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SAFETY EVALUATION FOR CHANGES

TO TECHNICAL SPECIFICATION SECTION 3/4.0

BACKGROUND

As a result of the issuance of the Unit 2 Operating License (including technical specifications) certain changes need to be made in order to make Unit 1 Technical Specifications consistent with Unit 2 Technical Specifications. Additional changes need to be made as a result of recent NRC directives.

REFERENCES

Technical Specifications, Section 3/4.0.

BASES

In general, subject changes reflect NRC requirements or duplicate existing Unit 2 Technical Specification sections.

CONCLUSION

The proposed changes do not involve an unreviewed safety question as defined by 10CFR50.59.

SAFETY EVALUATION FOR CHANGES

TO TECHNICAL SPECIFICATION SECTION 6

BACKGROUND

On October 23, 1980, an operating license (including technical specifications) was issued for Farley Nuclear Plant - Unit 2. Section 6 of the Unit 2 Technical Specifications was changed considerably from Section 6 of the Unit 1 Technical Specifications in order to reflect additional NRC requirements.

REFERENCES

Unit 2 Technical Specifications, Section 6.

BASES

The proposed changes primarily duplicate Section 6 of the Unit 2 Technical Specifications; however, certain changes were made in the interest of more efficient administrative controls for Farley Nuclear Plant.

CONCLUSION

The proposed changes do not involve an unreviewed safety question as defined by 10CFR50.59.

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour ACTION shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT STANDBY within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual specifications.

3.0.5 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant systems(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied within 2 hours, ACTION shall be initiated to place the unit in a MODE in which the applicable Limiting Condition for Operation does not apply by placing it, as applicable, in:

APPLICABILITY

SURVEILLANCE REQUIREMENTS

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

This specification is not applicable in MODES 5 or 6.

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, and

b. The combined time interval for any 3 consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2 and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the ACTION to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specificaton 3.5.1 requires each Reactor Coolant System accumulator to be OPERABLE and provides explicit ACTION requirements if one accumulator is inoperable. Under the terms of Specification 3.0.3, if more than one accumulator is inoperable, the unit is required to be in at least HOT STANDBY within 1 hour and in at least HOT SHUTDOWN within the following 6 hours. As a further example, Specification 3.6.2.1 requires two Containment Spray Systems to be OPERABLE and provides explicit ACTION requirements if one spray system is inoperable: Under the terms of Specification 3.0.3, if both of the required Containment Spray Systems are inoperable, the unit is required to be in at least HOT STANDBY within 1 hour, in at least HOT SHUTDOWN within the following 6 hours and in at least COLD SHUTDOWN in the next 30 hours. It is assumed that the unit is brought to the required mode within the required times by promptly initiating and carrying out the appropriate ACTION statement.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to insure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

3.0.5 This specification delineates what additional conditions must be satisifed to permit operation to continue, consistent with the ACTION statements for power sources, when a normal or emergency power source is not OPERABLE. It specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason.

The provisions of this specification permit the ACTION statements associated with individual systems, subsystems, trains, components, or devices to be consistent with the ACTION statements of the associated electrical power source. It allows operation to be governed by the time limits of the ACTION statement associated with the Limiting Condition for Operation for the normal or emergency power source, not the individual ACTION statements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source.

For example, Specification 3.8.1.1 requires in part that two emergency diesel generators be OPERABLE. The ACTION statement provides for a 72 hours out-of-service time when one emergency diesel generator is not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable emergency power source would also be inoperable. This would dictate invoking the applicable ACTION statement for each of the applicable Limiting Conditions for Operation. However, the provisions of Specification 3.0.5 permit the time limits for continued operation to be consistent with the ACTION statement for the inoperable emergency diesel generator instead, provided the other specified conditions are satisfied. In this case, this would mean that the corresponding normal power source must be OPERABLE, and all redundant systems, subsystems, trains, components, and devices must be OPERABLE, or otherwise satisfy Specification 3.0.5 (i.e., be capable of performing their design function and have at least one normal or one emergency power source OPERABLE). If they are not satisfied, action is required in accordance with this specification.

As a further example, Specification 3.8.1.1 requires in part that two physically independent circuits between the offsite transmission network and the onsite Class IE distribution system be OPERABLE. The ACTION statement provides a 24 hours out-of-service time when both required offsite circuits are not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable normal power sources, both of the offsite circuits, would also be inoperable. This would dictate invoking the applicable ACTION statements for each of the applicable LCOs. However, the provisions of Specification 3.0.5 permit the time limits for continued operation to be consistent with the ACTION statement for the inoperable normal power sources instead, provided the other specified conditions are satisfied. In case, this would mean that for one division the emergency power source must be OPERABLE (as must be the components supplied by the emergency power source) and all redundant systems, subsystems, trains, components and devices in the other division must be OPERABLE, or likewise satisfy Specification 3.0.5 (i.e., be capable of performing their design functions and have an emergency power source OPERABLE). In other words, both emergency power sources must be OPERABLE and all redundant systems, subsystems, trains, components and devices in both divisions must also be OPERABLE. If these conditions are not satisfied, action is required in accordance with this specification.

In MODES 5 or 6 Specification 3.0.5 is not applicable, and thus the individual ACTION statements for each applicable Limiting Condition for Operation in these MODES must be adhered to.

4.0.1 This specification provides that surveillance activities necessary to insure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specificaton.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over 3 test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these technical specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these technical specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATING MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

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

6.1.2 The Shift Supervisor or during his absence from the Control Room a designated individual shall be responsible for the Control Room Command function. A management directive to this effect, signed by the Senior Vice President-Alabama Power Company shall be reissued on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2-1.

FACILITY STAFF

6.2.2 The facility organization shall be as shown on Figure 6.2-2 and:

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

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

c. A health physics technician# shall be on site when fuel is in the reactor.

