QUESTION 400.1

Provide the following estimated costs:

(1) Nuclear plant production costs

(2) Transmission, distribution and general plant costs

(3) Nuclear fuel inventory cost for the first core

RESPONSE:

(1) Nuclear Production Plant costs are estimated to be $1,938 million. This cost, based upon Applicants current estimate includes land, structures, reactor, turbine, miscellaneous electrical plant and other costs such as engineering, construction management, escalation, and allowance for funds. It is based upon Commercial Operation dates of February 1, 1981 and May 1, 1982 for Units 1 & 2 respectively.

(2) The capital cost of the bulk power transmission system including switchyards associated with the plant is $145 million.

(3) The nuclear fuel inventory costs for the first covers are $84 million for Unit 1 and $88 million for Unit 2. These costs include AFDC.

QUESTION 410.1

1.a. Indicate the estimated annual cost by year to operate each unit of the subject facility for the first seven full years of each unit's commercial operation. The types of costs included in the estimates should be indicated and include (but not necessarily be limited to) operation and maintenance expense (with fuel costs shown separately), depreciation, taxes and a reasonable return on investment. (Enclosed is a form which should be used for each year of the seven year period.) Indicate the projected plant capacity factor (in percent) for each unit during each of the seven years. Provide separate estimates using 50 percent and 60 percent plant capacity factors.

b. Indicate the unit price per kWh experienced by each applicant on system-wide sales of electric power to all customers for the most recent 12-month period.

RESPONSE:

The information requested was provided by letter dated April 22, 1981 (PLA-744, Curtis to Youngblood).

ATTACHMENT FOR ITEM NO. 1.a.

ESTIMATED ANNUAL COST OF OPERATING NUCLEAR GENERATING UNIT:

____________________ FOR THE CALENDAR YEAR 19__
(thousands of dollars)

Operation and maintenance expenses	
Nuclear power generation	
Nuclear fuel expense (plant factor_____%) .	$__________
Other operating expenses	__________
Maintenance expenses	__________
Total nuclear power generation	__________
Transmission expenses	__________
Administrative and general expenses	
Property and liability insurance	__________
Other A.&G. expenses	__________
Total A.&G. expenses	__________
TOTAL O&M EXPENSES	__________
Depreciation expense	__________
Taxes other than income taxes	
Property taxes	__________
Other	__________
Total taxes other than income taxes .	__________
Income taxes - Federal	__________
Income taxes - Other	__________
Deferred income taxes - Net	__________
Investment tax credit adjustments - Net . .	__________
Return (rate of return: _____%)	__________
TOTAL ANNUAL COST OF OPERATION	$__________

QUESTION 410.2

Indicate the estimated costs of permanently shutting down each unit of the facility, stating what is included in such costs, the assumptions made in estimating the costs, the type of shutdown contemplated, and the intended source of funds to cover these costs.

RESPONSE:

This information was provided by letter dated April 22, 1981 (PLA-744, Curtis to Youngblood).

QUESTION 410.3

Provide an estimate of the annual cost to maintain each unit of the shutdown facility in a safe condition. Indicate what is included in the estimate, assumptions made in estimating costs, and the intended source of funds to cover these costs.

RESPONSE:

This information was provided by letter dated April 22, 1981 (PLA-744, Curtis to Youngblood).

QUESTION 410.4

If the facility is jointly-owned provide copies of the joint participation agreement setting forth the procedures by which the applicants will share operating expenses and decommissioning costs.

RESPONSE:

This information was provided by letter dated April 22, 1981 (PLA-744, Curtis to Youngblood).

QUESTION 410.5

Provide copies of the prospectus for the most recent security issue and copies of the most recent SEC Form 10-K and 10-Q. Provide copies of the preliminary prospectus for any pending security issue. Submit copies of the Annual Report to Stockholders each year as required by 10 CFR 50.71(b).

RESPONSE:

This information was provided by letter dated April 22, 1981 (PLA-744, Curtis to Youngblood).

QUESTION 410.6

Describe aspects of its regulatory environment including, but not necessarily limited to, the following: prescribed treatment of allowance for funds used during construction; rate base (original cost, fair value, other); accounting for deferred income taxes and investment tax credits; fuel adjustment clauses in effect or proposed; historical, partially projected, or fully projected test year.

RESPONSE:

This information was provided by letter dated April 22, 1981 (PLA-744, Curtis to Youngblood).

QUESTION 410.7

Describe the nature and amount of its most recent rate relief action(s). In addition, indicate the nature and amount of any pending rate relief action(s). Use the attached form to provide this information. Provide copies of the submitted, financially related testimony and exhibits of the staff and company in the most recent rate relief action or pending action. Furnish copies of the hearing examiner's report and recommendation, and final opinion last issued with respect to each participant, including all financially related exhibits referred to therein.

RESPONSE:

This information was provided by letter dated April 22, 1981 (PLA-744, Curtis to Youngblood).

ATTACHMENT FOR ITEM NO. 7

RATE DEVELOPMENTS

	Electric	Gas	Steam
Granted			
Test year utilized			
Annual amount of revenue increase requested- test year basis (000's)			
Date petition filed			
Annual amount of revenue increase allowed- test year basis (000's)			
Percent increase in revenues allowed			
Date of final order			
Effective date			
Rate base finding (000's)			
Construction work in progress included in Rate base (000's)			
Rate of return on rate base authorized			
Rate of return on common equity authorized			
Revenue Effect (000's)			
Amount received in year granted			
Amount received in subsequent year (If not available, annualize amounts received in year granted)			
Pending Requests			
Test year utilized			
Amount (00's)			
Percent increase			
Date petition filed			
Date by which decision must be issued			
Rate of return on rate base requested			
Rate of return on common equity requested			
Amount of rate base requested			
Amount of construction work in progress requested for inclusion in rate base			

QUESTION 410.8

Complete the enclosed form entitled, "Financial Statistics," for the most recent twelve-month period and for the previous three calendar years.

RESPONSE:

This information was provided by letter dated April 22, 1981 (PLA-744, Curtis to Youngblood).

ATTACHMENT FOR ITEM NO. 8
FINANCIAL STATICS

12 months ended

(dollars in millions)

Earnings available to common equity
Average common equity
Rate of return on average common equity

Times total interest earned before FIT:
Gross income (both including and excluding AFDC) + current and deferred FIT + total interest charges + amortization of debt discount and expense

Times long-term interest earned before FIT:
Gross income (both including and excluding AFDC) + current and deferred FIT long term interest charges + amortization of debt discount and expense

Bond ratings (end of period)
Standard and Poor's
Moody's

Times interest and preferred dividends earned after FIT:
Gross income (both including and excluding AFDC) + total interest charges + amortization of debt discount and expense + preferred dividends.

AFUDC
Net income after preferred dividends
%

Market price of common
Book value of common
Market-book ratio (end of period)*

Earnings avail. for common less AFDC + depreciation and amortization, deferred taxes, and invest. tax credit adjust.- deferred.

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* If subsidiary company, use parent's data.

	12 months' ended
	(dollars in millions)
Common dividends	
Ratio	
Short-term debt	
Bank loans	
Commercial paper	
Capitalization (Amount & Percent)	
Long-term debt	
Preferred stock	
Common equity	

QUESTION 410.9

Is each participant's percentage ownership share in the facility equal to its percentage entitlement in the electrical capacity and output of the plant? If not, explain the difference(s) and any resultant effect on any participant's obligation to provide its share of operating costs.

RESPONSE:

This information was provided by letter dated April 22, 1981 (PLA-744, Curtis to Youngblood).

QUESTION 410.10

Describe the rate-setting authority and rate covenants of the cooperatives and how that authority will be used to ensure the satisfaction of financial obligations in relation to operation and eventual shutdown of the facility.

RESPONSE:

This information was provided by letter dated April 22, 1981 (PLA-744, Curtis to Youngblood).

QUESTION 410.11

Describe the nature and amount of the cooperative's most recent rate relief action(s) and its anticipated effect on net margins. In addition, indicate the nature and amount of any pending rate relief action(s).

RESPONSE:

This information was provided by letter dated April 22, 1981 (PLA-744, Curtis to Youngblood).

QUESTION 410.12

If membership cooperatives are involved, explain the contractual arrangements between the cooperative and its members that will provide funds for operation and eventual shutdown of the facility. Provide representative copies of such contracts.

RESPONSE:

This information was provided by letter dated April 22, 1981 (PLA-744, Curtis to Youngblood).

QUESTION 410.13

Provide copies of the latest annual and interim financial statements. Also provide copies of similar statements for the corresponding periods ended in the previous year. Continue to submit copies of the annual financial statements each year as required by 10 CFR 50.71(b).

RESPONSE:

This information was provided by letter dated April 22, 1981 (PLA-744, Curtis to Youngblood).

QUESTION 421.1

The Quality Assurance Branch (QAB) has reviewed Pennsylvania Power and Light Company's (PP&L) Fire Protection Report (dated January 18, 1978) for Susquehanna Steam Electric Station (SSES) Units 1 and 2. This report was submitted in response to Mr. Boyd's letter of September 30, 1976. Based on our review of this information, we find that adequate information has not been submitted by PP&L to permit completion of the QAB review of the fire protection program.

Item 26 (pg. 3-48) of your submittal does not indicate what the management control of the QA organization consists of. The description for QA management should consist of (1) formulating and/or verifying that the fire protection QA program incorporates suitable requirements and is acceptable to the management responsible for fire protection through review, surveillance, and audits. Performance of other QA program functions for meeting the fire protection program requirements may be performed by personnel outside of the QA organization. The QA program for fire protection should be part of the overall plant QA program. These QA criteria apply to those items within the scope of the fire protection program, such as fire protection systems, emergency lighting, communication and breathing apparatus, as well as the fire protection requirements of applicable safety-related equipment.

RESPONSE*:

Subsections 17.2.1.1.2 and 17.2.2 have been revised to include this information.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.2

We find that your response to Mr. Boyd's letter of September 30, 1976, does not describe sufficient detail to address the ten specific quality assurance criteria in Branch Technical Position ASB 9.5-1. In order for the QAB to fully evaluate your approach for meeting these criteria, additional detailed description is necessary. Examples of the detail we would expect PP&L to consider are provided in Attachment 6 of Mr. D. B. Vassallo's letter of August 29, 1977. If, however, you choose not to provide this detail, you may apply the same controls to each criterion that are commensurate with the controls described in your QA program for operations. These controls would apply to the remaining construction activities and for the operations phase of Unit Nos. 1 and 2. If you select this method, a statement to this effect would be adequate for our review of the fire protection QA program.

RESPONSE*:

Subsection 17.2.2 and Table 17.2.1 have been revised to provide this information. It is emphasized that this commitment does not take effect until the fire protection systems are turned over from the responsible contractor to PP&L control.

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* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.3

Provide a description of how the QA Supervisor (located onsite) communicates with the offsite QA organizations relative to matters concerning QA/QC, and describe those conditions for determining when these actions should take place. The offsite/onsite interface should also be shown on the applicable organizational charts in the QA program description.

RESPONSE*:

Subsection 17.2.1.1.1.4.1.1.1 and Figure 17.2-4 have been revised to include this information.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.4

Identify on organizational charts the reporting relationship of the Nuclear Review Board.

RESPONSE:

Figures 17.2-2 and 17.2-3 have been revised to include this information.

QUESTION 421.5

FSAR Figure 17.2-2 has an organizational block listed as "others." Clarify what "others" are and describe their QA/QC functions, if any.

RESPONSE*:

FSAR Figures 17.2-2 and 17.2-3 have been revised.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.6

Describe in more detail the specific responsibilities of the Nuclear Quality Assurance Staff in executing the SSES QA program.

RESPONSE*:

Subsection 17.2.1.1.2 has been revised to include this information.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.7

Describe in more detail those "quality activities" (ref. FSAR page 17.2-6) performed by the Manager, Power Production.

