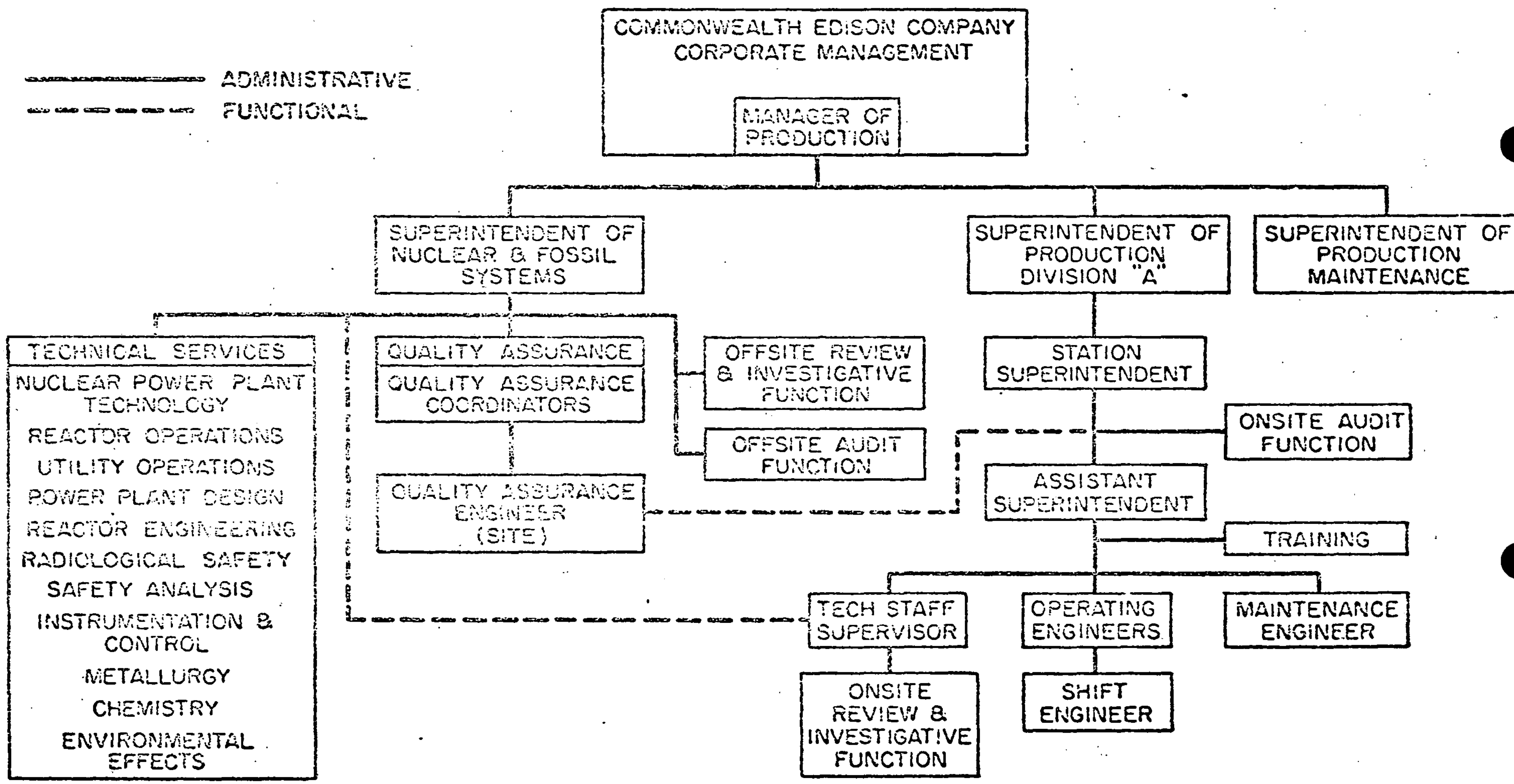

FIGURE 6.1.1

STATION ORGANIZATION CHART
(THREE UNITS AT HOT SHUTDOWN OR POWER)