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

e. A site Fire Brigade of at least 5 members shall be maintained onsite at all times.# The Fire Brigade shall not include 3 members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

#The health physics technician and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of the health physics technician and/or Fire Brigade members provided immediate action is taken to restore the health physics technician and/or Fire Brigade to within the minimum requirements.

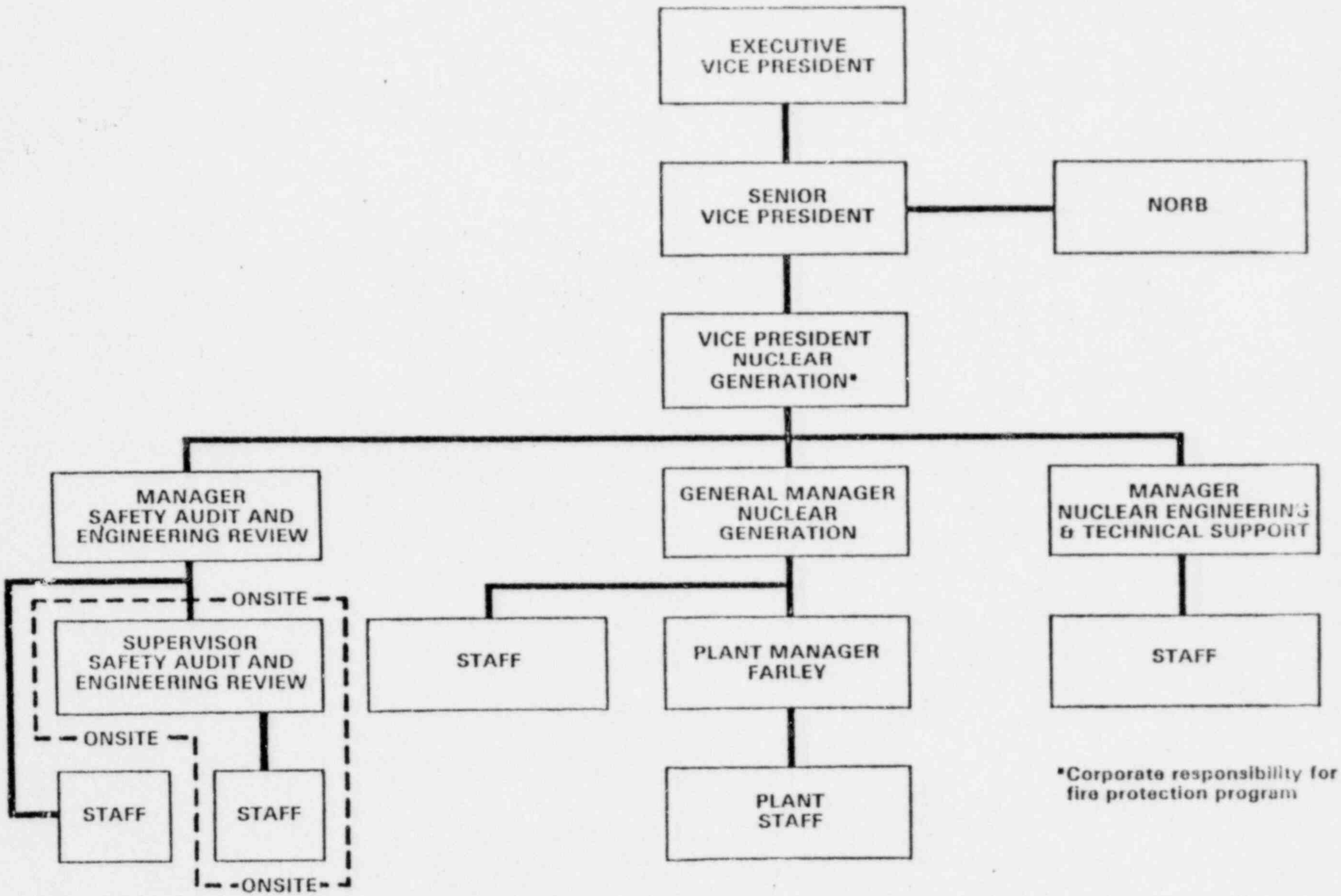

Figure 6.2-1 Offsite Organization for Facility Management and Technical Support

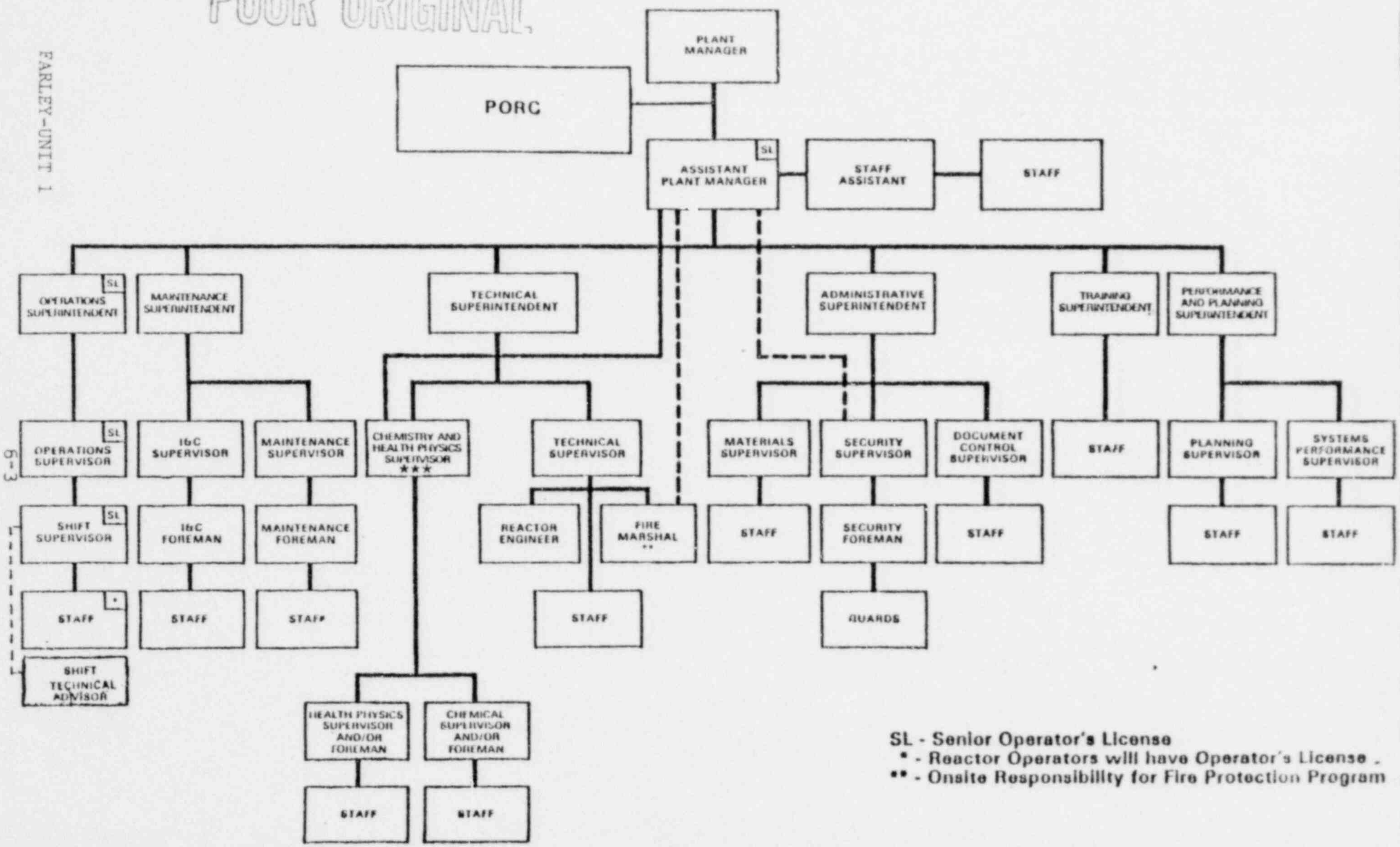

Figure 6.2-2 Facility Organization — Joseph M. Farley - Unit No. 1

***In routine matters, the Chemistry and Health Physics Supervisor reports directly to the Technical Superintendent, in matters of radiation policy determination, interpretation or implementation (based upon the Chemistry and Health Physics Supervisor's judgment) the Chemistry and Health Physics Supervisor may report directly to the Assistant Plant Manager.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION
FARLEY UNIT 1

WITH UNIT 2 IN MODE 5 OR 6 DE-FUELED		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SS	1[a/c]	1[a/c]
SRO	1[a/d]	none
RO	2	1
AO	2	2[b]
STA	1	none

WITH UNIT 2 IN MODES 1, 2, 3 OR 4		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SS	1[a/c]	1[a/c]
SRO	1[a/d]	none
RO	2[b]	1
AO	2[b]	1
STA	1[a]	none

a/ Individual may fill the same position on Unit 2.

b/ One of the two required individuals may fill the same position on Unit 2.

c/ A Shift Supervisor will be assigned to each unit who is licensed on that unit.

d/ This position at a minimum will be licensed on Unit 1. As a part of his responsibilities he will perform routine inplant equipment inspections and walkdowns in Unit 1 and report the results to the Unit 1 Shift Supervisor. These inspections and reporting requirements will be strictly controlled and incorporated in administrative procedures.

TABLE 6.2-2 (Continued)

SS - Shift Supervisor with a Senior Reactor Operators License on Unit 1
SRO - Individual with a Senior Reactor Operators License on Unit 1
RO - Individual with a Reactor Operators License on Unit 1
AO - Auxiliary Operator
STA - Shift Technical Advisor

The Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 1, 2, 3, 4, 5 or 6, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room command function and shall remain in the Control Room until the Shift Supervisor returns and reassumes the command function.

The shift crew individuals indicated in Table 6.2-1 shall not be permitted to work more than*:

1. 12 hours straight,
2. 24 hours in any 48 hours period
3. 72 hours in any 7-day period
4. 14 consecutive days without having 2 consecutive days off.

---