RESPONSE*:

Subsection 17.2.1.1.1.4 has been revised to include this information.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.8

Describe provisions which assure that the Vice-President, Systems Power and Engineering, maintains a continuing involvement in QA matters and how he communicates through intermediate levels of management. (e.g., review and concurrence of SSES operations, administrative control, and operational QA program.)

RESPONSE*:

Subsection 17.2.1.1 has been revised to include this information.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.9

Clearly identify the individual/position responsible for having overall responsibility and authority for the SSES operational QA program.

RESPONSE*:

Subsection 17.2.1.1 has been revised to include this information.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.10

Describe the amount of nuclear quality assurance experience required for the position of Quality Assurance Manager. The amount of experience should be at least equal to the one year experience listed in paragraph 4.4.5 of ANSI/ANS-3.1-1978, "Selection and Training of Nuclear Power Plant Personnel."

RESPONSE*:

Subsection 17.2.1.1.2 has been revised to include this information.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.11

Describe the qualifications established for the QA Supervisor regarding quality assurance and quality control related experience.

RESPONSE*:

Subsection 17.2.1.1.1.4.1.1.1 has been revised to include this information.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.12

Describe the measures which assure that personnel (including those outside the QA/QC organization) performing QA/QC functions have sufficient authority and organizational freedom to:

a) Identify quality problems.
b) Initiate, recommend, or provide solutions through designated channels, and
c) Verify implementation of solutions.

This description should also include measures to assure that verification of conformance to established requirements is accomplished by individuals or groups who do not have direct responsibility for performing the work being verified.

RESPONSE*:

10CFR50 Appendix B, Criterion I, states in part, "The persons and organizations performing quality assurance functions shall have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions."

The overall quality assurance program responsibilities lie with the NQA Section. Its organizational reporting path is shown in Figures 17.2-2 and 17.2-3. By reporting directly to the Executive Vice President - Operations and in accordance with statements made in Subsection 17.2.1.1.2, the Manager - NQA has the required authority and freedom.

The reporting path of the Station Quality Supervisor is shown in Figure 17.2-4. Since the Quality Supervisor reports to the Superintendent of Plant, he and his staff are independent of the individuals who are directly responsible for performing the work being verified. In addition, the Quality Supervisor has direct recourse to the Manager - NQA in situations where he and the Superintendent of Plant disagree over quality requirements. (Refer to Subsection 17.2.1.1.1.4.1.1.1.)

Other organizations within PP&L do not perform quality assurance functions as used in the context of 10CFR50 Appendix B, Criterion I.

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* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.13

Clarify whether the stop work authority vested in the Manager - NQA is delineated in writing.

RESPONSE*:

In addition to the description in Subsection 17.2.1.1.2, the authority of the Manager - NQA to stop work is contained in Operational Policy Statement, OPS-5, Deficiency Control (refer to Table 17.2-2).

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.14

Describe provisions which assure that management (i.e., above or outside the QA organization) annually assesses the scope, status, implementation, and effectiveness of the QA program to assure that the program is functioning adequately and complies with 10 CFR Part 50, Appendix B criteria, and that the results of this assessment are documented.

RESPONSE*:

See Subsection 17.2.1.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.15

Table 17.2-1 of the FSAR addresses those Regulatory Guides and ANSI standards applicable to the operational QA program and the degree of compliance thereto. Since the docketing of your application (July 31, 1978), certain of these Regulatory Guides (RG) and ANSI standards have been upgraded and differ from the dates stated in Table 17.2-1. Therefore, update your application, and provide a specific commitment to comply with the regulatory positions of each of the following Regulatory Guides and ANSI standards: (RG 1.28, Rev. 1; RG 1.33, Rev. 2; RG 1.38, Rev. 2; RG 1.39, Rev. 2; RG 1.116, Rev. 0-R; RG 1.123, Rev. 1; and ANSI N45.2.12, Draft 3, Rev. 4, 2/22/74 or ANSI N45.2.12, Draft 4, Rev. 2, 1/1/76, as supplemented by regulatory position 4 of Regulatory Guide 1.33, Rev. 2 (2/78). Any exceptions and/or alternatives to the above Regulatory Guides/ANSI standards should be described in sufficient supporting detail to allow for NRC evaluation and acceptance.

RESPONSE*:

Refer to Table 17.2-1. The subject table has been modified to reference each of the indicated Regulatory Guides. However, the version of ANSI N45.2.12 remains as Draft 4, Revision 3, November 29, 1976. PP&L feels that this standard is more desirable since it is more current than the drafts indicated in the subject question. Thus, it reflects more recent industry practice.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.16

It is not clear as to your interpretation of the term "Commitment of the extent required by ANSI N18.7-1976" as used in FSAR Table 17.2.1. Please provide a more detailed explanation of what "Commitment to the extent required by ANSI N18.7-1976" means to PP&L and how it is to be used to assure consistent interpretation within PP&L.

RESPONSE*:

"Commitment to the extent required by ANSI N18.7-1976" is based upon the guidance presented in ANSI 18.7-1976 as far as the application of certain standards that are identified in Table 17.2-1. ANSI N18.7-1976 requires that these standards be applied to"...those activities occurring during the operational phase that are comparable in nature and extent to related activities occurring during construction."

To assure consistent interpretation of this commitment within PP&L, the Manager - NQA is responsible for reviewing extraordinary activities (such as major modifications) and determining when the above "nature and extent" criteria have been met. In those cases where he determines that the "nature and extent" criteria have been met, he shall direct, with the concurrence of the Executive VP - Operations, that the affected OQA Program documents be augmented to include the appropriate additional ASNI standard requirements.

As permitted by ANSI N18.7-1976, the standards indicated on Table 17.2-1 will be used as guidance in the preparation of program documents and procedures when the "nature and extent" criteria above are not met.

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* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.17

Describe those provisions which assure that the docketed QA program description, particularly the commitment to Regulatory Guides and ANSI standards, will be properly carried out and with the use of QA procedures.

RESPONSE*:

Provisions which assure that the docketed QA program description, particularly the commitment to Regulatory Guides and ANSI Standards, will be properly carried out in accordance with instructions, procedures or drawings are as follows:

- By his review of Functional Unit Procedures, the Manager - NQA assures that each functional unit within PP&L recognizes the applicable OQA Program commitments and incorporates these commitments into their procedures.
- All functional units within PP&L are subject to formal periodic audits performed by NQA. These audits verify that the functional units are complying with and properly implementing their procedures.
- The Nuclear Review Board periodically assesses the scope, status, implementation and effectiveness of the OQA Program.

Refer to Subsections 17.2.1, 17.2.1.1.2 and 17.2.18.

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* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.18

Provide a summary description on how responsibilities and control of quality-related activities are transferred between PP&L and principal contractors during the phaseout of design and construction and during preoperational testing and plant turnover.

RESPONSE*:

The responsibilities and control of quality-related activities are assigned to the organization retaining jurisdiction over the material, equipment, structure or system in question. This has been defined as:

1. Prior to turnover - the responsible contractor

2. Following turnover - PP&L

3. Items returned to a contractor for repair, rework, modification after having been turned over to PP&L - the responsible contractor.

The requirements for the transfer of material, equipment, structures or systems have been defined in the PP&L Quality Assurance Manual which provides procedures that are applicable to the construction and preoperational testing phases. The Startup Administrative Manual provides the specific details for implementing the responsibilities for the interface and control of quality-related activities for the preoperational testing and plant turnover phases.

In addition, the NQA Section performs audits to determine that the programmatic and procedural requirements are being fulfilled.

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* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.19

Describe what measures to assure that appropriate 10 CFR Part 50 Appendix B requirements will be applied to the preoperational test program.

RESPONSE*:

The preoperational testing program is being conducted in accordance with the requirements of the PP&L Quality Assurance Manual which is applicable to both plant construction and preoperational testing. The provisions in this manual have been subjected to several NRC I&E Region I inspections and are assessed annually through an independent audit which is authorized by the PP&L QA Council. In all cases, the manual has been found to satisfactorily meet the requirements of 10CFR50 Appendix B.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.20

Describe provisions which assure that the NRC will be notified of changes to the accepted SAR QA program description prior to implementation and of changed to organizational elements within 30 days after announcement. (Note - minor editorial changes or personnel reassignments of a nonsubstantive nature do not require NRC notification.)

RESPONSE*:

PP&L intends to keep the NRC fully informed in regard to changes to the OQA Program description through annual updates to the FSAR, revisions issued to controlled QA Manuals which NRC staff members may have in their possession, and other appropriate means.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.21

Identify those individuals evaluating the suppliers' capabilities to provide acceptable quality services and products prior to the award of procurement order or contract. (QA and Engineering should participate in the evaluation of those suppliers providing critical components.)

RESPONSE*:

Depending upon the item being procured, supplier evaluation is a joint effort of Power Plant Engineering, Nuclear Fuels and Nuclear Quality Assurance. Responsibility to provide this evaluation is assigned to the respective managers of these functional units in Subsections 17.2.1.1.1.2, 17.2.1.1.1.5 and 17.2.1.1.2.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.22

Clarify whether the purchase of spare or replacement parts of safety-related structures, systems, and components are subject to controls at least equivalent to those used for the original equipment.

RESPONSE*:

As stated in Subsection 17.2.4, "Procurement documents for safety-related spare or replacement parts for structures, systems and components are subject to controls the same as, or equivalent to, those used for the original equipment." The remainder of the procurement process will be at least equivalent to that used for the original equipment because of PP&L's commitment to more current regulatory guidance embodied in ASNI N45.2.13-1976 and ANSI N18.7-1976.

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* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.23

Describe measures which assure that records are identifiable and retrievable.

RESPONSE*:

Per Subsection 17.2.17, the respective managers are responsible for developing procedures which control the origination of documents and provide for the inclusion of those documents in the QA Records System. PP&L is developing a microfilm based record management system with an on-line interactive computerized index that will provide access and retrievability of records in a reasonable time.

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* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.24

Describe provisions to assure that the "offsite" QA organization:

a) Conducts sufficient audits to verify the activities conducted by the "onsite" QA organization.

b) Reviews and concurs in the schedule and scope of audits performed by the onsite QA organization.

RESPONSE*:

(a) The Nuclear Quality Assurance Section is responsible for auditing all safety-related aspects of nuclear plant operations. Subsection 17.2.18 describes the provisions used by the NQA Section in determining the frequency and types of activities that will be audited at the plant site. The schedule for auditing site activities will be based upon past audit results, observed trends, and the amount of success exhibited in implementing corrective action. As a minimum, the frequencies for audits performed by the NQA Section will parallel those specified in ANSI N45.2.12 (Draft 4, Rev. 3) and ANSI N18.7-1976. Furthermore, plant operations will be audited against the Susquehanna Technical Specifications to assure compliance with licensing commitments.

(b) The Station Quality Group reporting to the Superintendent of Plant has no responsibility for performing audits.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.25

Paragraph 17.2.1.1.1.4.1.1.1 on page 17.2-6 states the Quality Supervisor is assisted by Quality Specialists and engineers without identifying their reporting relationship on the appropriate organizational charts in Section 17.2 of the FSAR. The above paragraph implies there is more than one individual performing in these positions which appears to contradict the numbers specified in Figure 13.1-7. Please correct this discrepancy.

RESPONSE*:

Figure 17.2-4, Susquehanna Plant Staff Organization, only details the Plant Staff to the section head level with the inference that each of the supervisors have their own support personnel. Refer to Figure 13.1-7 for the reporting relationship of the Quality Supervisor and the quality specialists and engineers.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.26

The response to Q421.14 is not totally acceptable. Provide a description of how management (above or outside the QA organization) regularly assesses the scope, status, adequacy, and compliance of the QA program to 10 CFR 50 Appendix B. These measures should include:

(1) Frequent review of program status through reports, meetings and/or audits.