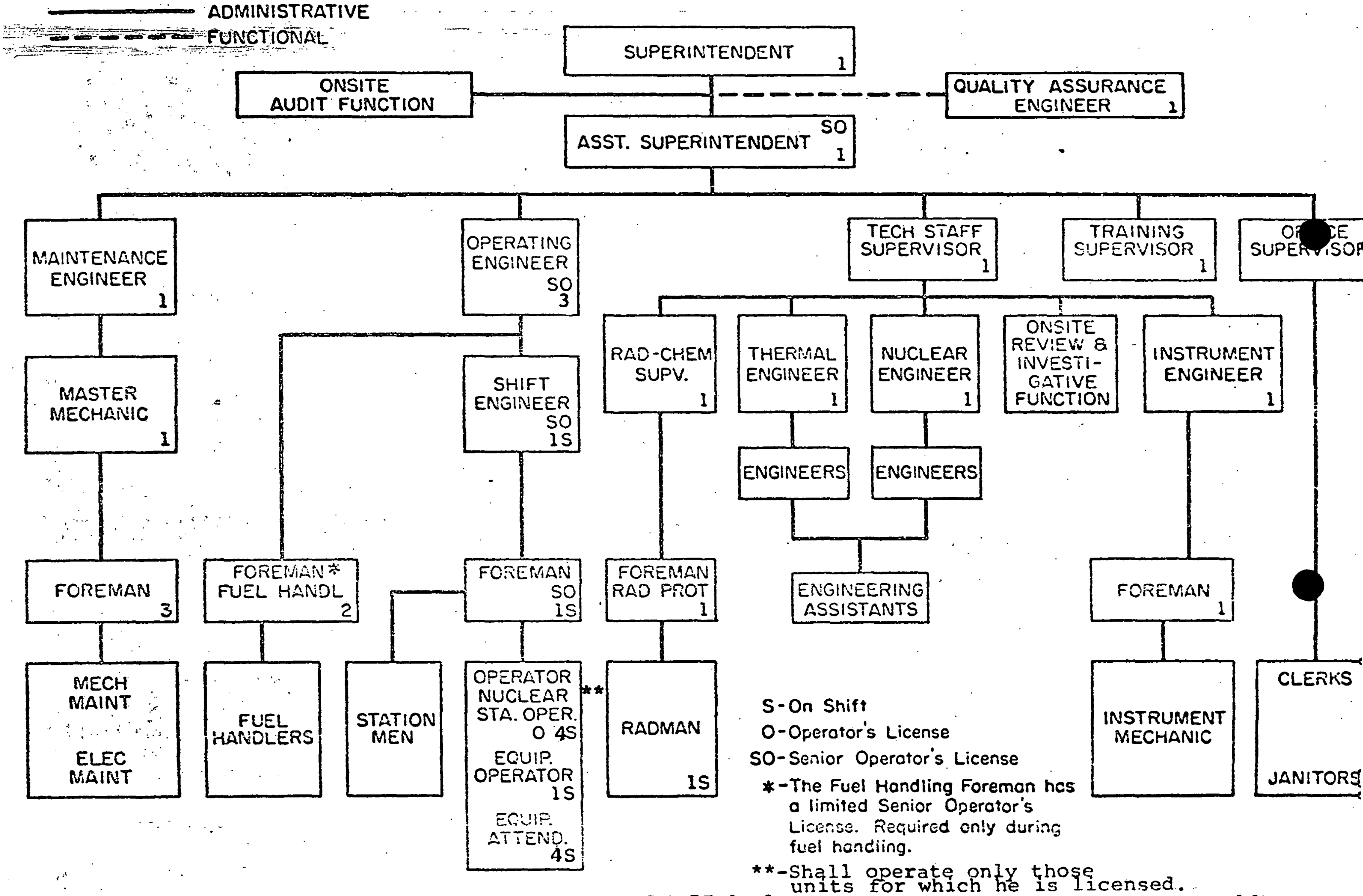

FIGURE 6.1.2

TABLE 6.1.1

SHIFT MANNING CHART

Condition of one Unit	Condition of Second Unit	Condition of Third Unit	No. of Men in Each Position SRO*	RO*	Non-Lic	Rad Men
COLD SHUTDOWN	Cold Shutdown	Cold Shutdown	1	1	2	1
	Cold Shutdown	Refuel	1	1	2	1
	Cold Shutdown	Above cold Shutdown	1	2	3	1
	Refuel	Refuel	2	2	3	1
	Refuel	Above Cold Shutdown	2	2	3	1
	Above Cold Shutdown	Above Cold Shutdown	2	2	4	1
REFUEL	Refuel	Refuel	2	3	3	1
	Refuel	Above Cold Shutdown	2	4	3	1
	Above Cold Shutdown	Above Cold Shutdown	2	4	4	1
ABOVE COLD SHUTDOWN	Above Cold Shutdown	Above Cold Shutdown	2	4	5	1

SRO - Senior Reactor Operator
RO - Reactor Operator
NON-LIC - Equipment Operators and Equipment Attendants
RAD MEN - Radiation Protection Men
* - Shall not operate units on which they are not licensed.

(1) Review and report findings and recommendations regarding proposed changes to the operating license of each unit, including technical specifications, analyses, and reports concerning unreviewed safety questions, and the Safety Analysis Report for submittal to the AEC.

(2) Review and report findings and recommendations regarding reports to be submitted to the AEC of proposed modifications to plant systems or components involving a proposed revision to the technical specifications or an unreviewed safety question.

(3) Advise in matters considered to involve unreviewed safety questions or changes to the license, technical specifications, or the Safety Analysis Report.

(4) Investigate reported instances of abnormal occurrences and violations of technical specifications, including review of recommendations to prevent a reoccurrence. Review and report recommendations regarding abnormal occurrence reports submitted to the AEC.

(5) Review and report findings and recommendations regarding all changes to the Generating Stations Emergency Plan prior to implementation of such change.