*Deviaiton from these requirements may be authorized by the Plant Manager or higher levels of management shown on Figure 6.2-1 in accordance with established procedures and with documentation of the cause. Overtime limits do not include shift turnover time.

d/ Refer to note d/ on page 6-4.

6.2.3 SAFETY AUDIT AND ENGINEERING REVIEW GROUP (SAERG)

FUNCTION

6.2.3.1 The SAERG shall function to conduct operational evaluatiions, engineering reviews, and audits for the purpose of improving safety.

COMPOSITION

6.2.3.2 The SAERG shall be composed of a multi-disciplined dedicated onsite group with a minimum assigned complement of five engineers or appropriate specialists.

RESPONSIBILITIES

6.2.3.3 The SAERG shall be responsible for the following:

a. Participating in operational evaluations for improvement of safety wherein such evaluations and recommendations therefrom are not limited to the fulfillment of existing programs, policies, procedures, or capabilities of existing equipment and installations.

b. Systematic engineering reviews of plant performance and activities with results reported independently of onsite operational management to offsite upper management.

c. Comprehensive plant audits in accordance with audit requirements set forth in quality assurance programs, licensing documents, and other policies and procedures.

AUTHORITY

6.2.3.4 The onsite SAERG shall carry out its function reporting offsite directly to the Manager-Safety Audit and Engineering Review who in turn reports directly to the Vice President-Nuclear Generation.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall serve in an advisory capacity to the Shift Supervisor primarily in the assessment of accident and transient occurrences.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, except for (1) the Chemistry and Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Superintendent and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N13.1-1971 and Appendix "A" of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with the relevant operational experience.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Training Superintendent and shall meet or exceed the requirements of Section 27 of the NFPA Code-1976, except for Fire Brigade training sessions which shall be held at least quarterly.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The PORC shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PORC shall be composed of the:

Chairman:	Plant Manager
Vice Chairman	Assistant Plant Manager
Member:	Operations Superintendent
Member:	Technical Superintendent
Member:	Maintenance Superintendent
Member (Non-Voting):	Supervisor-Safety Audit and Engineering Review
Member:	Performance and Planning Superintendent

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PORC Chairman to serve on a temporary basis; however, no more than one alternate shall participate as a voting member in PORC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PORC shall meet at least once per calendar month and as convened by the PORC Chairman or Vice Chairman.