(2) Performance of an annual preplanned and documented assessment. Corrective action is identified and tracked.

Modify your QA program accordingly.

RESPONSE*:

Refer to Subsection 17.2.1.1 for response.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.27

The response to Q421.15 is not acceptable. The NRC has not endorsed N45.2.12 Draft 4, Revision 3 and therefore you should commit to comply with the provisions of ANSI N45.2.12 Draft 3, Revision 4, 2/22/74 or ANSI N45.2.12 Draft 4, Revision 2, 1/1/76 as supplemented by regulatory position C.4 of Regulatory Guide 1.33, Revision 2 (2/78). Any exceptions and/or alternatives to these controls should be described in sufficient supporting detail to allow for NRC evaluation and acceptance. Modify your QA program accordingly.

RESPONSE*:

Refer to Table 17.2-1 for response. This table has been updated to reference ANSI N45.2.12-1977 which has been endorsed by NRC Regulatory Guide 1.144.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.28

The response to Q421.16 requires clarification concerning the qualification requirements for individuals performing certain QA functions during the operational phase of your plant. Our position is as follows:

(1) The individuals performing inspection, examination and testing functions associated with normal operations of the plant, such as surveillance testing, routine maintenance and certain technical reviews normally assigned to the onsite operation organization shall be qualified to ANSI N18.1-1971.

(2) Personnel whose qualifications are not required to meet those specified in ANSI N18.1 and who are performing inspection, examination and testing activities during the operational phase of the plant shall be qualified to ANSI N45.2.6-1973 except that the QA experience cited for Levels I, II, and III shall be interpreted to mean actual experience in carrying out the types of inspection, examination and testing activity being performed.

This position is consistent with ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," Section 3.4.2. Modify your QA program accordingly to address this staff position.

RESPONSE*:

Refer to Subsection 17.2.2 for response to this question.

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* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.29

The response to Q421.20 is not acceptable. Describe provisions to notify the NRC of changes (1) to the accepted SAR QA program description prior to implementation, and (2) to organizational elements within 30 days after announcement. (Note - editorial changes or personnel reassignments of a non-substantive nature do not require NRC notification.) Modify your QA program accordingly.

RESPONSE*:

For response to this question, refer to Subsection 17.2.1.1.2.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.30

Describe measures to assure that responsible plant personnel are made aware of design changes/modifications which may affect the performance of their duties.

RESPONSE*:

Refer to Subsection 17.2.3 for response to this question.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.31

Clarify whether the Manager-Nuclear Quality Assurance reviews and concurs with changes to the onsite QA program.

RESPONSE*:

Please be advised that PP&L's QA program is an integrated program and is not divided into an "on-site QA program" and an "off-site QA program." Quality-related activities are performed on-site in accordance with Functional Unit Procedures which are responsive to the requirements of the OQA Manual (Refer to Figure 17.2-1). The Manager-NQA reviews the Functional Unit Procedures of all PP&L departments to assure compliance with the OQA Program, per Subsection 17.2.1.1.2. The review process provides a documented comment and resolution cycle that is subject to verification audits. Subsection 17.2.2 describes the contention process for resolving disagreements between NQA and other PP&L departments.

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QUESTION 421.32

Describe provisions which assure that maintenance, modification and inspection procedures are revised by qualified personnel knowledgeable in QA disciplines (normally the QA organization) to determine that the necessary inspection requirements, methods, and acceptance criteria have been identified.

RESPONSE*:

Refer to Subsection 17.2.1.1.1.4.1.1.1 for response. The adequacy and implementation of maintenance, modification and inspection procedures are verified by NQA in conjunction with its normally scheduled audits of such activities.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.33

Describe measures to assure that the QA organization or an individual qualified in quality assurance but other than the person who generated the document, reviews and concurs with the documents and changes thereto with regards to QA-related aspects to assure technical adequacy and inclusion of appropriate quality requirements prior to implementation. Such documents as a minimum include: design, procurement, as-built drawings, QA/QC manuals, SAR, non-conformance reports and instructions and procedures for inspection and testing.

RESPONSE*:

Refer to Subsection 17.2.6 for response to this question.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.34

Describe the organizational responsibilities for the control of purchased material, equipment, and services including the interface responsibilities of the QA organization relative to procurement.

RESPONSE*:

For response to this question, refer to Subsections 17.2.1.1.1.2, 17.2.1.1.1.2.1, 17.2.1.1.1.4.1.1.1, 17.2.1.1.1.5, 17.2.1.1.2 and 17.2.7.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.35

Describe the criteria for determining those processes that are controlled as special processes. As complete a listing as possible of special processes, which are generally those processes where direct inspection is impossible or disadvantageous, should be provided. Some examples are welding, heat treating, NDT, and chemical cleaning.

RESPONSE*:

See Subsection 17.2.9 for response to this question.

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QUESTION 421.36

Describe measures which assure that program procedures provide criteria for determining the accuracy requirements of inspection equipment and criteria for determining when inspections and tests are required.

RESPONSE*:

For response to this question, refer to Subsections 17.2.10 and 17.2.11.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.37

Describe provisions to assure that inspection procedures, instructions, or checklists provide, as required, for the following:

(1) Identification of required procedures, drawings and specifications and revisions.

(2) Specifying necessary measuring and test equipment including accuracy requirements.

RESPONSE*:

Refer to Subsection 17.2.10 for response to this question.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.38

Describe measures which assure that inspection results are documented, evaluated and their acceptability determined by a responsible individual or group.

RESPONSE*:

Refer to Subsection 17.2.10 for response to this question.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.39

Describe the organizational responsibilities for establishing, implementing, and assuring effectiveness of the calibration program.

RESPONSE*:

For response to this question, refer to Subsections 17.2.1.1.1.4.1.1.1 and 13.1.2.2.7.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.40

Describe the provisions established for the storage of chemical, reagents (including control of shelf life), lubricants, and other consumable materials.

RESPONSE*:

Refer to Subsection 17.2.13 for response to this question.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.41

The responses to the 421 series of questions appear to be separated from Section 17.2 of the FSAR. Incorporate or reference all responses to these QA questions in Section 17.2 of the FSAR. It is further requested that a statement be provided whereby the responses to Q421.1 and Q421.2 supersede previous submittals relative to QA for fire protection.

RESPONSE*:

The responses to the 421 series of questions have been incorporated, by revision, to Section 17.2 wherever possible. The few exceptions to this approach involve responses which only provided clarifying information that was felt to be unnecessary for inclusion directly in Section 17.2.

The responses to Questions 421.1 and 421.2 do, indeed, supersede previous submittals relative to QA for fire protection.

* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 421.42

It has come to our attention that some applicants did not intend to conduct confirmatory tests of some distribution systems and transformers supplying power to vital buses as required by Position 3 of Regulatory Guide 1.68, and more specifically by Part 4 of the staff position on degraded grid voltage (applied to all plants in licensing review by the Power Systems Branch since 1976). Part 4 of the degraded grid voltage position states as follows:

"4. The voltage levels at the safety-related buses should be optimized for the full load and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite power source by appropriate adjustment of the voltage tap settings of the intervening transformers. We require that the adequacy of the design in this regard be verified by actual measurement and by correlation of measured values with analysis results. Provide a description of the method for making this verification; before initial reactor power operation, provide the documentation required to establish that this verification has been accomplished."

Your test description in FSAR Chapter 14 does not contain sufficient detail for us to determine if you intend to conduct such a test. It is our position that confirmatory tests of all vital buses must be conducted including all sources of power supplies to the buses. Modify your test description to indicate that this testing will be conducted in accordance with Regulatory Guide 1.68 and the above cited position.

RESPONSE*:

Voltages recorded during the P100.1 Preoperational test (Subsection 14.2.12.1), were reviewed and analyzed against design calculations to assure optimal tap settings have been selected.

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* The response provided to this question was to address a NRC concern at the time of the application for the operating license. This question and response are provided here for historic purposes only and will not be updated. Any revisions to the Nuclear Quality Assurance Organization have been submitted per 10CFR50.54(a) and are described in Section 17.2 of the FSAR.

QUESTION 422.1

In regard to the technical groups providing support for the operation of the Susquehanna Station, provide the following information for the three groups reporting to the Manager-Power Plant Engineering, Figure 13.1-3 (Power Plant Additions, Power Plant Engineering Development, and Project Engineering Manager-Susquehanna) and the three groups reporting to the Manager Nuclear Support, Figure 13.1-4 (Simulator Supervisor-Susquehanna, Senior Nuclear Support Engineers, and Staff Health Physicists):

(1) The number of professional persons assigned or expected to be assigned at the time of plant startup for the group.

(2) The expected proportion of time that they can be assigned to support the operation of Susquehanna if they have assignment for other activities.

RESPONSE:

Power Plant Additions

1) This group will include approximately 100 professional technical personnel at the time of plant start-up (1981).

2) This group's primary responsibilities are the engineering required in support of the company's existing fossil power plants and company buildings and will not have regular assignments in support of Susquehanna SES. They may, on occasion, be called upon for some specific assignment in a specialty area such as an architectural project to expand the office facilities at the plant. Less than 5% of this group's time is anticipated to be spent on support of Susquehanna SES operation. Under emergency conditions engineers from this group could be called upon for support as needed.

Power Plant Engineering Development

1) This group will include approximately 40 professional technical personnel at the time of plant start-up.

2) This group will provide operational support for Susquehanna SES in the area of engineering standards, guidelines and procedures, economic and feasibility studies, and technical consulting. Approximately 20% of this group's time is expected to be directed to

Susquehanna SES support activities. Under emergency conditions, additional support could be provided as needed.

Project Engineering Manager-Susquehanna

This title has been revised to Manager-Nuclear Plant Engineering and the Group name is Nuclear Plant Engineering.

1) This group will include approximately 75 professional technical personnel at the time of plant start-up.

2) This group is dedicated to Susquehanna SES and will spend 100% of its time on operational support.

Simulator Supervisor-Susquehanna

1) This group will include 5 professional technical personnel at the time of plant start-up.

2) This group is dedicated to Susquehanna SES and will spend 100% of its time on operational support.

Senior Nuclear Support Engineers

1) This group will include 6 professional technical personnel at the time of plant start-up.

2) This group is dedicated to Susquehanna SES and will spend 100% of its time on operational support.

Staff Health Physicists

1) This group will include 3 professional technical personnel at the time of plant start-up.

2) This group is dedicated to Susquehanna SES and will spend 100% of its time on operational support.

QUESTION 422.2

Describe the qualification requirements for the positions of Senior Results Engineer and Chemistry Analyst either by reference to ANSI N18.1 or by describing the specific requirements for the positions.

RESPONSE:

The Senior Results Engineer will have, prior to fuel load or appointment to the position (whichever is later), a bachelor's degree in Engineering or the Physical Sciences and three years of power plant experience.

The position of Chemistry Analyst requires a high school diploma or equivalent, successful completion of a high school level chemistry course, and the demonstrated ability to complete Chemistry Analyst training as evidenced by successful completion of a selection examination. After appointment to the position, the employee will complete a training program consistent with training and experience previously received. Prior to becoming fully qualified upon completion of this training, the Analyst may perform work for which proficiency has been demonstrated in accordance with the training program.

QUESTION 422.3

Please provide the qualifications of the person filling the position of Quality Supervisor.

RESPONSE:

As described in Subsection 17.2.1.1.1.4.1.1, the Quality Supervisor meets the requirements set forth in Section 4.4.5 of ANSI/ANS 3.1-1978.

QUESTION 422.4

Provide the position titles of members to be assigned to the Nuclear Review Board or describe the qualification requirements for members of the Board.

RESPONSE:

See Subsection 13.4.2.2 for this information.