(6) Review and report findings and recommendations regarding all items referred by the Technical Staff Supervisor, Station Superintendent, and Superintendent of Production Division "A".

b. Offsite Audit Function

The Offsite Audit Function shall be directly supervised by the Superintendent of Nuclear and Fossil Systems or his designated alternate. He shall appoint the auditors and approve the findings and reports of each audit. The audit functions are itemized below:

(1) Perform, at least semiannually, audits of station operations including records, logs, reports, tests, procedures, and changes thereto which may affect safety or radiation exposure and verify that operations comply with the terms, conditions, and intent of licenses or permits and other applicable regulations.

(2) Approve, review, and audit the system of onsite audit of the station operations.

(d) Power Plant Design

Graduate in engineering or a scientific discipline with at least 5 years of experience in technical or technical management positions involving nuclear power plant design of which at least 3 years are related to the system under audit or investigation.

(e) Reactor Engineering

Graduate in engineering or a scientific discipline with at least one year additional academic work in nuclear engineering and/or nuclear physics relating to nuclear
5 power reactors. In addition, at least 5 years of experience in technical or technical management positions performing nuclear power plant engineering or technical support for operating nuclear power plants are required.

(f) Radiological Safety

Graduate in engineering or a scientific discipline and 5 years of experience as a technical member or supervisor of a radiation control organization of which at least 2 years have been directly associated with an operating nuclear power plant.

(g) Reactor Safety Analysis

Graduate in engineering or a scientific discipline with at least 5 years of experience in nuclear engineering of which at least 3 years have been in technical or technical management positions that perform reactor safety analyses of nuclear power plants.

(h) Instrumentation and Control

Graduate in engineering or a scientific discipline with at least 5 years of experience in instrumentation and control design or operation of which at least 3 years have been in technical or technical manage-
5 ment positions involving nuclear power plant instrumentation and controls.

(i) Metallurgy

Graduate in metallurgical engineering or in mechanical engineering with special training in metallurgy and at least 5 years' experience in technical or technical management positions in the metallurgical field including at least 3 years' experience related to nuclear power plants.

2. The Onsite Review and Investigative Function and Audit Function shall be supervised by the Station Superintendent.

a. <u>Onsite Review and Investigative Function</u>

The Station Superintendent shall: (i)
provide direction for the Review and
Investigative Function and appoint the
Technical Staff Supervisor as a senior
participant to provide appropriate direction,
(ii) select each participant for this
function, (iii) review and approve the
findings and recommendations developed by
personnel performing the Review and
Investigative Function, (iv) report all
findings of violations, and provide
5 recommendations to the Superintendent of
Production Division "A" and the
Superintendent of Nuclear and Fossil
Systems, and (v) submit to the Offsite
Review and Investigative Function for
concurrence those items described in
Specifications 6.1.G.1.a which have been
approved by the Onsite Review and
Investigative Function.

The responsibilities of the personnel performing this function are stated below:

(1) Review and report findings and recommendations regarding all station and company orders which affect operations.

(2) Review and report findings and recommendations regarding all tests and experiments, proposed to be performed at the station, which could involve hazards not previously evaluated in the Safety Analysis Report or which require evaluation to determine whether or not they are within the technical specifications.

(3) Review and report findings and recommendations regarding proposed changes to the Technical Specification, license, and Safety Analysis Report.

5 (4) Review and report findings and recommendations regarding proposed modifications to plant systems or equipment.

(5) Investigate reported instances of abnormal occurrences and violations of technical specifications and recommend corrective actions to prevent recurrences.

(6) Review plant operation and maintenance logs to detect potential safety hazards.

(7) Perform special reviews and investigations and rendor reports thereon as requested by the Superintendent of Production Division "A" and the Superintendent of Nuclear and Fossil Systems.

b. Onsite Audit Function

The Onsite Audit Function shall be directly supervised by the Station Superintendent. He shall appoint the auditors and approve the findings and reports of each audit. The audit functions are itemized below:

(1) Make at least quarterly audits of station operation including the review of a representative sample of records, logs, reports, tests, procedures and changes thereto as well as other items which may affect nuclear safety and radiation exposure and verify that the station operation complies with the terms, conditions and intent of licenses or permits and other applicable regulations. All station documents discussed above and relating to the safe operation of the station shall be audited at least once each calendar year.

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(2) Report all findings of violations and recommendations and results of each audit to the Superintendent of Production Division "A" and the Superintendent of Nuclear and Fossil Systems.

c. Authority - The Technical Staff Supervisor is responsible to the Station Superintendent and shall make recommendations in all areas of review, investigation, audit, and quality assurance phases of plant maintenance, operation and administrative procedures relating to facility operations and shall have the authority to request the action necessary to ensure compliance with rules, regulations, and procedures when in his opinion such action is necessary. The Station Superintendent shall follow such recommendations or select a course of action that is more conservative regarding safe operation of the facility. All such disagreements shall be reported immediately to the Superintendents of Production Division "A" and Nuclear and Fossil Systems.

d. Records

(1) Reports, reviews, investigations, audits, and recommendations shall be documented with copies to the Superintendents of Production Division "A", Nuclear and Fossil Systems, and the Station Superintendent.