QUORUM

6.5.1.5 The minimum quorum of the PORC necessary for one performance of the PROC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or Vice Chairman and two voting members including alternates.

RESPONSIBILITIES

6.5.1.6 The PORC shall review:

a. All administrative procedures and changes thereto,

b. The safety evaluations for 1) procedures, 2) changes to procedures, equipment or systems, and 3) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

c. Proposed procedures and changes to procedures, equipment or systems which may involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed tests or experiments which may involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

e. Proposed changes to Technical Specifications or this Operating License.

f. Reports of violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions, having nuclear safety significance or reports of abnormal degradation of systems designed to contain radioactive material.

g. Reports of significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

h. All written reports concerning events requiring 24 hour notification to the Commission.

i. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

j. The plant Security Plan.

k. The Emergency Plan.

l. Facility operations to detect potential nuclear safety hazards.

m. Investigations or analyses of special subjects as requested by the Chairman of the Nuclear Operations Review Board.

n. Every unplanned onsite release of radioactive material to the environs, as defined in Sections 6.9.1.12 and 6.9.1.13, including the preparation and forwarding of reports covering evaluation recommendations and disposition of the corrective action to prevent recurrence to the Plant Manager and to the Nuclear Operations Review Board.

o. Changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL.

AUTHORITY

6.5.1.7 The PORC shall:

a. Recommend to the Plant Manager in writing, approval or disapproval of items considered under 6.5.1.6(a) through (e) and (j) and (k) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a), (c) and (d) above constitutes an unreviewed safety question.

c. Make recommendations to the Plant Manager in writing that actions reviewed under 6.5.1.6(b) above did not constitute an unreviewed safety question.

RECORDS

6.5.1.8 The PORC shall maintain written minutes of each meeting and copies shall be provided to the Vice President-Nuclear Generation and Chairman of the Nuclear Operations Review Board.

6.5.2 NUCLEAR OPERATIONS REVIEW BOARD (NORB)

FUNCTION

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

a. Nuclear power plant operations
b. Nuclear engineering
c. Chemistry and radiochemistry
d. Metallurgy
e. Instrumentation and control
f. Radiological safety
g. Mechanical and electrical engineering
h. Quality assurance practices

COMPOSITION

6.5.2.2 The NORB shall be composed of at least five persons including:

Chairman:	Senior Vice President
Vice Chairman:	Vice President-Nuclear Generation
Secretary:	Manager-Safety Audit & Engineering Review
Member:	General Manager-Nuclear Generation
Member:	Manager-Nuclear Engineering and Technical Support

and other appointed personnel having an academic degree in an engineering or physical science field and a minimum of five years technical experience of which a minimum of three years shall be in one or more of the areas given in 6.5.2.1.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the NORB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in NORB activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the NORB Chairman to provide expert advice to the NORB.

MEETING FREQUENCY

6.5.2.5 The NORB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 A quorum shall consist of the Chairman or Vice Chairman plus enough voting members to constitute a majority of the NORB. No more than a minority of the quorum shall have line responsibility for operation of the facility. For the purpose of a quorum those considered to have line responsibility will include the General Manager - Nuclear Generation, Plant Manager and personnel reporting to the Plant Manager.

REVIEW

6.5.2.7 The NORB shall review:

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

b. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance or abnormal degradation of systems designed to contain radioactive material.

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

e. Written reports concerning events requiring 24 hour notification to the Commission.

f. Recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

g. Reports and meetings minutes of the PORC.

h. Proposed changes to Technical Specifications or this Operating License.

i. The safety evaluations for proposed 1) procedures 2) changes to procedures, equipment or systems and 3) test or experiments completed under the provision of Section 50.59 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

AUDITS

6.5.2.8 The following audits shall be conducted under the direction of APCo's Manager - Safety Audit and Engineering Review.

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire facility staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The Facility Emergency Plan at least once per 24 months.

f. The Facility Security Plan at least once per 24 months.

g. Any other area of facility operation considered appropriate by the NORB or the Senior Vice President.

h. The Facility Fire Protection Program and implementing procedures at least once per 24 months.

i. An independent fire protection and loss prevention program inspection and audit of the unit at least once per 12 months utilizing either qualified offsite licensee personnel or an outside fire protection firm.

j. An inspection and audit of the unit fire protection and loss prevention program by a qualified outside fire consultant at least once per 36 months.

k. The radiological effluent and environmental monitoring programs and the results thereof at least once per 12 months.

l. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

m. The PROCESS CONTROL PROGGRAM and implementing procedures for solidification of radioactive wastes at least once per 24 months.

n. A summary report of each scheduled NORB meeting shall be made by the Manager - Safety Audit and Engineering Review.