QUESTION 423.1

Provide minimum education and experience requirements, at the time of assignment to the function, for: (1) Personnel assigned to conduct preoperational tests (test directors); (2) personnel assigned to conduct startup tests (test directors); (3) personnel assigned to the group responsible for review of preoperational test procedures and results (Test Review Board members), and (4) personnel assigned to the group responsible for review of startup test procedures and results (Plant Operating Review Committee).

RESPONSE:

(1) The minimum qualifications for Preoperational Test Directors at the time of test performance are:

- Bachelor's degree in engineering or the physical sciences or

- High School graduate and four years experience in related testing or operations (or both) of power plants, nuclear facilities or similar industrial installation. Up to two years of this experience may be replaced on a one-for-one basis by successfully completed technical training time in a recognized associated degree program

and

- One year of applicable nuclear power plant experience consisting of:

- Test procedure preparation

- Component initial checkout and testing during Technical Test Phase

- Initial system operation

- System flushing and initial integrated system operation

- Documentation of the above applicable activities per approved Technical Test Procedures

- Attendance at any of the following courses as determined by the ISG supervisor

    - Susquehanna Technology (General Physics)

- BWR Fundamentals (General Electric)
- BWR Design Orientation (General Electric)
- BWR Technology (General Electric)
- Training seminars on Quality

Note that while the Test Director is responsible for directing the preoperational tests, the test procedures and any changes thereto are reviewed by the Test Review Board. Test results are also reviewed and approved by the Test Review Board.

(2) The minimum qualifications of Startup Test directors at the time of test performance are:

- Bachelors degree in engineering or the physical sciences or the equivalent

and

- Two years of applicable power plant experience of which at least one year shall be applicable nuclear power plant experience.

Note that startup test performance is coordinated by the Test Director; however, the actual manipulation of plant equipment is done by or under the direct supervision of the PP&L Shift Supervisor, Assistant Shift Supervisor or Plant Control Operator all of whom will be NRC licensed individuals. The test procedure and any revisions thereto and test results must be approved by the Plant Operations Review Committee.

(3) The minimum qualifications of Test Review Board (TRB) members are as follows:

- Personnel assigned to the Test Review Board shall possess that combination of education and experience recommended in ANSI N18.1-1971 for the position of Operations Manager (Section 4.2.2) or that combination specified in Subsection 14.2.2.2.1, ISG Supervisor.

(4) The qualifications for Plant Operating Review Committee (PORC) members are listed in FSAR Subsection 13.1.3 as referenced in Subsection 13.4.1.1 since this is a permanent plant committee.

QUESTION 423.2

The description of your planned degree of conformance with certain regulatory positions contained in Regulatory Guide 1.68 requires clarification and modification. Regulatory position C.1. describes criteria for selection of plant structures, systems, and components to be tested. Further, Appendix A to the guide provides a representative list of such structures, systems, and components that should be considered for preoperational testing and startup testing. The regulatory cases for this testing includes both Criterion I of Appendix A to 10 CFR 50 and Criterion XI of Appendix B to 10 CFR 50. Your reference to Table 3.2-2 in the FSAR is not acceptable and, therefore, your response should be modified. Further, your categorization of "acceptance tests" should be modified to identify these tests as preoperational tests and the specific controls that will govern the review and approval of test procedures and test results and the conduct of tests for this category should be described. Also, state your plans for review of the results of the tests currently listed as "acceptance tests" prior to fuel loading and provide test abstracts for each test that will identify the test objectives, test methods, and acceptance criteria. Your classification of "power tests" should be modified to "startup tests" to achieve consistency with the terminology used in Regulatory Guide 1.68 and to avoid anticipated interpretation problems between your plant staff and the I&E inspection staff.

RESPONSE:

Testing of safety-related structures, systems and components will be done according to Table 14.2-1, Preoperational Test List. Subsection 14.2.7 of the FSAR has been revised to reference the correct table.

The Preoperational Tests are performed on safety-related equipment. The Acceptance Tests are similar to the Preoperational tests in format, preparation, review and approval. The only difference is that Acceptance tests are done on equipment other than that which is safety-related. While it is our intention to perform all the Acceptance Tests identified on Table 14.2-2, it is not a requirement that they be performed.

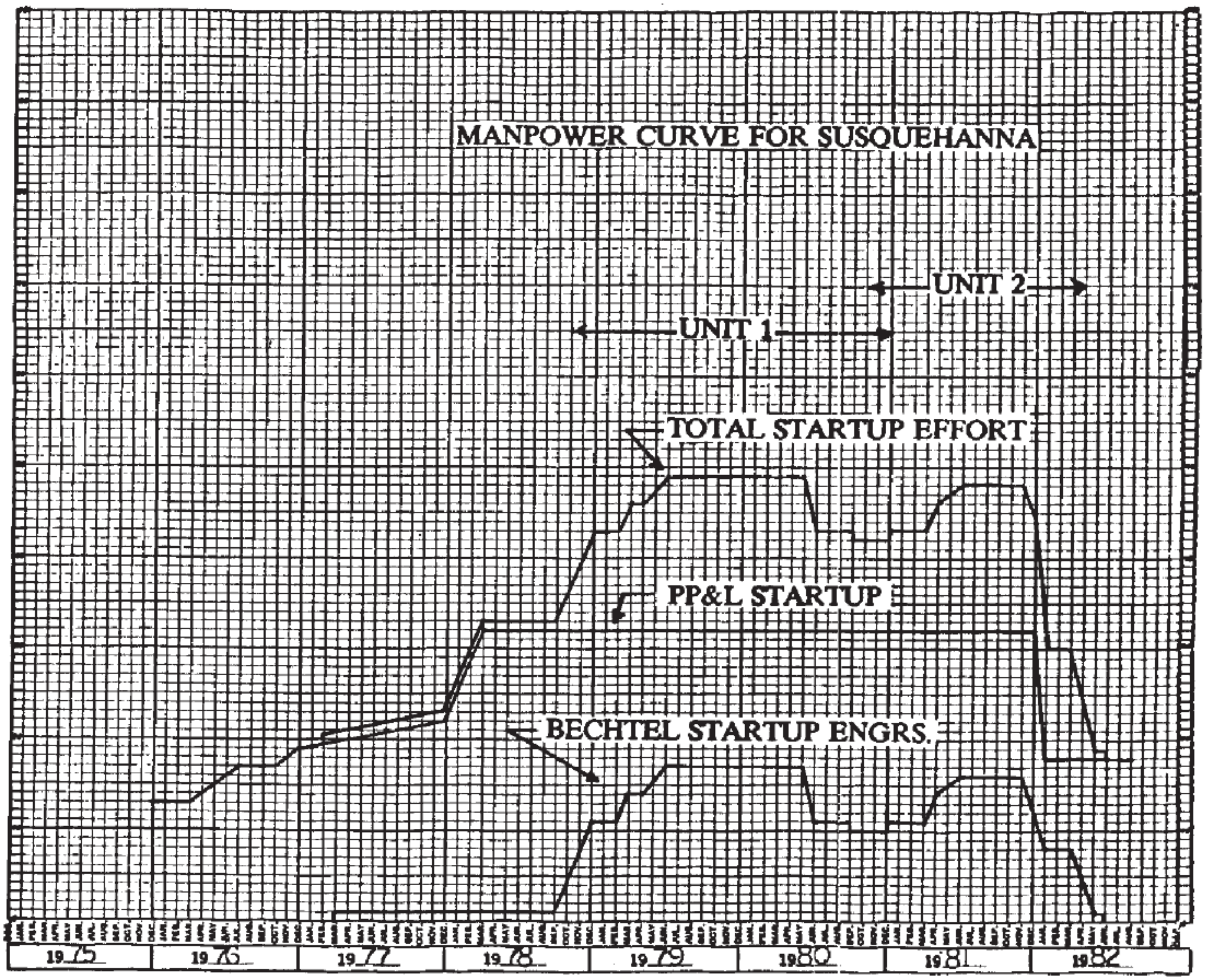
The term "power tests" was adopted by PP&L at the beginning of the startup program development; however, we will change the term "power tests" to "startup tests" as defined in Reg. Guide 1.68. Existing PP&L documents will be modified by Dec. 31, 1979 to reflect this change.

QUESTION 423.3

State the approximate number of test personnel that will be assigned to augment the plant staff (e.g., the integrated startup group) and the approximate schedules (relative to fuel loading) for assignment.

RESPONSES:

The Integrated Startup Group plans to have approximately forty-nine (49) engineers for the Initial Test Program in addition to the Plant Staff people. A schedule for this estimated manpower loading is attached.



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

INTEGRATED STARTUP GROUP
ESTIMATED MANPOWER
LOADING CURVE

FIGURE 423.3-1, Rev 47

AutoCAD: Figure Fsar 423_3_1.dwg

QUESTION 423.4

There appears to be a discrepancy between FSAR Sections 14.2.2 and 14.2.3. To clarify this issue provide a clear statement regarding the Test Review Board responsibility for review of startup test procedures (Phases III, IV, & V).

RESPONSE:

To clarify the duties of the TRB, their participation in the Startup Test Program is to respond to any review requests made by the Plant Operations Review Committee (PORC).

The TRB will not review startup test procedures or results unless requested by PORC whose responsibility it is to recommend approval of these procedures and results to the Superintendent of Plant.

QUESTION 423.5

Describe your controls to assure that plant modifications and repairs identified as a result of plant testing are reviewed, approved, and completed and to assure retesting following such work is completed.

RESPONSE:

Historically, any plant modification or repair done on a system was done under a Start-up Work Authorization (SWA)--Start-up Administration Manual Procedure AD6.4, or a Work Authorization (WA), Plant Administration Procedures Manual Procedure AD-00-046. The current Work Authorization Procedure is NDAP-QA-0502. All these procedures address review, approval, completion and post-work testing involved with modifications and rework. These procedures are available on site for NRC review.

QUESTION 423.6

Describe your provisions for the retention of test records (note: N45.2.9).

RESPONSE:

Retention of test records is addressed in Start-up Administrative Manual Procedures AD 7.6--Preoperational/Acceptance Test Procedure Control and AD 3.3--Start-up Filing Control.

Material filed in any startup file is not indiscriminately removed. A standard "out" card system is used to indicate removal of material from any startup file. The documents will be kept in fire-resistant, lockable cabinets and are considered QA records. These records will be transferred to the permanent plant filing system at some time at the completion of the Initial Test Program.

QUESTION 423.7

You state in response to Question 423.2 that testing of safety-related structures, systems, and components will be done according to Table 14.2-1 and that Subsection 14.2.7 of the FSAR has been revised to reference the correct table. However, you state in 14.2.7 regarding conformance to Regulatory Guide 1.68 that "testing will be conducted . . . identified in Table 3.2-2." Please correct this inconsistency to conform to the position stated in Question 423.2.

RESPONSE:

Subsection 14.2.7 has been revised to reference the proper table (Table 14.2-1).

QUESTION 423.8

Your position to have approved test procedures available for NRC review at least 30 days prior to intended use is not acceptable. Revision 1 to the Standard Review Plan has been revised to, among other items, state that these procedures should be suitable for review at least 60 days prior to their intended use. Revise Section 14.2.7 to be consistent with this position.

RESPONSE:

See revised Subsection 14.2.3

QUESTION 423.9

The information provided in Section 14.2.4.3 states that "if necessary, procedures may be modified to complete testing." This implies that the tests may not be conducted in a manner consistent with that described in your FSAR. Your application should be modified to provide a clear commitment that the tests will be conducted as described or that the FSAR will be modified to reflect identified changes.

RESPONSE:

Subsection 14.2.4 addresses administrative procedures. Subsection 14.2.4.3 addresses test procedures. There is no discrepancy with respect to test changes. Changes to test procedures will not change the intent of the procedure. Any change in FSAR commitment will be proceeded by an FSAR revision.