(2) Copies of all records and documentation shall be kept on file at the station.

e. Procedures - Written administrative procedures shall be prepared and maintained for conduct of the Onsite Review and Investigative Function and Audit Function. These procedures shall include the following:

(1) Content and method of submission and presentation to the Station Superintendent, Superintendent of Production Division "A", and Superintendent of Nuclear and Fossil Systems.

(2) Use of committees.

(3) Review and approval

(4) Detailed listing of items to be reviewed.

(5) Procedures for administration of the Quality Assurance Program.

(6) Assignment of responsibilities.

F. <u>Personnel</u>

(1) The personnel performing the Onsite Review and Investigative Function and Audit Function, in addition to the Station Superintendent, shall consist of persons having expertise in:

(a) nuclear power plant technology
(b) reactor operations
(c) reactor engineering
(d) radiological and chemistry
(e) instrumentation and control
(f) mechanical and electric systems

(2) Personnel performing the Onsite Review and Investigative Function and Audit Function shall meet minimum acceptable levels as described in ANSI N18.1.

(3) The audits performed by the Onsite Audit Function shall be performed by personnel representing at any time no less than four of the technical disciplines in specification 6.1.G.2.f(1).

6.2 Plant Operating Procedures

A. Detailed written procedures including applicable checkoff lists covering items listed below shall be prepared, approved, and adhered to:

1. Normal startup, operation, and shutdown of the reactor and other systems and components involving nuclear safety of the facility.

2. Refueling operations.

5 3. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components including response to alarms, suspected primary system leaks, and abnormal reactivity changes.

4. Emergency conditions involving potential or actual release of radioactivity - "Generating Stations Emergency Plan" and station emergency and abnormal procedure

5. Instrumentation operation which could have an effect on the safety of the facility.

6. Preventive and corrective maintenance operations which could have an effect on 5 the safety of the facility.

7. Surveillance and testing requirements.

8. Tests and experiments.

9. A procedure to ensure safe shutdown of the plant in the event of a flood designated as a Probable Maximum Flood (PMF).

B. Radiation control procedures shall be maintained and made available to all station personnel. These procedures shall show per- 2 missible radiation exposure and shall be consistent with the requirements of 10 CFR 20 This radiation protection program shall be organized to meet the requirements of 10 CFR 20.

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c. The licensee advises each respirator user that he may leave the area at any time for relief from respirator use in case of equipment malfunction, physical or psychological discomfort, or any other condition that might cause reduction in the protection afforded the wearer.

d. The licensee maintains a respiratory protective program adequate to assure that the requirements above are met and incorporates practices for respiratory protection consistent with those recommended by the American National Standards Institute (ANSI-Z88.2-1969). Such a program shall include:

(1) Air sampling and other surveys sufficient to identify the hazard, to evaluate individual exposures,
2 to permit proper selection of respiratory protective equipment.

(2) Written procedures to assure proper selection, supervision, and training of personnel using such protective equipment.

(3) Written procedures to assure the adequate fitting of respirators and the testing of respiratory protective equipment for operability immediately prior to use.

(4) Written procedures for maintenance to assure full effectiveness of respiratory protective equipment including issuance, cleaning, and decontamination, inspection, repair, and storage.

(5) Written operational and administrative procedures for proper use of respiratory protective equipment including provisions for planned limitations on working times as necessitated by operational conditions.

(6) Bioassays and/or whole body counts of individuals (and other surveys, as appropriate) to evaluate individual exposures and to assess protection actually provided.

e. The licensee uses equipment approved by the U. S. Bureau of Mines under its appropriate Approval Schedules as set forth in Table 6.2.1 below. Equipment not approved under U. S. Bureau of
2 Mines Approval Schedules may be used only if the licensee has evaluated the equipment and can demonstrate by testing, or on the basis of reliable test information, that the material and performance characteristics of the equipment are at least equal to those afforded by U. S. Bureau of Mines approved equipment of the same type as specified in Table 6.2.1 below.

f. Unless otherwise authorized by the Commission, the licensee does not assign protection factors in excess of those specified in Table 6.2.1 below in selecting and using respiratory protective equipment.