AUTHORITY

6.5.2.9 The NORB shall report to and advise the Senior Vice President on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of NORB activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NORB meeting shall be prepared, approved and forwarded to the Senior Vice President within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Senior Vice President within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Senior Vice President and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.5.3 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.3.1 Activities which affect nuclear safety shall be conducted as follows:

a. Procedures required by Technical Specification 6.8 and other procedures which affect plant nuclear safety, and changes (other than editorial or typographical changes) thereto, shall be prepared, reviewed and approved. Each such procedure or procedure change shall be reviewed by an individual/group other than the individual/group which prepared the procedure or procedure change, but who may be from the same organization as the individual/group which prepared the procedure or procedure change. Procedures other than Administrative Procedures will be approved by either the Technical Superintendent, the Operations Superintendent, the Maintenance Superintendent, the Performance and Planning Superintendent, the Administrative Superintendent (Document Control and Storeroom),or the Assistant Plant Manager as applicable. The Plant Manager will approve administrative procedures, security implementing procedures and emergency plan implementing procedures. Temporary changes to procedures which clearly do not change the intent of the approved procedures will be approved by two members of the plant staff, at least one of whom holds a

Senior Reactor Operator's License. For changes to procedures which may involve a change in intent of the approved procedures, the person authorized above to approve the procedure shall approve the change.

b. Proposed changes or modifications to plant nuclear safety-related structures, systems and components shall be reviewed as designated by the Plant Manager. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modifications. Proposed modifications to plant nuclear safety-related structures, systems and components shall be approved prior to implementation by the Plant Manager.

c. Proposed tests and experiments which affect plant nuclear safety and are not addressed in the Final Safety Analysis Report shall be prepared, reviewed, and approved. Each such test or experiment shall be reviewed by an individual/group other than the individual/group which prepared the proposed test or experiment. Proposed test and experiments shall be approved before implementation by the Plant Manager.

d. Occurrences reportable pursuant to the Technical Specification 6.0 violations of Technical Specifications shall be investigated and a report prepared which evaluates the occurrence and which provides recommendations to prevent recurrence. Such reports shall be approved by the Plant Manager and forwarded to the General Manager of Nuclear Generation; and to the Chairman of the Nuclear Operations Review Board.

e. Individuals responsible for reviews performed in accordance with 6.5.3.1.a, 6.5.3.1.b, 6.5.3.1.c, and 6.5.3.1.d shall be members of the plant supervisory staff previously designated by the Plant Manager. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the review personnel of the appropriate discipline.

f. Each review will include a determination of whether or not an unreviewed safety question is involved. Pursuant to 10 CFR 50.59 NRC approval of items involving unreviewed safety question will be obtained prior to Planc Manager approval for implementation.

RECORDS

6.5.3.2 Records of the above activities shall be provided to the Plant Manager, PORC and/or NORB as necessary for required reviews.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the PORC and submitted to the NORB and the General Manager of Nuclear Generation.

6.7 SAFETY LIMIT VIOLATION

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

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The General Manager of Nuclear Generation and the Vice President-Nuclear Generation shall be notified within 24 hours.

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

d. The Safety Limit Violation Report shall be submitted to the Commission, and the General Manager of Nuclear Generation for NORB review within 14 days of the violation.

6.8 PROCEDURES

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

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

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

i. Programs for effluent and environmental monitoring, using the guidance in Regulatory Guide 4.15, February 1979.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, including temporary changes shall be reviewed prior to implementation as set forth in 6.5 above.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

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

STARTUP REPORT

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

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

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

ANNUAL REPORT[1/]

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

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated manrem exposure according to work and job functions,[2/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

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

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

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT[3/]

6.9.1.6 Routine radiological environmental operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

6.9.1.7 The annual radiological environmental operating reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

The annual radiological environmental operating reports shall include summarized and tabulated results in the format of Regulatory Guide 4.8, December 1975 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT[4/]

6.9.1.8 Routine radioactive effluent release reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 50 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

3/ A single submittal may be made for a multiple unit st[illegible]on.

4/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

6.9.1.9 The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Release of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed, wind direction, and atmospheric stability, and precipitation (if measured) on magnetic tape, or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous 12 consecutive months to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation.

The radioactive effluents release shall include the following information for each type of solid waste shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimated),

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

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

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the PROCESS CONTROL PROGRAM (PCP) made during the reporting period.

MONTHLY OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the Reactor Coolant System PORV's or safety valves, shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days from which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the change was implemented.

REPORTABLE OCCURRENCES

6.9.1.11 The REPORTABLE OCCURRENCES of Specifications 6.9.1.12 and 6.9.1.13 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report data.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.12 The types of events listed shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the Regional Office, or his designate, no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the reactor protection system or other systems subject to Limiting Safety System Settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the Limiting Safety System Setting in the Technical Specifications or failure to complete the required protective function.

b. Operation of the unit or affected systems when any parameter or operation subject to a Limiting Condition for Operation is less conservative than the least conservative aspect of the Limiting Condition for Operation established in the Technical Specifications.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation greater than or equal to 1% delta k/k; a calculated reactivity balance indicating a SHUTDOWN MARGIN less conservative than specified in the Technical Specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% delta k/k; or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the Safety Analysis Report.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the Safety Analysis Report.

g. Conditions arising from natural or man-made events that, as a direct result of the event require unit shutdown, operation of safety systems, or other protective measures required by Technical Specifications.

h. Errors discovered 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 that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the Safety Analysis Report or Technical Specifications bases; or discovery during unit life of conditions not specifically considered in the Safety Analysis Report or Technical Specifications that require remedial action or corrective measures to prevent the existence of development.

j. Offsite releases of radioactive materials in liquid and gaseous effluents which exceed the limits of Specification 3.11.1.1 or 3.11.2.1.

k. Exceeding the limits in Specification 3.11.1.4 or 3.11.2.6 for the storage of radioactive materials in the listed tanks. The written follow-up report shall include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.

THIRTY-DAY WRITTEN REPORTS

6.9.1.13 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within 30 days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fullfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a Limiting Condition for Operation or plant shutdown required by a Limiting Condition for Operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.12.c above designed to contain radioactive material resulting from the fission process.

e. An unplanned offsite release of 1) more than 1 curie of radioactive material in liquid effluents, 2) more than 150 curies of noble gas in gaseous effluents, or 3) more than 0.05 curies of radioiodine in gaseous effluents. The report of an unplanned offsite release of radioactive material shall include the following information:

    1. A description of the event and equipment involved.

    2. Cause(s) for the unplanned release.

    3. Actions taken to prevent recurrence.

    4. Consequences of the unplanned release.

f. Measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level values of Table 3.12-2 when average over any calendar quarter sampling period.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.