QUESTION 423.10

Provide test abstracts for the acceptance tests shown in Table 14.2-2 except for the following:

A-3.2 Station Ground System; A-8.1 Domestic Water System; A-9.1 River Water Makeup System; A-9.2 Intake Structure Compressed Air System; A-10.1 Screens and Screen Waste System except for the Emergency Service Water System; A-20.1 Building Drains-Nonradioactive except for those in the ESF equipment rooms; A-21.1 Water Pretreatment System; A-27.1 Auxiliary Boiler System; A-28.2 River Intake Structure H&V System; A-28.4 Chlorination Bldg. H&V System; A-28.5 Circulating Water Pump House H&V System; A-29.1 Administration Bldg. H&V System; A-29.2 Administration Bldg. Chilled Water System; A-37.1 Demineralized Water Transfer System; A-43.2 Condenser Tube Cleaning System; A-74.2 Bulk Hydrogen System; A-85.1 Cathodic Protection System; A-95.1 H Seal Oil System; A-97.1 Stator Cooling System; A-98.1 Main Generator and Excitation System; and A-99.4 Personnel Access Monitors.

Note: We consider that the acceptance tests, except as noted above, should require the same reviews and approvals as your Phase II tests. Modify your FSAR to include these administrative controls.

RESPONSE:

For response see Subsection 14.2.12.3.

QUESTION 423.11

You state that in your testing of containment recirculation fans that it may be possible that they will not be tested to verify that fan motor current is within design. Provide a description of how you plan to verify fan motor currents at conditions representative of accident conditions or provide technical justification for not conforming to the regulatory guide position.

RESPONSE:

See revised discussion on Regulatory Guide 1.68, Appendix A, Section 1.h(10) given in Subsection 14.2.7.

QUESTION 423.12

Regulatory Guide 1.68, Revision 1 (January 1977) is the applicable guide for your facility. However, Revision 2 (August 1978) which incorporates additional industry and ACRS comments provides better guidance than Revision 1. Therefore, we request that you address Revision 2. Our review of your test program description disclosed that the operability of several of the systems and components listed in Regulatory Guide 1.68 (Revision 2), Appendix A may not be demonstrated by your initial test program. Expand your FSAR to include appropriate test descriptions (or modify existing descriptions) to address the following items from Appendix A of the guide:

(1) Preoperational Testing

1.a(4)	Pressure boundary integrity tests.
1.b(3)	Standby liquid control system tests; verification of operability of heaters.
1.c	Demonstration of redundancy, electrical, independence, coincidence, and safe failure on loss of power.
1.d(1)	Turbine bypass valves.
1.d(3)	Relief valves.
1.d(4)	Safety valves.
1.d(9)	Condensate storage system.
1.d(11)	Cooling water system.
1.e(5)	Steam extraction system.
1.e(6)	Turbine stop, control, bypass, and intercept valves.
1.e(8)	Condensate system.
1.e(10)	Feedwater heater and drain systems.
1.e(11)	Makeup water and chemical treatment systems.
1.e(12)	Main condenser auxiliaries used for maintaining vacuum.
1.f(1)	Circulating water system.

1.f(2)	Cooling towers and associated auxiliaries.
1.f(3)	Raw water and service water cooling systems.
1.g(1)	Normal A.C. power distribution system.
1.g(2)	Emergency A.C. power distribution system.
1.h	Tests of structures and equipment (e.g., watertight hatches, walls, floor drains) that protect engineered safety features from flooding (internal and external).
1.h(1)(d)	Demonstration of operability of interlocks and isolation valves provided for overpressure protection for low pressure cooling systems connected to the reactor coolant system.
1.h(2)	Auto depressurization system, including such items as operability using alternate power and pneumatic supplies.
1.h(3)	Containment post-accident heat removal system testing of the containment spray nozzles, spray headers; and demonstration that piping is free of debris.
1.h(8)	Tanks and other sources of water used for ECCS (e.g., condensate storage tanks and suppression pool).
1.i(1)	Containment design overpressure structural tests.
1.i(2)	Containment isolation valve functional and closure timing tests.
1.i(3)	Containment isolation valve leak rate tests.
1.i(4)	Containment penetration leakage tests.
1.i(5)	Containment airlock leak rate tests.
1.i(6)	Integrated containment leakage tests.
1.i(7)	Main steam line leakage sealing systems.
1.i(8)	Primary and secondary containment isolation initiation logic tests.

1.i(9)	Containment purge system tests.
1.i(10)	Containment vacuum-breaker tests (drywell/wetwell).
1.i(13)	Containment inerting system tests.
1.i(15)	Containment penetration pressurization system tests.
1.i(17)	Secondary containment system ventilation tests.
1.i(19)	Bypass leakage tests on the pressure suppression containment.
1.i(21)	Containment penetration cooling system tests.
1.j(2)	Feedwater control system.
1.j(7)	Leak detection systems to detect failures in ECCS.
1.j(9)	Pressure control systems used to maintain design differential pressures to prevent leakage across boundaries (feedwater leakage control).
1.j(10)	Seismic instrumentation.
1.j(11)	Traversing incore probe system.
1.j(12)	Failed fuel detection system.
1.j(16)	Hotwell level control system.
1.j(17)	Feedwater heater temperature, level, and bypass control systems.
1.j(18)	Auxiliary startup instrument tests (neutron response checks).
1.j(19)	Instrumentation and controls used for shutdown from outside the control room.
1.j(21)	Reactor mode switch and associated functions.

1.j(22)	Instrumentation that can be used to track the course of postulated accidents such as containment wide-range pressure indicators, reactor vessel water level monitors, pressure suppression level monitors, high-range radiation detection devices, and humidity monitors.
1.j(24)	Annunciators for reactor control and engineered safety features.
1.j(25)	Process computers.
1.k(2)	Personnel monitors and radiation survey instrument tests.
1.k(3)	Laboratory equipment used to analyze or measure radiation levels and radioactivity concentrations.
1.k(4)	High Efficiency Particulate Air (HEPA) filter and charcoal absorber efficiency and in-place leak tests.
1.1(2)	Gaseous radioactive waste handling systems.
1.1(3)	Solid waste handling systems. Solidification system tests should include verification that no free liquids are present in packaged wastes.
1.1(5)	Isolation features for condenser offgas systems.
1.1(6)	Isolation features for ventilation systems.
1.1(7)	Isolation features for liquid radwaste effluent systems.
1.1(8)	Plant sampling systems.
1.m(1)	Spent fuel pit cooling system tests, including the testing of antisiphon devices, high radiation alarms, and low water level alarms.
1.m(3)	Operability and leak tests of sectionalizing devices and drains and leak tests of gaskets or bellows in the refueling canal and fuel storage pool.

1.m(4) Dynamic and static load testing of cranes, hoists, and associated lifting and rigging equipment, including the fuel cask handling crane. Static testing at 125% of rated load and full operational testing at 100% of rated load.

1.m(5) Fuel transfer devices.

1.m(6) Irradiated fuel pool or building ventilation system tests.

1.n(1) Service water cooling system.

1.n(2) Turbine building cooling water systems.

1.n(5) Sampling systems.

1.n(6) Chemistry control systems for the reactor coolant system (condensate demineralizers).

1.n(7) Fire protection systems.

1.n(8) Seal water systems.

1.n(9) Vent and drain systems for contaminated or potentially contaminated systems and areas and drain and pumping systems serving essential areas, e.g., spaces housing diesel generators, essential electrical equipment, and essential pumps.

1.n(11) Compressed gas systems.

1.n(13) Communication systems.

1.n(14) Heating, cooling, and ventilation systems serving the following:

(a) Diesel generator buildings.

(b) Turbine building and radioactive waste handling building.

1.n(15) Shield cooling systems.

1.n(18) Heat tracing and freeze protection systems.

1.o(1) Dynamic and static load tests of cranes, hoists, and associated lifting and rigging equipment (e.g., slings and strongbacks used during refueling or the preparation for refueling).Static testing at 125% of rated load and full operational testing at 100% of rated load.

1.o(2) Demonstration of the operability of protective devices and interlocks.

1.o(3) Demonstration of the operability of safety devices on equipment.

(2) Initial Fuel Loading and Precritical Tests

2.d Final test of the reactor coolant system to verify that system leak rates are within specified limits.

2.h Mechanical and electrical tests of incore monitors, including traversing incore monitors, if installed.

(4) Low Power Testing

4.d Verification that proper operations of associated protective functions and alarms provide for plant protection in the low-power range.

4.e Flux distribution measurements.

4.g Determination of proper response of process and effluent radiation monitors.

4.i Demonstration of the operability of rod inhibit or block functions.

4.l Demonstration of the operability, including stroke times, of branch steam line valves and bypass valves.

4.m Demonstration of the operability of main steam line isolation valve leakage control system at hot standby conditions.

4.r Demonstration of the operability of reactor condensate cleanup system.

(5) Power-Ascension Tests

5.a Demonstration that power vs. flow characteristics are in accordance with design values.

5.c Control rod pattern, the exchange demonstration.

5.g Demonstrate that control rod sequencers, control rod worth minimizers, and rod withdrawal block functions operate in accordance with design.

5.1 Demonstrate design capability of turbine bypass valves.

5.m Demonstrate that the reactor coolant system flows, pressure drops, and vibrations are in accordance with design for various operating modes.

5.o Calibration of instrumentation and demonstration of proper response of reactor coolant leak detection systems.

5.t Verify, as appropriate, response times and set points for main steam line relief valves; turbine bypass valves; and turbine stop, intercept, and control valves.

5.u Verify response times of branch steam line isolation.

5.w Demonstrate adequate performance margins for shielding and penetration cooling systems capable of maintaining temperatures of cooled components within design limits with the minimum design capability of cooling system components available (100%).

5.x Demonstrate adequate beginning-of-life performance margins for auxiliary systems required to support the operation of engineered safety features or to maintain the environment in spaces that house engineered safety features. Engineered safety features will be capable of performing their design functions over the range of design capability of operable components in these auxiliary systems (50%, 100%).

5.z Demonstrate that process and effluent radiation monitoring systems are responding correctly.

5.c.c Demonstrate that gaseous and liquid radioactive waste processing, storage, and release systems operate in accordance with design.

5.f.f Demonstrate that the ventilation system that serves the main steam line tunnel maintains temperature within the design limits.

5.h.h Demonstrate that the dynamic response of the plant to the design load swings for the facility.

5.i.i Demonstrate that the dynamic response of the plant is in accordance with design for closure of reactor coolant system flow control valves.

5.1.1 Demonstrate that the dynamic response of the plant is in accordance with design requirements for turbine trip.

RESPONSE:

Preoperational tests of safety related systems are described by the test abstracts provided in Subsection 14.2.12.1. Specific detailed guidelines for testing such a loss of power, air, etc. are described in the startup administration manual Section 7.5. Loss of power is tested if it causes an evolution to occur within the system such as switching automatically to a different power source. Loss of air testing is performed by placing the valve in its non-failed position by normal actuator operation, then isolating the actuator air supply, bleeding off air pressure and verifying valve movement to the failed position. Each automatic containment isolation valve is tested in the system pre-op test for proper operation and closure timing as required by the design sections of the FSAR. Leak detection systems such as steam leak detection are tested in the system pre-ops affected by the detection system.
Each item is answered as follows:

1. 1.a(4) - Hydro - All ANSI B31.1, ASME Boiler and Pressure Vessel Code Sections I, III and VIII, NFPA code, and plumbing code piping is hydrostatically tested. Two primary hydrostatic tests are conducted on the Reactor Pressure Vessel, recirculation system and main steam lines: A primary hydro at 125% of generating pressure with the internals removed and an operational hydro at 100% operating pressure with the internals installed.

2. 1.b(3) - Verification of chemical mixing and sampling will be covered by the Technical Specification Surveillance requirements per 4.1.5.