TABLE 6.2.1
PROTECTION FACTORS FOR RESPIRATORS

DESCRIPTION	MODES[1/]	PROTECTION FACTORS[2/] PARTICULATES AND VAPORS AND GASES EXCEPT TRITIUM OXIDE[3/]	GUIDES TO SELECTION OF EQUIPMENT BUREAU OF MINES APPROVAL SCHEDULES* FOR EQUIPMENT CAPABLE OF PROVIDING AT LEAST EQUIVALENT PROTECTION FACTORS *or schedule superseding for equipment of type listed
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask 4/7/	NP	5	21B 30 CFR § 14.4(b)(4)
Facepiece, full 7/	NP	100	21B 30 CFR § 14.4(b)(5); 14F 30 CFR 13
2 II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Airline respirator			
Facepiece, half-mask	CF	100	19B 30 CFR § 12.2(c)(2) Type C(i)
Facepiece, full	CF	1,000	19B 30 CFR § 12.2(c)(2) Type C(i)
Facepiece, full 7/	D	100	19B 30 CFR § 12.2(c)(2) Type C(ii)
Facepiece, full	PD	1,000	19B 30 CFR § 12.2(c)(2) Type C(iii)
Hood	CF	5/	6/
Suit	CF	5/	6/
2. Self-contained breathing apparatus (SCBA)			
5 Facepiece, full 7/	D	100	13E 30 CFR § 11.4(b)(2)(i)
Facepiece, full	PD	1,000	13E 30 CFR § 11.4(b)(2)(ii)
Facepiece, full	R	1,000	13E 30 CFR § 11.4(b)(1)
III. COMBINATION RESPIRATOR Any combination of air-purifying and atmosphere-supplying respirator		Protection factor for type and mode of operation as listed above	19 B CFR § 12.2(e) or applicable schedules as listed above

1/, 2/, 3/, 4/, 5/, 6/, 7/ - [These notes are on the following pages]

1/ See the following symbols:

CF: continuous flow
D: demand
NP: negative pressure (i.e., negative phase during inhalation)
PD: pressure demand (i.e., always positive pressure)
R: recirculating (closed circuit)

2/ (a) For purposes of this specification, the protection factor is a measure of the degree of protection afforded by a respirator, defined as the ratio of the concentration of airborne radioactive material outside the respiratory protective equipment to that inside the equipment (usually inside the face-piece) under conditions of use. It is applied to the ambient airborne concentration to estimate the concentration inhaled by the wearer according to the following formula:

$$\text{Concentration Inhaled} = \frac{\text{Ambient Airborne Concentration}}{\text{Protection Factor}}$$

(b) The protection factors apply:

2 (i) only for trained individuals wearing properly fitted respirators used and maintained under supervision in a well-planned respiratory protective program.

(ii) for air-purifying respirators only when high efficiency [above 99.9% removal efficiency by U. S. Bureau of Mines type dioctyl phthalate (DOP) test] particulate filters and/or sorbents appropriate to the hazard are used in atmospheres not deficient in oxygen.

(iii) for atmosphere-supplying respirators only when supplied with adequate respirable air.

3/ Excluding radioactive contaminants that present an absorption or submersion hazard. For tritium oxide approximately half of the intake occurs by absorption through the skin so that an overall protection factor of not more than approximately 2 is appropriate when atmosphere-supplying respirators are used to protect against tritium oxide. Air-purifying respirators are not recommended for use against tritium oxide. See also footnote 5/, below, concerning supplied-air suits and hoods.

4/ Under chin type only. Not recommended for use where it might be possible for the ambient airborne concentration to reach instantaneous values greater than 50 times the pertinent values in Appendix B, Table I, Column 1 of 10 CFR, Part 20.

5/ Appropriate protection factors must be determined taking account of the design of the suit or hood and its permeability to the contaminant under conditions of use. No protection factor greater than 1,000 shall be used except as authorized by the Commission.

6/ No approval schedules currently available for this equipment. Equipment must be evaluated by testing or on basis of available test information.

7/ Only for shaven faces.

NOTE 1: Protection factors for respirators, as may be approved by the U. S. Bureau of Mines according to approval schedules for respirators to protect against airborne radionuclides, may be used to the extent
2 that they do not exceed the protection factors listed in this Table. The protection factors in this Table may not be appropriate to circumstances where chemical or other respiratory hazards exist in addition to radioactive hazards. The selection and use of respirators for such circumstances should take into account approvals of the U. S. Bureau of Mines in accordance with its applicable schedules.

5 NOTE 2: Radioactive contaminants for which the concentration values in Appendix B, Table I of 10 CFR Part 20 are based on internal dose due to inhalation may, in addition, present external exposure hazards at higher concentrations. Under such circumstances, limitations on occupancy may have to be governed by external

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2 3. These specifications with respect to the provisions of paragraph 20.103 shall be superseded by adoption of proposed changes to 10 CFR 20, Section 20.103, which would make this specification unnecessary.

C. Standing Orders to the operating staff shall require that the procedures in Specifications
5 6.2.A and B above are to be followed in conducting activities identified in the order.

D. Work instructions or special test procedures for the operating or maintenance staff shall require that the procedures in Specifications 6.2.A and B above are to be followed in conducting activities identified therein.

E. All procedures identified in Specification 6.2.A and any changes to those procedures shall be reviewed and approved by the Operating Engineer and the Technical Staff Supervisor in areas of operation, fuel handling, or instrument maintenance, and by Maintenance Engineer and the Technical Staff Supervisor in the areas
5 of plant maintenance and plant inspection. All procedures identified in Specification 6.2.B and any changes to those procedures shall be reviewed and approved by the Technical Staff Supervisor and the Radiological-Chemical Supervisor. The procedures, and changes thereto, must have authorization by the Station Superintendent before being implemented.