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

c. ALL REPORTABLE OCCURRENCES submitted to the Commission.

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

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

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

6.10.2 The following records shall be retained for the duration of the Unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

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

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

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

k. Records of meetings of the PORC and the NORB.

l. Records of secondary water sampling and water quality.

m. Records of analyses required by the radiological environmental monitoring program.

n. Records for Environmental Qualification which are covered under the provisions of paragraph 6.1.6.

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit.* Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

*Health Physics personnel or personnel escorted by Health Physics 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.

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 facility Health Physicist in the Radiation Work Permit.

6.12.2 In addition to the requirements of 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in any one hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift foreman and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved Radiation Work Permit (RWP) which shall specify the dose rate levels in the immediate work area and he maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in any one hour a dose in excess of 1000 mrem** that are located within large areas, such as PWR containment, where no enclosure exists for purpose of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

**Measurement made at 18" from source of radioactivity.

6.13 PROCESS CONTROL PROGRAM (PCP)

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

6.13.2 Licensee initiated changes to the PCP:

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

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information.

   b. A determination that the change did not reduce the overall conformance of the solidified waste program to existing criteria for solid waste; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the PORC.

2. Shall become effective upon review and approval in accordance with Specification 6.5.3.1.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

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

6.14.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Monthly Operating Report within 90 days of the date the change(s) was made effective. This submittal shall contain:

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

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the PORC

2. Shall become effective upon review and approval in accordance with Specification 6.5.3.1.

6.15 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Liquid, Gaseous, and solid)

6.15.1 Licensee initiated major changes to the radioactive waste systems (Liquid, gaseous and solid):

1. Shall be reported to the Commission in the Monthly Operating Report for the period in which the evaluation was implemented. The discussion of each change shall contain:

    a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

    b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

    c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

    d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

    e. An evaluation of the change which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

    f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

    g. An estimate of the exposure to plant operating personnel as a result of the change; and

    h. Documentation of the fact that the change was reviewed and found acceptable by the PORC.

2. Shall become effective upon review and approval in accordance with 6.5.3.1.

6.16 ENVIRONMENTAL QUALIFICATION

6.16.1 By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," December 1979. Copies of these documents are attached to the Order for Modification of License NO. NPF-2 dated October 24, 1980.

6.16.2 By no later than December 1, 1980, complete and auditible records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

Attachment 2

SAFETY EVALUATION FOR
POST TMI-REQUIRED TECHNICAL SPECIFICATONS

Background:

On October 31, 1980, the NRC staff issued NUREG-0737 which incorporated into one document all post TMI-related items approved for implementation by the Commission at this time. Included in NUREG-0737 is a requirement for additional technical specifications for operating reactors. This safety evaluation supports the addition of some of these requirements to the Farley Unit 1 Technical Specifications.

References:

(1) NRC NUREG-0737 dated October 31, 1980.

(2) Technical Specifications 3.3.3.8, 4.3.3.8, 3.4.4, 4.4.4, 3.4.4.a, 4.4.4.a, 3.4.6.3 and 4.4.6.3.

Bases:

NUREG-0737 requires additional technical specifications be incorporated into the Farley Unit 1 Operating License. Technical Specifications are required by NUREG-0737 Items; II.D.3, "Direct Indication of Relief and Safety-Valve Position"; II.E.1.2, Auxiliary Feedwater System Automatic Initiation and Flow Indication"; II.E.3.1, "Emergency Power Supply for Pressurizer Heaters"; II.F.2, "Instrumentation for Detection of Inadequate Core Cooling"; II.G.1, "Emergency Power for Pressurizer Equipment"; and III.D.1.1, "Integrity of Systems Outside Containment Likely to Contain Radioactive Material for Pressurized-Water Reactors." Additional specifications required by Items II.D.3, II.E.1.2, and II.F.2 are provided in the proposed revisions to Specifications 3.3.3.8 and 4.3.3.8. Additional specifications required by Item II.E.3.1 are provided in the proposed revision to Specifications 3.4.4 and 4.4.4. Additional specifications required by Items II.G.1 and III.D.1.1 are provided in proposed Specifications 3.4.4.a, 4.4.4.a and 3.4.6.3, 4.4.6.3, respectively. The proposed revisions to the technical specifications will provide assurance of system operability for those changes identified in NUREG-0737. In addition, the proposed revisions have been tailored after the Unit 2 Technical Specifications which reflect the latest NRC approved version of the standard technical specifications.

Conclusion:

The proposed changes to the Technical Specifications to incorporate the requirements of NUREG-0737 do not involve an unreviewed safety questions as defined by 10CFR50.59.

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The accident monitoring instrumentation channels shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE accident monitoring channels less than the Required Number of channels shown in Table 3.3-11, restore the inoperable channel to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11; restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Reactor Coolant Outlet Temperature-T_{Hot}-Wide Range	2	1
2. Reactor Coolant Inlet Temperature-T_{Cold}-Wide Range	2	1
3. Reactor Coolant Pressure-Wide Range	2	1
4. Steam Generator Water Level-Wide Range or Narrow Range	2/steam generator	1/steam generator
5. Refueling Water Storage Tank Water Level	2	1
6. Containment Pressure	2	1
7. Pressurizer Water Level	2	1
8. Steam Line Pressure	2/steam generator	1/steam generator
9. Auxiliary Feedwater Flow Rate	2	1
10. Reactor Coolant System Subcooling Margin Monitor	2	1
*11. PORV Position Indicator	1/valve	1/valve
**12. PORV Block Valve Position Indicator	1/valve	1/valve
13. Safety Valve Position Indicator	1/valve	1/valve

*Not applicable if the associated block valve is in the closed position.
**Not applicable if the block valve is verified in the closed position and power removed.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Coolant Outlet Temperature-T_{Hot}-Wide Range	M	R
2. Reactor Coolant Temperature-T_{Cold}-Wide Range	M	R
3. Reactor Coolant Pressure-Wide Range	M	R
4. Steam Generator Water Level- Wide Range or Narrow Range	M	R
5. Refueling Water Storage Tank Water Level	M	R
6. Containment Pressure	M	R
7. Pressurizer Water Level	M	R
8. Steam Line Pressure	M	R
9. Auxiliary Feedwater Flow Rate	M	R
10. Reactor Coolant System Subcooling Margin Monitor	M	R
*11. PORV Position Indicator	M	R
**12. PORV Block Valve Position Indicator	M	R
13. Safety Valve Position Indicator	M	R

*Not applicable if the associated block valve is in the closed position.
**Not applicable if the block valve is verified in the closed position and power removed.