3. 1.c - See abstract for P100
   - See General Test Statement

4. 1.d(1) - See abstract for A93.2

5. 1.d(3) - See abstract for P83.1

6. 1.d(4) - See abstract for P83.1

7. 1.d(9) - See abstract for A37.1

8. 1.d(11) - Service Water is not safety-related. It is tested by Acceptance Test A11.1. The RHR Service Water System is the plant system which falls under section 1d of Regulatory Guide 1.68. The RHR Service Water System is tested in P16.1.

9. 1.e(5) - Extraction Steam - See abstract A46.1

10. 1.e(6) - Expansion monitoring is done on NSSS and the feedwater piping inside containment after fuel load. No other monitoring of BOP systems is anticipated. (ST-17)

11. 1.e(8) - See abstract for A44.1

12. 1.e(10) - Feedwater Heaters & Drain Systems - See abstract A46.1

13. 1.e(11) - With the condensate polisher under normal operating conditions, Bechtel Corp. will make a complete inspection of all piping and hangers to verify adequate expansion and restraint capability. Test No. A22.1 will be performed to verify correct system operation.

14. 1.e(12) - See abstract for A43.1

15. 1.f(1) - See abstract for A42.1

16. 1.f(2) - See abstract for A41.1

17. 1.f(3) - Service Water is not safety-related. It is tested by Acceptance Test A11.1.

18. 1.g(1) - See abstracts for A3.1, P4.1, P5.1 and A7.1.

    19.1.g(2) - See abstracts for A3.1, P4.1, P5.1 and A7.1.

20. 1.h - These features are tested under 2 tests:

    1) P69.1 - Liquid Radwaste Collection

    2) P76.1 - Plant Leak Detection

21. 1.h(1) (d) - Added to abstract P49.1

22. 1.h(2) - See abstract for P83.1 23.

23. 1.h(3) - Demonstrated during flush; not part of P.O. No change.

24. 1.h(8) - Proper operation of valve sequencing for ECCS pump suction from the Condensate Storage Tank and suppression pool is tested in the system preop tests for those systems supplied by water from these systems. Alarms, etc., are tested in A37.1 for the CST and in P59.1 for the suppression.

25. 1.i(1) - The containment design overpressure structural test is the Structural Integrity Test performed as a construction test.

26. 1.i(2) - See General Test Statement

    1.i(2) - Revised abstract for Reactor Water Cleanup

    1.i(2) - Added to abstract P59.1

    1.i(2) - See abstract for P59.1

27, 28,
29. 1.i(3), (4), (5) - The tests covered by these portions of Reg. Guide 1.68 are Type B and Type C local leakage rate tests. The tests are conducted as part of the Component Inspection and Testing Phase. These local leakage rate tests are conducted prior to and as prerequisites to the Containment Integrated Leak Rate Test. Each Type B and Type C test is conducted in accordance with the requirements of Subsection 6.2.6 of the FSAR. Acceptance criteria for the Type B and Type C tests is in accordance with the requirements of Chapter 16 of the FSAR.

30. 1.i(6) - See abstract for P59.2

31. 1.i(7) - See abstract for P83.1

32. 1.i(8) - Primary containment isolation initiation logic is tested in P59.1. Secondary containment isolation initiation logic is tested in P34.1.

33. 1.i(9) - See revised abstract for P73.1

34. 1.i(10) - See revised abstract for P73.1

35. 1.i(13) - See revised abstract for ST-37.

36. 1.i(15) - This is not applicable to Susquehanna since leakage surveillance by means of a permanently-installed system with provisions for continuous or intermittent pressurization of individual or groups of containment penetrations is not part of Susquehanna design.

37. 1.i(17) - See abstract for P34.1

38. 1.i(19) - See abstract for P59.1

39. 1.i(21) - Not applicable to Susquehanna SES design.

40. 1.j(2) - See abstract for P45.2

41. 1.j(7) - Leak detection for the HPCI (ECCS) and RCIC systems is tested in their respective pre-operational tests. There is no leak detection system for core spray or the containment spray mode of RHR. The leak detection and isolation of the RHR shutdown cooling mode is tested in the RHR pre-op. Overall steam leak detection logic is tested in one of the Main Stream Pre-op's.

42. 1.j(10) - See abstract for A99.6.

43. 1.j(11) - See revised abstract

44. 1.j(12) - The off-gas pre-treatment system linear Wide Range Monitor detects failed fuel and is tested with other Process Radiation Monitors in P79.20.

45. 1.j(16) - See abstract for A44.1

46. 1.j(17) - Feedwater heater temp, level and by-pass control systems - See abstract A46.1.

47. 1.j(18) - Neutron response checks are part of the Power Test Program (STs 6, 10, 11, 12, & 18). Preoperational testing is addressed in abstracts P78.1, P78.2, P78.3 and P78.4.

48. 1.j(19) - Not a separate system tested in each ECCS System

    1.j(19) - See revised abstract for P54.1

    1.j(19) - Instrumentation and controls used for shutdown from outside the control room are tested under their respective system pre-operational tests.

49. 1.j(21) - See abstract for P58.1

50. 1.j(22) - Containment instrumentation is tested in the following pre-op tests:

    - Reactor Wide Range Pressure - P45.1 Feedwater Control
    - Reactor Level - P45.1 Feedwater Control and P80.1 Reactor Non-Nuclear Instrumentation
    - Suppression Pool Level - P59.1 Containment and Suppression
    - Radiation Detection - P79.1 Area Radiation Monitoring and P79.2 Process Radiation Monitoring.
    - Humidity Monitors - Not in present Susquehanna SES design.

51. 1.j(24) - See abstract for Annunciator System

52. 1.j(25) - See abstract for Process Computer

53. 1.k(2) - See answer below for 1.k(3)

54. 1.k(3) - Laboratory equipment testing, calibration, etc., is discussed in Subsection 12.5.2 of the FSAR.

55. 1.k(4) - HEPA filters and charcoal efficiency were tested by factory representatives on-site but not prior performing HVAC pre-op tests. The pre-op test was reviewed and it was verified that pretesting of the HEPA filters and charcoal efficiency was not required.

56. 1.1(2) - See abstract for A72.1

57. 1.1(3) - See abstract for A68.1

58. 1.1(5) - See abstract for A43.1

59. 1.1(6) - See abstract for P34.1 and General Test Statement

60. 1.1(7) - Liquid radwaste effluent discharge to the environment is tested in Acceptance Test A69.2.1.

61. 1.1(8) - Plant Sampling System - Test A76.2 is Process Sampling Test, and tests all the Sample Stations on site. Test P76.1 is Plant Leak Detection Test and verifies the operability of the leak detection.

62. 1.m(1) - See revised abstract, part of TP1.9 for fuel pool

63. 1.m(3) - Following erection of the liner plates for the spent fuel pool, dryer separator pool and reactor basin cavity, the pools are filled with water and left to stand for 48 hours during which leakage is monitored. Helium leak testing is utilized to locate leaks. The pool gates are hydrostatically tested by filling the spent fuel pool and monitoring the leakage to the reactor cavity side of the gates.

64. 1.m(4) & 1.o(1) - For testing of the fuel handling system, see the abstract for P81.1. For testing of the reactor building crane see the abstract of P99.1.

65. 1.m(5) - See abstract for P81.1 66.

66. 1.m(6) - The refueling floor HVAC system is considered Zone 3 of the Reactor Bldg. HVAC system and is tested in P34.1.

67. 1.n(1) - Service Water is not safety-related at Susquehanna SES. It is tested per A11.1.

68. 1.n(2) - See abstract for A15.1.

69. 1.n(5) - Reactor Coolant and Secondary Sampling systems - See abstract A76.2.

70. 1.n(6) - System 39 Condensate Demineralizer and Regeneration System is tested under Acceptance Test A39.1. See abstract A39.1.

71. 1.n(7) - Fire Protection Systems are tested by Preoperational Tests P13.1 through P13.4.

72. 1.n(8) - The seal water for the reactor recirculation pumps is supplied by the CRD system. The seal water is tested in the Recirculation Pre-op Test P64.1.

73. 1.n(9) - See abstract for A20.1

74. 1.n(11) - Tested in P25.1

75. 1.n(13) - See abstract A99.2.

76. 1.n(14) - See abstracts for P28.3, A33.1, A33.2, A65.1 and A65.2.

77. 1.n(15) - Not applicable to Susquehanna SES design.

78. 1.n(18) - See abstract A85.2.

79. 1.m(4) & 1.o(1) - For testing of the fuel handling system, see the abstract for P81.1. For testing of the reactor building crane, see the abstract for P99.1.

80. 1.o(2) - For testing of protective devices and interlocks on the fuel handling system and reactor building crane, see the abstracts for P8.1 and P99.1

81. 1.o(3) - For testing of safety devices on the fuel handling system and reactor building crane, see the abstracts for P81.1 and P99.1.

82. 2.d - Reactor coolant leak detection systems are placed in service and tested per Plant Technical Specifications. These systems are pre-op tested in the appropriate pre-op tests. In addition, an operational

hydro of the reactor is performed per Plant Surveillance Tests. No change to the test description is required.

83. 2.h - Mechanical tests of the SRM, IRM and TIP drive mechanism are tested in P78.1, 78.2 and 78.4. The APRM (including LPRM's) system is electrically tested in P78.3. Further, ST-6, ST-10, ST-11, ST-12, and ST-18 demonstrate the overall operability of the nuclear instrumentation systems. As such, no change to the test description is required.

84. 4.d - SRM and IRM alarms are tested in their respective preop, P78.1 and P78.2. The SCRAM function is tested in P78.1 and P78.2, and the Reactor Protection System preop P58.1. No change to the test description is required.

85. 4.e - Flux distributions are not used to verify the items identified in section 4.e. Enrichment of the fuel rods and subsequently the fuel bundles is verified by the fuel manufacturer prior to shipping. The required location of the fuel assemblies is verified in ST-3. Proper control rod positioning is verified in P55.1 and control rod coupling is verified in ST-6.

However, it should be noted that during ST-18 (TIP Uncertainty), which is performed at Test Condition (TC) 3 and 6, the random noise, geometric, and total uncertainty of the TIP trace are determined for an octant symmetrical core and rod pattern. Some of the factors which would cause excessive uncertainty are fuel enrichment and/or poisoning errors, improper fuel loading, and mispositioned fuel rods.

86. 4.g - Proper responses of the Area and Process Radiation Monitoring Systems are verified in P79.1 and P79.2 respectively, by using radioactive samples. ST-1 (Chemical and Radiochemical) provides for calibration of monitors in the liquid waste system and liquid process lines. ST-37 (Gaseous Radwaste) provides for demonstrating proper operation of the Gaseous Radwaste System. Further, Plant Tech Specs require periodic surveillance of the radiation monitoring systems to ensure proper operation during the appropriate plant conditions.

87. 4.i - The Operation of the Reactor Manual Control System, including RSCS and RWM, is verified in P56.1. These systems are required by Plant Tech Specs to be

operable during startup and to demonstrate their operability prior to initiating startup. Therefore, there is not a dedicated startup test which demonstrates their operability. As such, no test description is required.

88. 4.1 - MSIV's are demonstrated operable including stroke times in ST-25 (MSIV) at TC 1,2,3,5 and 6. Main Steam bypass valves are demonstrated operable, including stroke times, in ST-24 (Turbine Valve Surveillance) at TC 3,5, and 6. Branch Steam Line isolation valves (HPCI, RCIC, and MSIV-LCS) are not tested in a Startup Test. However, they are demonstrated operable, including stroke times, in their surveillance procedures as required by Administrative Procedure AD-000-75 (Station Inservice Inspection Programs). No changes to the test descriptions need be made.

<u>NOTE</u>: MSIV-LCS information maintained here for historical purposes. The MSIV-LCS has been deleted. The function is now performed by the Isolated Condenser Treatment Method (Section 6.7).

89. 4.m - The MSIV-LCS is initially verified operable in P83.1. Subsequently, the system is periodically verified operable per surveillance procedures as required by the Technical Specifications. There is no additional Startup Test deemed necessary.