F. Temporary changes to operating procedures described in Specification 6.2.A above, which do not change the intent of the original procedure, may be made with the concurrence of two individuals holding Senior Operator Licenses. Temporary changes to electrical and mechanical maintenance procedures described in Specification 6.2.A above, which do not change the intent of the original procedures may be made with the concurrence of the Master Mechanic, Maintenance Foreman, and the Senior Operator of the affected unit. Temporary changes to instrument maintenance procedures described in Specification 6.2.A above, which do not change the intent of the original procedures, may be made with the concurrence of the Instrument Engineer, Instrument Foreman, and the Senior Operator of the affected unit. Such changes shall be documented and subsequently reviewed, approved, and authorized as provided in
5 Specifications 6.2.E above.

G. Drills of the emergency procedures describe in Specification 6.2.A.4 shall be conducted quarterly. These drills will be planned so that during the course of the year communication links are tested and outside agencie are contacted.

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6.3 Action to be Taken in the Event of an Abnormal Occurrence in Plant Operation

Any abnormal occurrence shall be promptly reported to the Superintendent of Production Division "A" or his designated alternate. The incident shall be promptly reviewed pursuant to Specification 6.1.G.2.a.(5) and a separate report for each abnormal occurrence shall be prepared in accordance with the requirements of Specification 6.6.B.1.

6.4 Action to be Taken in the Event a Safety Limit
5 is Exceeded

If a safety limit is exceeded, the reactor shall be shut down immediately and reactor operation shall not be resumed until authorized by the AEC. The conditions of shutdown shall be promptly reported to the Superintendent of Production Division "A" or his designated alternate. The incident shall be reviewed pursuant to Specification 6.1.G.2.a.(5) and a separate report for each occurrence shall be prepared in accordance with Specification 6.6.B.1.

6.5 Plant Operating Records

A. Records and/or logs relative to the following items shall be kept in a manner convenient for review and shall be retained for at least five years.

1. Records of normal plant operation, including power levels and periods of operation at each power level.
2. Records of principal maintenance and activities, including inspection and repair, regarding principal items of equipment pertaining to nuclear safety.
3. Records and reports of abnormal and safety limit occurrences.
4. Records and periodic checks, inspection and/or calibrations performed to verify the Surveillance Requirements (see Section 4 of these
5 Specifications) are being met. All equipment failing to meet surveillance requirements and the corrective action taken shall be recorded.
5. Records of changes made to the equipment or reviews of tests and experiments to comply with 10 CFR 50.59.

6. Records of radioactive shipments.

7. Records of physic tests and other tests pertaining to nuclear safety.

5 8. Records of changes to operating procedures.

9. Shift Engineers Logs.

B. Records and/or logs relative to the following items shall be recorded in a manner convenient for review and shall be retained for the life
5 of the plant.

1. Substitution or replacement of principal items of equipment pertaining to nuclear safety.

2. Changes made to the plant as it is described in the Safety Analysis Report.

3. Records of new and spent fuel inventory and assembly histories.

4. By-product material inventory records and source leak test results.

5 5. Updated, corrected, and as-built drawings of the plant.

6. Records of plant radiation and contamination surveys.

7. Records of off-site environmental monitoring surveys.

8. Records of radiation exposure for all plant personnel, including all contractors and visitors to the plant in accordance with 10 CFR 20.

9. Records of radioactivity in liquid and gaseous wastes released to the environment.

10. Records of transient or operational cycling for those components that have been designed to operate safely for a limited number of
5 transient or operational cycles.

11. Records of individual staff members indicating qualifications, experience, training, and retraining.

12. Inservice inspections of the reactor coolant system.

13. Minutes of meetings and results of reviews and audits performed by the off-site and on-site review and audit functions.

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6.6 Plant Reporting Requirements

4 The following information shall be submitted in addition to those reports required by Title 10, Code of Federal Regulations.

A. Operation Reports

5 Operation reports shall be submitted in writing to the Director of Licensing, USAEC, Washington, D. C. 20545.

1. Startup Report

4 A summary report of unit startup and power escalation testing shall be submitted following receipt of operating licenses, following amendments to the licenses involving a planned increase in power level, following the installation of fuel that has a different design or was fabricated by a different vendor, or following NSSS modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the facility. The report shall include a comparison of measured and predicted values and describe any corrective action taken to obtain acceptable operation. Startup reports shall be submitted within 60 days following commencement or resumption of commercial power operation (i.e., initially following synchronization of the turbo-generator to produce commercial power or resuming power production).

2. First Year Operation Report

A report shall be submitted within 60 days after completion of the first year of commercial power operation as defined above. This report may be incorporated into the semiannual operating report and shall cover the following:

a. an evaluation of unit performance to date in comparison with design predictions and specifications;

4 b. a reassessment of the safety analysis submitted with the license application in light of measured operating characteristics when such measurements indicate that there may be substantial variance from prior analyses;

c. an assessment of the performance of structures, systems and components important to safety;

d. a progress and status report on any items identified as requiring additional information during the operating license review or during the startup of the plant, including items discussed in the AEC's safety evaluation, items on which additional information was required as conditions of the license and items identified in the licensee's startup report.