3/4.4.4.a RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4.a Two power relief valves (PORV's) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.a.1 Each PORV shall be demonstrated OPERABLE at least once per 18 months by performance of a CHANNEL CALIBRATION and operating the valve through one cycle of full travel.

4.4.4.a.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel.

POST ACCIDENT RECIRCULATION LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.3 Leakage outside containment from systems which could carry radioactive fluids following an accident shall be limited to:

a. 5 GPM from the recirculation portion of the high pressure safety injection system,

b. 5 GPM from the containment spray system,

c. 5 GPM from the reactor coolant system letdown and makeup system.*

d. 5 GPM from the residual heat removal system,

e. 5 GPM from the recirculation portion of the low pressure safety injection system.

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

ACTION:

a. With any leakage greater than any one of the above limits, reduce the leakage to within its limit within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.3 Leakages shall be demonstrated to be within each of the above limits at least once per 18 months.

*Excluding mixed bed and cation bed demineralizer flow paths.

3/4.3.3.7 HIGH ENERGY LINE BREAK ISOLATION SENSORS

The high energy line break isolation sensors are designed to mitigate the consequences of the discharge of steam and/or water to the affected room and other lines and systems contained therein. In addition, the sensors will initiate signals that will alert the operator to bring the plant to a shutdown condition.

3/4.3.3.8 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available for selected plant parameters to monitor and assess these variables following an accident.

3/4.3.3.9 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.4.2 and 3/4.4.3 SAFETY VALVES (Continued)

than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

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

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12 hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE assures that the plant will be able to establish natural circulation.

3/4.4.4.a RELIEF VALVES (PORV'S)

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operaton of the PORV's minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. The PORV's are also available to remove non-condensable gases from the reactor coolant system by means of remote manual operations from the control room.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for Inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, 3/4.4.5 STEAM GENERATORS (Continued)

manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the

3/4.4.4 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with at least 125 kw of pressurizer heaters and a water volume of less than or equal to 868 (63.5% indicated) cubic feet.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.4.2 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal to the emergency power supply and energizing the heaters.

*Limit not applicable during either a THERMAL POWER ramp change in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step change in excess of 10% of RATED THERMAL POWER.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.6.3 POST ACCIDENT RECIRCULATION LEAKAGE

The leakage limitations required by this specification provide allowance for a limited amount of leakage outside containment from systems which could carry radioactive fluids following a major accident. These systems are the high pressure safety injection, containment spray, chemical volume and control, residual heat removal, and low pressure safety injection. The leakage limits apply only to the recirculation portion of these systems outside containment, potential leakage points of these systems without leak-off collection system connections, and leakage sources in the active portion of these systems up to and including the first normally closed valve on lines connecting to the active portion of the system. A leakage limit of 5 GPM from each of these systems ensures that the dosage contribution is well within Part 100 limits.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

Tl surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific parameters of the Farley-Unit site, such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

FARLEY NUCLEAR PLANT
NUCLEAR SAFETY EVALUATION CHECK LIST

(1) UNIT 1
(2) CHECK LIST APPLICABLE TO: TECH. SPEC. REV. Revision — TCN —
(3) SAFETY EVALUATION - PART A SECT. 3/4.0

The procedure, procedure change or modifcation to which this evaluation is applicable represents:

(3.1) Yes____ No ✓ A change to the plant as described in the FSAR?
(3.2) Yes____ No ✓ A change to procedures as described in the FSAR?
(3.3) Yes____ No ✓ A test or experiment not described in the FSAR?
(3.4) Yes ✓ No____ A change to the Technical Specifications?

If the answer to any of the above questions is "Yes," complete Item (4) and attach a 10CFR50.59 evaluation. If the answer to all of the above is "No," omit Item (4) and Item (9).

(4) SAFETY EVALUATION - PART B

(4.1) Yes____ No ✓ Will the probability of an accident previously evaluated in the FSAR be increased?
(4.2) Yes____ No ✓ Will the consequences of an accident previously evaluated in the FSAR be increased?
(4.3) Yes____ No ✓ May the possibility of an accident which is different than any already evaluated in the FSAR be created?
(4.4) Yes____ No ✓ Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
(4.5) Yes____ No ✓ Will the consequences of a malfunction of equipment important to safety different than any already evaluated in the FSAR be increased?
(4.6) Yes____ No ✓ May the possiblity of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
(4.7) Yes____ No ✓ Will the margin of safety as defined in the basis to any Technical Specification be reduced?

If the answer to any of the above questions is "Yes," an unreviewed safety question is involved. Explain the basis for each answer provided in Section 4.