90. 4.r - The RWCU system is partially tested in P61.1. The balance of testing required nuclear heating and is performed in ST-7 (RWCU). No change to test description is warranted.

91. 5.a - Demonstrations that power vs. flow characteristics are in accordance with design values are done in various Startup Tests (ST's) as described below. Refer to Figure 14.2-6-1, for definitions of terms used in the descriptions.

(1) ST-6 demonstrates line b from 0% to ~ 25% power, and line c from ~ 25% to the intersection of line c with the 100% rod line.

(2) ST-21 demonstrates the intersection of line b with the 100% rod line.

(3) ST-35 demonstrates line d at ~ 50% and 100% power.

(4) ST-30 demonstrates line d from ~ 50% power to the intersection with the flow interlock line and also demonstrates that cavitation does not occur above and at the flow interlock line.

(5) ST-29 performs testing along the 100% rod line (with Xenon buildup).

(6) ST-19 demonstrates that the plant operates below the 100% rod line at TC 4, 5, and 6.

92. 5.c - Rod sequence exchange is performed in ST-34 (Control Rod Sequence Exchange). The test description has been modified.

93. 5.g - See response for 4.i.

94. 5.1 - The design capability of turbine bypass valves is demonstrated in ST-27 (Turbine Trip and Generator Load Rejection). The test description has been modified accordingly.

95. 5.m - The reactor recirculation system is initially tested, calibrated, and evaluated against design performance parameters in P64.1. During Startup Test Program, the system operating parameters are evaluated in ST-19 (Core Performance), ST-30 (Recirculation System) and ST-35 (Recirculation System Flow Calibration). Vibration levels for piping in the Recirculation System are evaluated in ST-33 (Drywell Piping Vibration). No change to the respective test descriptions is deemed necessary.

96. 5.o - Calibration of instrumentation for reactor coolant leak detection systems is performed during the turnover/checkout phase prior to preoperational testing. Demonstration of proper instrumentation response is performed during the system's preoperational test. In addition, the reactor coolant system leakage detection systems are periodically verified operated and calibrated as required by Tech Specs. No change to test descriptions is required.

97. 5.t - Main Steam Safety Relief Valves are factory tested to verify operability, response times, relieving capacities, setpoints and reseat pressures. Startup Test ST-26 verifies proper SRV operation and relative relieving capacities. Periodic surveillance operating tests are conducted to demonstrate SRV operability in accordance with the Technical Specifications. For all

tests performed in the factory, the test method, test results and methods of extrapolation (if required) of the data to actual plant conditions are reviewed, documented and retained.

Turbine bypass valve operability, response time and relieving capacity is qualitatively verified in the Generator Load Reject Within Bypass Valve Capacity test (part of ST-27). Operability is also verified in ST-24.

Turbine Stop, Control and Combined Intermediate Valve operability is verified in ST-24. The response times of these valves is qualitatively verified in ST-27.

98. 5.u - Main Steam Isolation Valves are tested for operability and response time in ST-25. Periodic surveillance tests are also performed per Technical Specifications. Reactor Feedwater Pump Turbine Steam isolation valve is tested for operability in P45.1

RCIC and HPCI steam line isolation valves are tested for operability and response time in P50.1 and P52.1. Periodic surveillance testing is conducted to verify continued proper response times.

99. 5.w - Not applicable to Susquehanna SES design.

100. 5.x - RBCCW, TBCCW, and Service Water systems are tested in ST-36 (Cooling Water Systems) to verify their adequate performance. The tests are performed at TC2, 3 and 6. The Containment Atmosphere Circulation System is tested in ST-32 at TC 2 and 6. The H&V systems for the DG Building, ESSW Pumphouse, Reactor Building, and Control Structure are tested in P28.3, P28.1, P34.1 and P30.1 respectively. The RBCW system is tested in P34.2.

The RHR Service Water System is tested in ST8 (RHR System) and is also verified operable in ST-28 (Shutdown From Outside the Main Control Room) These tests are performed at TC6 (ST-8) and TC1 (ST-28)

Emergency Service Water is tested in ST-36 (Cooling Water Systems)

101. 5.z - See response to Item 4.g.

102. 5.c.c - Gaseous radwaste system is tested in ST-37 at TC 1, 3, 5, and 6. Liquid Radwaste Collection System is demonstrated operable in P69.1. Solid Radwaste

Systems, Liquid Radwaste System, and Gaseous Radwaste Systems are demonstrated operable in A68.1, A69.2, and A72.1 respectively.

103. 5.f.f - ST-32 (Containment Atmosphere and Main Steam Tunnel Cooling) demonstrates the operability of the systems (or portions of systems) which provide cooling for the primary containment and the main steam tunnel. These tests are demonstrated at TC2 and 6. Refer to revised abstract.

104. 5.h.h - Load swings for the plant, both upward and downward step and ramp changes, are tested in ST-29 (Recirculation Flow Control) at TC-1, 2, 3, and 5. Plant response to load swings are also demonstrated in ST-30 (Recirculation System) Refer to revised abstracts ST-29 and ST-30.

105. &
106. 5.i.i - ST-30 (Recirculation System) tests one-pump trip at TC-3 and 6, and tests a two pump trip at TC-3 only. Reactor coolant flow control valve not applicable on SSES.

107. 5.1.1 - ST-27 (Turbine Trip and Generator Load Rejection) tests a turbine trip at TC 3 and tests a generator load rejection at TC6. No change to test description required.

QUESTION 423.13

Expand your test abstracts of Section 14.2.12 and those provided in answer to Questions 423.10 and 423.12 to describe in more detail the test objectives, prerequisites, test method, and acceptance criteria in regard to applicable parameters and functions (e.g., pressure, temperature, flow, valve operability, valve opening and closure times, controls, logics, and interlocks).

RESPONSE:

Abstracts for the following preops have been revised: P2.1, 4.1, 5.1, 16.1, 17.1, 24.1, 30.1, 34.1, 45.1, 49.1, 51.1, 52.1, 53.1, 54.1, 55.1, 57.1, 58.1, 59.1, 59.2, 61.1, 73.1, 75.1, 76.1, 78.4, 83.1, 99.1, and 100.1. These abstracts are consistent with other licensing applications. The NRC will receive a draft copy of each test as it is developed and a copy of the approved test 60 days prior to its run date.

QUESTION 423.14

We note your position relative to Regulatory Guide 1.80 contained in Section 14.2.7 of the FSAR and disagree with your position. This guide is applicable since the instrument air system is used for a source of air for systems and components that provide a safety function. Modify your application to show that your test program will be consistent with the guide or show that you will conduct equivalent testing for the air system and supplied loads.

RESPONSE:

The primary containment instrument gas system will be tested in accordance with the requirements of Regulatory Guide 1.80 Sections C.1 through C.6. The portions of the instrument air system which supply safety related equipment will also be tested in accordance with sections C.1 through C.6 of Regulatory Guide 1.80 (June, 1974).

The various components fed by the instrument air and instrument gas system will be tested to ensure proper operation on loss of air/gas. This testing will be done as part of the various systems preoperation testing in which the components are located.

The action and flow of decay air is not an essential criteria of operation in relation to the affected components. The components are to fail with loss of air/gas to a safe position. Whether decaying pressure will hold some or all of the valves in normal operating positions is not of critical importance.

In addition to the above testing, the systems/components which have separate accumulators (MSIV's, safety relief valves) will be tested with a loss of air/gas to ensure that the accumulators function in accordance with design.

Loss of air testing as described above will be done in the various system preoperational tests. Therefore testing described in Regulatory Guide 1.80 Sections C.7 through C.10 will not be done in the instrument air system or primary containment instrument gas system tests.

See revised Section 3.13.

QUESTION 423.15

We could not conclude from our review of the preoperational test phase and the test abstracts provided in Table 14.2 that comprehensive testing is scheduled for several of the described tests. Therefore, clarify or expand the description of the preoperational test phase to address the following:

(1) Modify the individual A.C. and D.C. distribution system test descriptions or provide an integrated test description to verify proper load group assignments (reference Regulatory Guide 1.41).

(2) Class 1E 125 Volt D.C. System Preoperational Tests - State your plans for demonstrating the following: (a) that emergency loads are in accordance with battery sizing assumptions; and (b) that each emergency load can operate at the minimum voltage level at which it can be postulated to operate.

(3) State how operability of emergency loads using offsite power will be demonstrated during A.C. and D.C. system tests.

(4) Identify testing that will be accomplished to verify drywell floor bypass leakage and provide quantitative acceptance criteria.

(5) State your plans for assuring that the effects of interfacing hardware (e.g., snubbers, pulse dampers) located between measured variables and the input to the sensors for the Reactor Protection System do not compromise the channel response time requirements.

(6) Control Room HVAC System Preoperational Test - Expand the test description to include a demonstration that outleakage from the control room is in accordance with design assumptions when the system is on the emergency outside air supply.

RESPONSE:

(1) See abstracts for P2.1, A3.1, P4.1 and P5.1.

(2) See revised abstract for P2.1.

(3) See abstract for P100.1.

(4) Testing will be done per P69.1.

(5) In our tests, we do not address the effects of interfacing hardware located between measured variables and the input to the sensors on the channel response time for the Reactor Protection System.

(6) See revised abstract for P30.1.

QUESTION 423.16

Describe your tests to demonstrate that the core spray flow distribution header provides adequate cooling flow to each fuel assembly.

RESPONSE:

Preoperational test P51.1A, Core Spray System Pattern Test, describes the demonstration of adequate core spray cooling flow to each fuel assembly.

QUESTION 423.17

Provide a description of the electrical lineup for Unit No. 2 during preoperational tests that will be conducted to satisfy regulatory positions in Regulatory Guide 1.41 for Unit No. 1. Provide a description of the lineup for both plants during similar preoperational testing on Unit No. 2 subsequent to initial criticality of Unit No. 1. The descriptions should address both normal and emergency A.C. and D.C. power distribution systems. Provide assurance that crossties will not exist which could cause loss of emergency bus power to one unit due to testing of the other unit.

RESPONSE:

Unit 1 and Unit 2 13.8 KV systems will be jointly tested before Unit 1 initial criticality (Acceptance Test A3.1).

Unit 1 and Unit 2 4.16 KV systems will be jointly tested before Unit 1 initial criticality (Preoperational Test P4.1).

Lineups for these tests will be for Unit 1 testing. There will be no testing of the Unit 2 systems after initial criticality of Unit 1.

This integrated testing and system design will satisfy the requirements of Regulatory Guide 1.41 and will assure that no crossties exist which might cause loss of emergency bus power to one unit due to testing the other unit.

The design of the 480 volt systems allows for the isolation of each unit from the other. This allows testing on one unit without affecting the other.

The Unit 1 and Unit 2 DC systems are entirely separated. There are no crossties.

QUESTION 423.18

Provide a commitment to include in your test program any design features to prevent or mitigate anticipated transients without scram (ATWS) that may be incorporated in your plant design.

RESPONSE:

FSAR Table 15.0-1a lists ATWS analysis as, "still under discussion." Testing will be done to the extent practicable to ensure compliance to any ATWS design when that design is finalized for Susquehanna SES.

QUESTION 423.19

Provide preoperational test descriptions (or modify existing descriptions) to assure that each engineered safety feature pump operates in accordance with the manufacturer's head-flow curve. Include in the description the bases for the acceptance criteria. (The bases provided should consider both flow requirements for ESF functions and pump NPSH requirements.)

RESPONSE:

Testing to verify that ESF pumps operate within their design pump-head curves and with adequate NPSH will be done. This testing is committed to in the General Test Statement as part of the answer to Q423.12.

Steam conditions from the two station auxiliary boilers will permit turbine testing of RCIC and HPCI systems. However it will prohibit any pump testing. This will be verified under the power test program using nuclear steam.