3. Semiannual Operating Reports

Semiannual operating reports covering the previous six months operations shall be submitted within 60 days after January 1 and July 1 of each year. The first such period shall begin with the date of initial
4 criticality. These reports shall include the following:

a. Operations Summary

A summary of operating experience occurring during the reporting period that relates to the safe operation of the plant, including a summary of:

(1) changes in plant design,

(2) performance characteristics (e.g., equipment and fuel performance),

(3) changes in procedures which were necessitated by (1) and (2) or which otherwise were required to improve the safety of facility operations,

(4) results of surveillance tests and inspections required by these technical specifications,

(5) the results of any periodic containment leak rate tests performed during the reporting period,

(6) a brief summary of those changes, tests and experiments requiring authorization from the Commission pursuant to
4 10 CFR 50.59(a), and

(7) any changes in the plant operating organization which involve positions which are designated as key supervisory personnel on Figure 6.1.2.

b. Power Generation

A summary of power generated during the reporting period including:

(1) gross thermal power generated (in MWH)

(2) gross electrical power generated (in MWH)

(3) net electrical power generated (in MWH)

(4) number of hours the reactor was critical

(5) number of hours the generator was on-line

(6) histogram of thermal power vs time

c. Shutdowns

Descriptive material covering all outages occurring during the reporting period. For each outage, information
4 shall be provided on:

(1) the cause of the outage,

(2) the method of shutting down the reactor; e.g., trip automatic rundown, or manually controlled deliberate shutdown,

(3) duration of the outage (in hours),

(4) unit status during the outage; e.g., cold shutdown or hot shutdown,

(5) corrective action taken to prevent repetition, if appropriate.

d. Maintenance

A discussion of corrective maintenance (excluding preventative maintenance) performed during the reporting period on safety related systems and components [safety related is defined in ANSI 18.7-1972 (ANS-3.2, November 2, 1972)].
5 For any malfunctions for which corrective maintenance was required, information shall be provided on:

(1) the system or component involved,

(2) the cause of the malfunction,

(3) the results and effect on safe
4 operation,

(4) corrective action taken to prevent repetition,

(5) precautions taken to provide for reactor safety during repair.

e. Changes, Tests and Experiments

The report shall include a brief description and the summary of the safety evaluation for those changes,
5 tests, and experiments carried out without prior Commission approval pursuant to the requirements of subsection 50.59(b) of the Commission's regulations.

f. Radioactive Effluent Releases

A statement of the quantities of radioactive effluents released from
4 the plant with data summarized on a monthly basis following the format
5 of the USAEC Regulatory Guide 1.21.

(1) Gaseous Effluents

(a) Gross Radioactivity Releases

(1) Total gross radioactivity (in curies) primarily noble and activation gases released.

(2) Maximum gross radioactivity release rate during any one-hour period.

(3) Total gross radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed.

4 (4) Percent of technical specification limit.

(b) Iodine Releases

(1) Total iodine radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed.

(2) Percent of technical specification limit for I-131 released.

(c) Particulate Releases

(1) Total gross radioactivity (β,γ) released (in curies) excluding background radioactivity.

(2) Total gross alpha radioactivity released (in curies) excluding background radioactivity.

(3) Total gross radioactivity released (in curies) of nuclides with half-lives greater than eight days.

(4) Percent of technical specification limit
4 for particulate radioactivity with half-lives greater than eight days.

(2) Liquid Effluents

(a) Total gross radioactivity (β,γ) released (in curies) excluding tritium and average concentration released to the unrestricted area.

(b) Total tritium and total alpha radioactivity released (in curies) and average concentration released to the unrestricted area.

(c) Total dissolved noble gas radioactivity released (in curies) and average concentration released to the unrestricted area.

(d) Total volume (in liters) of liquid waste released.

(e) Total volume (in liters) of dilution water used prior to release from the restricted area.

4 (f) The maximum concentration of gross radioactivity (β,γ) released to the unrestricted area (averaged over the period of release).

(g) Total gross radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed.

(h) Percent of technical specification limit.

g. Solid Radioactive Waste

(1) The total amount of solid waste shipped (in cubic feet).

(2) The total estimated radioactivity (in curies) involved.

(3) The dates of shipment and disposition (if shipped offsite).

h. Environmental Monitoring

(1) For each medium sampled during the reporting period, e.g., air, baybottom, surface water, soil,
4 fish, include:

(a) Number of sampling locations,

(b) Total number of samples,

(c) Number of locations at which levels are found to be significantly above local backgrounds,

(d) Highest, lowest, and the average concentrations or levels of radiation for the sampling point with the highest average and description of the location of that point with respect to the site.