(5) REMARKS: (Attach additional pages if necessary)
SEE ATTACHED

(6) PREPARED BY: H. M. McClellan DATE 12-3-80
(7) REVIEWED BY: R. D Rogers DATE 12/3/80
(8) PORC REVIEW: W. J. Hairston III DATE 12-4-80
(9) NORB REVIEW: [illegible] DATE 12-11-80

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Attachment 3

Unit 1 0737 Required Tech Spec (1/01/81)

1. STA - I.A.1.1, Unit 1, Revised Figure 6.2-2, Section 6.2.4, Table 6.2-1 (T.S. due 1/01/81)

2. Shift Manning - I.A.1.3, Unit 1, Revised Section 6.2.2 (T.S. due 1/01/81)

3. RCS Vents (Head Vent) - II.B.1, (T.S. due 7/01/81)

4. Post Accident Sampling - II.B.3, (T.S. due 1/01/82)

5. Valve Position Indication - II.D.3, Unit 1 T.S. Revised Table 3.3-11, Items 11 and 12 (T.S. due 1/01/81)

6. AFW Evaluation - II.E.1.1, T.S. not due until evaluation completed.

7. AFW Initiation - II.E.1.2(1) - (T.S. due 1/01/81) Current Section 3.3.2.1 and Table 3.3-3 and 3.3-4 satisfy these requirements.

8. AFW Flow Indication - II.E.1.2(2) (T.S. due 1/01/81) Unit 1 Revised Section 3.3.8 and Table 3.3-11 satisfy these requirements.

9. Emergency Power Supply for Przr Heaters - II.E.3.1 (T.S. due 1/1/81) Revised Sec. 3.4.4.

10. Containment Isolation Dependability - II.E.4.2 (T.S. due 1/1/81) Containment Pressure setpoint is discussed in Attachment 4. The Unit 1 Technical Specification will be modified relating to containment purge when this issue is resolved for Unit 2.

11. a. Additional Accident Monitoring Instrumentation II.F.1.1 Noble Gas (T.S. due 1/01/82)

    b. II.F.1.2 Iodine/Particulate (T.S. due 1/01/82)

    c. II.F.1.3 Containment High Range Radiation Monitor (T.S. due 1/01/82)

    d. II.F.1.4 Containment Pressure Monitor (T.S. due 1/01/82)

    e. II.F.1.5 Containment Level (T.S. due 1/01/82)

    f. II.F.1.6 Containment Hydrogen Monitor (T.S. due 1/01/82)

12. II.F.2 Inadequate Core Cooling (T.S. 1/01/81)
    T.S. required 1/01/81 per NUREG 737. Proposed Unit 1 T.S. Section 3.3.3.8 includes subcooling monitor. T.S. for the reactor vessel water level system is not included in the draft Unit 1 T.S. but will be required by closest refueling after 1/01/82. T.S. required for incore thermouples by 1/01/82.

13. II.G.1 Emergency Power for Pressurizer Equipment. (T.S. due 1/01/81). Revised Unit 1 T.S. Section 3.4.4 includes requirements for emergency power to pressurizer heaters. Technical Specification requirements for PORV's, Block Valves, and Pressurizer Level are included in current Sections 3.8.2.1, 3.8.2.2, 3.8.2.3 and 3.8.2.4 (see note below).

14. II.K.2.13 Thermal Mechanical Report - (T.S. as required by 1/01/82).

15. II.K.3.1 Auto PORV Isolation - (T.S. required 7-01-81 if required by II.K.3.2).

16. II.K.3.3 SV failures and challenges (T.S. due 1/01/81) covered in revised Unit 1 Section 6.9.1.10.

17. II.K.3.5 Auto trip of RCP's (T.S. due if necessary by 3/01/82).

18. II.K.3.10 and 12 Anticipatory 50% trip (T.S. due when mod complete). Current Unit 1 T.S. Section 3.3.1.1 includes this trip.

19. II.K.3.17 ECCS Outages (T.S. due based on our results).

20. III.A.2 Emergency Preparedness (T.S. TBD).

21. III.D.1.1. Primary Coolant Outside Containment (T.S. due 1/01/81). Unit 1 Revised Section 3.4.6.3.

22. III.D.3.4 C.R. Habitability (T.S. due on chlorine detection and CR Emergency Filtration System by 1/01/81). These are now included in current Section 3.3.3.6 and Section 3.7.7.1 respectively.

Note: Each power operated relief valve (PORV) is equipped with two solenoid valves. These solenoid valves are powered from the class 1E 125V D.C. power system. Technical specifications are provided for the class 1E D.C. power system in sections 3.8.2.3 and 3.8.2.4. The pressurizer block valves received their power from motor control centers via one of the emergency load centers. Technical specifications are provided for the emergency load centers in sections 3.8.2.1 and 3.8.2.2. The pressurizer level transmitters receive power from the 120V A.C. vital buses. Technical specifications are provided for the 120V A.C. vital buses in sections 3.8.2.1 and 3.8.2.2.

ATTACHMENT 4

The containment pressure-high (CP-H) setpoint is selected to limit the maximum pressure inside containment following a design basis accident, to provide early isolation in the event of an accident and must be set high enough to avoid spurious isolation. Technical Specification 3.6.1.4 requires that the containment internal pressure be limited to a maximum of 3.0 psig under normal operating conditions. The 3.0 psig allowance is required to assure that the containment peak pressure does not exceed the design condition pressure of 54 psig during the LOCA condition.

All safety analyses affected by the CP-H setpoint have been reviewed. Based on this review, it has been determined that a decrease in the CP-H setpoint would have no adverse effect on any safety analyses. Thus, the analyses presented in the Farley Nuclear Plant FSAR remains conservative.

The minimum obtainable CP-H setpoint would be the sum of the maximum allowable normal operation containment pressure (3.0 psig) plus the instrument error of the containment pressure transmitter. The containment pressure transmitter error is equal to 2.5 percent of the instrument span based on instrument and rack drift, calibration accuracy and temperature effects (-5 to +65 psig) or 1.8 psig. Thus, it is possible to reduce the CP-H setpoint to 4.8 psig (5.4 psig is current setpoint).

If the setpoint were reduced to 4.8 psig, a setpoint reduction of only 0.6 psig, there would be no margin for minor containment pressure increases associated with plant operation and the probability of spurious safety injections would be significantly increased.

Based on a review of this information, Alabama Power Company has concluded that reducing the CP-H setpoint with a corresponding higher probability of spurious safety injections is not advisable.