(2) If levels of radioactive materials 5
in the environmental media as determined by the environmental monitoring program indicate the likelihood of public intakes in excess of 1% of those that could result from continuous annual
exposure to the concentration 4
values listed in Appendix B, Table II, Part 20, estimates of the likely resultant exposure to individuals and to population
4 groups, and assumptions upon
which estimates are based shall 5
be provided.

(3) If statistically significant variations of offsite environmental concentrations with time are observed, correlation of these results with radioactive
effluent releases shall be provided. 4

4. Occupational Personnel Radiation Exposure

5 Tabulate the number of personnel exposures for plant operations personnel (permanent and temporary) in the following exposure increments for the reporting period: Less than 100 mRem, 100-500 mRem, 500-1250 mRem, 1250-2500 mRem, and greater than 2500 mRem. Tabulate the number of personnel receiving more than 500 mRem exposure in the reporting period according to duty function, i.e., routine plant surveillance and inspection (regular duty), routine plant maintenance, special plant maintenance (describe maintenance), routine refueling operations, special refueling operation (described operation) and other job related exposures. Anually tabulate the number of personnel receiving more than 2500 mRem and report major cause(s).

B. Non-Routine Reports

1. Abnormal Occurrence Reports

a. Notification shall be made within 24 hours by telephone and telegraph to the Director of the Regional Regulatory Operations Office (cc to Director of Licensing) followed by a written report within 10 days to the Director of Licensing (cc to the Director of the Regional Regulatory Operations Office) in the event of the abnormal occurrences as defined in Section 1.0 of these technical specifications. The written report on these abnorma

5 occurrences, and to the extent possible the preliminary telephone and telegraph notification shall include the events leading up to and resulting from the occurrence, an evaluation of the cause of the occurrence, and: (a) describe, analyze and evaluate safety implications, (b) outline the measures taken to assure that the cause of the condition is determined, (c) indicate the corrective action (including any changes made to the procedures and to the quality assurance program) taken to prevent repetition of the occurrence and of similar occurrences involving similar components or systems, and (d) evaluate the safety implications of the incident in light of the cumulative experience obtained from the record of previous failures and malfunctions of similar systems and components.

b. Copies of all such reports shall be submitted to the Superintendent of Production Division "A" and Nuclear and Fossil Systems for review of any recommendations.

2. Unusual Events Reports

4 A written report shall be forwarded within 30 days to the Director of Licensing, and to the Director of the Regional Regulatory Operations Office, in the event of:

a. Discovery of any substantial errors in the transient or accident analyses, or in the methods used for such analyses, as described in the Safety Analysis Report or in the bases for the technical specifications.

4 b. Discovery of any substantial variance from performance specifications contained in the technical specifications or in the Safety Analysis Report.

c. Discovery of any condition involving a possible single failure which, for a system designed against assumed single failures, could result in a loss of the capability of the system to perform its safety function.

3. Special Reports

Special reports shall be submitted in writing within 90 days to the Director of Licensing, USAEC, Washington, D. C. 20545.

5 a. In the event a redundant component (or system) covered by these technical specifications is determined to be out of service for periods longer than those specified in other sections, it shall be the subject of a special maintenance report. If this report requests approval to continue plant operation, it shall be

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submitted prior to expiration of the out of service time of the above. The report shall describe:

(1) The nature of the problem and the specific steps to be taken to remedy the situation.

(2) An estimate of the time required to return the component (or system) to an operable condition.

(3) The amount of component (or system) redundancy remaining or the availability of other system(s) to perform the same function as the inoperable component (or system).

(4) Surveillance Requirements on the operable components (or systems).

b. Any significant changes in the information supplied in Specifications 6.6.B.3.a(1), (2), (3) or (4) shall be submitted as a report
5 within seven days of discovery.

c. Reports on the following areas shall be as indicated in Table 6.6.2.

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TABLE [illegible].2

[illegible] REPORTS

	AREA	SPECIFICATION REFERENCE	SUBMITTAL DATE
a.	Primary Containment Leak Rate Test (1)	4.7.A	Upon completion of each test.
b.	Secondary Containment Leak Rate Test (2)	4.7.C	Upon completion of each test.
c.	Summary Status of Fuel Performance	1.1 Bases	After each refueling outage starting with second refueling outage.
d.	Primary Coolant Leakage to Drywell	4.6.D Bases	2 years (3)
e.	In-Service Inspection Evaluation	Table 4.6.1	5 years (3)
f.	Materials Radiation Surveillance Specimens	4.6.B.2	After each specimen removal and completion of analyses.
g.	Evaluation of ADS operation	3.3.F Bases	Upon completion of initial testing.

NOTES:

1. Each integrated leak rate test of the primary containment shall be the subject of a summary technical report including results of the local leak rate tests since the last report.

2. Each integrated leak rate test of the secondary containment shall be the subject of a summary technical report. This report should include date on the wind speed, wind direction, outside and inside temperatures during the test, concurrent reactor building pressure, and emergency ventilation flow rate. The report shall also include analyses and interpretations of those data which demonstrate compliance with the specified leak rate limits.

3. The report shall be submitted within the period of time listed based on the commercial service date as the starting point.