QUESTION 423.20

Our review of the power test abstracts provided in your FSAR disclosed that they are not sufficiently descriptive to conclude that comprehensive testing is planned or that satisfactory test acceptance criteria have been established. The individual test abstracts should be modified as indicated below.

(1) Modify your acceptance criteria for test PT-1, Chemical and Radiochemical, to provide a level 2 acceptance criteria of design basis for your condensate demineralizers and RWCU.

(2) Your acceptance criteria for test PT-2, Radiation Measurements, is not consistent with the design objectives of ALARA. Therefore, revise your acceptance criteria to be consistent with your plant design objectives.

(3) Modify your test abstract for Full Core Shutdown Margin to specify the value R. In addition, specify a quantitative value for your level 2 criterion and that value be considered a level 1 criterion.

(4) Revise all power test acceptance criteria where you use the term "specified value" to provide a specific numerical value for those acceptance criteria.

(5) Test PT-4, Full Core Shutdown Margin - Provide the temperature of the core for the shutdown margin test.

(6) Test PT-5, Control Rod Drive System - The level 1 acceptance criteria for control rod withdrawal speeds are inconsistent and nonconservative in respect to the times assumed in your accident analysis. Resolve this inconsistency. Also, this test abstract should be expanded to provide assurance that dash-pot performance will be in accordance with design requirements and acceptance criteria should be provided for control rod scram times.

(7) Test PT-9, Water Level Measurement - Revise Figure 14.2-5 to include water level measurement tests at Test Conditions 1, 3, 4, & 5 in addition to those already specified.

(8) Test PT-10, IRM Performance - Revise the acceptance criteria to include a check for the IRM scram trip point.

(9) Test PT-14, RCIC System - State your plans to demonstrate the capability of the system to start from the "cold" condition. Also, clarify or justify the Level 1 acceptance criteria provided in Paragraph 4. Based on operating experience to date, the apparent reliability of the reactor core isolation cooling system (RCIC) in BWR plants has been poor. Because it appears that many of the causes for the failure should have been detected and corrected during initial testing of the RCIC system, this system should be given a very thorough checkout during the initial testing program. Your current test proposal does not appear adequate to establish confidence in the reliability of the system for your facility. Your application should be modified to show that several consecutive successful cold starts of the RCIC system will be demonstrated during your power ascension phase.

(10) Test PT-15, HPCI System - Based on operating experience to date, the apparent reliability of the high pressure core injection system (HPCI) in BWR plants has been poor. Because it appears that many of the causes for the failure should have been detected and corrected during initial testing of the HPCI system, this system should be given a very thorough checkout during the initial testing program. Your current test proposal does not appear adequate to establish confidence in the reliability of the system for your facility. Your application should be modified to show that several consecutive successful cold starts of the HPCI system will be demonstrated during your power ascension phase.

(11) Test PT-16, Selected Process Temperatures - Modify your Level 1 acceptance criteria to include the pump in an idle loop; and your Level 2 acceptance criteria to relate to loop temperature.

(12) Test PT-18, Core Power Distribution - Revise your test method to specify how many sets of TIP data will be taken to determine the overall TIP uncertainty.

(13) Test PT-22, Pressure Regulator - Specify the mode of control (auto or manual) of each of the other principal control systems at each test condition.

(14) Test PT-23, Feedwater System - Modify your test objectives to include the loss of a feedwater heater. Specify the mode of control (auto or manual) of each of the other principal control systems at each test condition for the feedwater control setpoint changes.

Also, the test description should be modified for the feedwater heater trip to specifically identify: (a) the type of trip to be initiated; (b) the feedwater heater(s) involved; and (c) a discussion of how the planned trip relates to the worst case limiting event for your design that could result from a single equipment failure or operator error. Modify your test method to include the loss of all feedwater flow. Provide justification for performing the feedwater pump trip in Master Manual Flow Control Mode rather than Automatic Flow Control Mode for feed water pump trip and for feedwater heater loss.

(15) Test PT-25, Main Steam Isolation Valves - Provide clear acceptance criteria for relief valve and RCIC performance during this transient.

(16) Test PT-26, Relief Valves - Describe your test method and acceptance criteria for bypass valve flow calibration and capacity. Modify your acceptance criteria to include opening times (to full capacity) of relief valves.

(17) Test PT-27, Turbine Trip and Generator Load Rejection - Modify your test abstract to: (a) identify the method of tripping the main generator breaker; (b) identify the conditions for each trip planned; (c) identify the variables or parameters to be monitored for each trip; (d) provide assurance that test results will be compared with predicted results for the actual tests to be run (for each trip); (e) provide quantitative acceptance criteria and their bases for the required degree of convergence of actual test results with predicted results for the monitored variables and parameters for each trip; and (f) provide acceptance criteria for grid stability, voltage and frequency following generator load rejection trips.

(18) Test PT-28, Shutdown From Outside the Main Control Room - State whether the plant's electrical system will be aligned for normal full power operation and provide acceptance criteria for the performance of plant equipment and the variables or parameters to be monitored during the test.

(19) Test PT-29, Recirculation Flow Control - Specify the mode of control (auto or manual) of each of the other principal control systems at each test condition.

(20) Test PT-30, Recirculation System - Modify the test

abstract to define the types of trips to be conducted at each test condition and the manner by which the pumps will be tripped. Also, modify the test description and provide quantitative acceptance criteria for flow coastdown and trip of both the recirculation pumps. Also, provide stability criteria for plant performance following the trips.

(21) Test PT-31, Loss of Turbine-Generator and Offsite Power - Modify the test abstract to: (a) describe the initial plant conditions for the test, including the lineup of the plant's electrical system; (b) describe the type of trip to be conducted; (c) identify the variables, parameters, and plant equipment to be monitored; (d) provide assurance that test results will be compared with predicted results for the actual test case; (e) provide quantitative acceptance criteria and their bases for the required degree of convergence of actual test results with predicted results for the monitored variables and parameters; and (f) provide functional acceptance criteria for plant equipment that should function during or following the test. Also, correct the Level 1 acceptance criteria to be consistent with your facility design.

(22) Test PT-32, Containment Atmosphere Circulation System - Modify your acceptance criteria to include Level 1 criteria based on concrete temperatures.

(23) Test PT-35, Recirculation System Flow Calibration - Modify the test method to add calibrations at test conditions 2 and 5.

<u>RESPONSE</u>:

1) The design basis for the condensate demineralizers and RWCU are contained in the Water Quality Specifications which is part of the Level 1 Acceptance Criteria. See revised abstract for ST-1.

2) Acceptance Criteria based on ALARA objectives are included. See revised abstract for ST-2.

3) The value of R, an exposure dependent correction factor, and the predicted critical is contained in the Cycle Management Report, which is not yet available. The test abstract will be updated when the report becomes available.

Level 1 Criterion normally relate to the value of a process variable assigned in the design of the plant. Since the predicted critical is an expected value relating to the performance of the plant and not a design variable, it should remain a Level 2 Criterion.

4) The term "specified value" has been replaced by specific numerical values for those tests for which such values are available.

5) The formula used in ST-4 to determine the shutdown margin includes a moderator Temperature Coefficient to relate the shutdown margin at actual moderator temperature to the shutdown margin at a moderator temperature of 68°F upon which the Acceptance Criterion is based. See revised abstract for ST-4.

6) The withdraw speeds given in ST-5 are in agreement with figure 15-0-2 curve c. The withdraw speeds are considerably slower than the SCRAM rod speeds. SCRAM speed acceptance criteria are in accordance with the Technical Specifications. Control rod buffer performance is tested in P55.1.

7) Figure 14-2-5 has been revised to include testing at Test Condition 1, 3, 4 & 5.

8) The IRM SCRAM Trip point is verified and tested during the preop test program. ST-10 does not plan to perform a functional test of the IRM trip point. The IRM's are further tested during normal plant Surveillance Testing.

9) &
10) See revised abstracts for HPCI and RCIC testing.

11) The Level 1 Acceptance Criteria have been modified to include a pump in an idle loop. The Level 2 Acceptance Criterion has been transferred to ST-7 and now does relate to loop temperature.

12) The number of sets of data to be taken is described as a note to the acceptance criteria for ST-18.

13) ST-22 Pressure Regulator - See Figure 14.2-5 Sheet 1 for a description of the control mode of the recirculation system for this test. The feedwater control system will be in the mode suitable to operating plant conditions (typically 3-element Master Auto).

14) ST-23-Feedwater System-See revised abstract for ST-23. Testing for loss of feedwater heating in the manual flow control mode is a more severe transient than testing in the Auto mode as described in Section 15.1.1. Testing for loss of feedwater flow will cause a Recirculation System Runback. The effect is the same regardless of Master Manual or Master Automatic Flow Control. The loss of power to the extraction steam bleeder trip valves for one feedwater heater train results from failure of the electrical feed to the valves and is the event which is tested in ST-23.

We have reviewed possibilities for loss of all feedwater flow including pump or valve failures, feedwater controller failures, operator errors, and reactor system variables. Based on our evaluation, total loss of feedwater flow testing will not be included in ST-23.

Of the above mentioned failures, no single failure will cause loss of all feedwater. Pump or valve failures may reduce system capacity but will not result in loss of all feedwater. Feedwater controller failure will energize an annunciator circuit if the control signal to the RFP is lost. The alarm furnishes contacts which are utilized by the F.P. turbine speed control circuit to maintain turbine speed at the level existing at the time of signal loss. The feedwater control will transfer from 3 element control (level, steam flow, feedwater flow) to single element control on level.

Reactor variables such as water level will cause scram at (L-4) low water level, and alarm on (L-7) high water level, but will not cause loss of all feedwater before scram or operator initiated shutdown.

Loss of feedwater is considered in Section 15.2 which addresses increase in reactor pressure. Increase in reactor pressure start-up tests of similar intent but greater impact are performed as part of the start-up test program (e.g. ST-25 and ST-27 Turbine Trip and Generator Load Rejection) MSIV closure. The sequence of equipment response and operator action for ST-25 and ST-27 are identical to the loss of feedwater test. Reactor pressure and level instrumentation functional tests are being added to the pre-operation test program described in Section 14.2 to verify proper operation of this instrumentation.

15) See revised abstract for ST-25 MSIV's.

16) Bypass valve flow calibration, capacity and opening times are not tested during the Startup Test Program. Testing is done which verifies proper operation and reseating of each relief valve and verifies that no major blockages in the relief valve discharge piping exist. Opening times and capacity for the relief valves are tested at the factory and are not repeated. See revised abstract for ST-26.

17) See revised abstract for ST-27 for items (a) thru (e). The Initial Test Program is designed to demonstrate the performance of structures, systems, components, and design features that will be used during normal operations of the facility and also demonstrate the performance of standby systems and features that must function to maintain the plant in a safe condition in the event of malfunctions or accidents. The Susquehanna SES Initial Test Program does not include Acceptance Criteria for non-Susquehanna SES designed systems, such as the electrical grid system, whose performance is not under the control of the plant.

18) Plant's electrical system alignment is included in revised abstract for ST-28. The main objective of this test is to demonstrate that the reactor can be shutdown from outside the main control room. The performance of plant systems in response to transients and abnormal conditions is demonstrated in individual system's tests.

19) ST-29 - Recirc. Flow Control - The mode of control of the recirc. system is specified in Figure 14.2-5 Sht. 2. The feedwater control system will be in the mode of control which is specified by plant operating procedures for the various power levels.

20) Quantitative acceptance criteria for the flow coastdown after a two pump RPT trip will be included in the Transient Safety Analysis Design Report which is not yet available. The acceptance criteria will be revised when the information becomes available. Revised abstract for ST-30 describes the pump trips in more detail and provides acceptance criteria.

21) See revised abstract for ST-31 for items a, b, c and f. This test is performed at 30% to demonstrate the proper performance of the electrical distribution system and safety systems during a loss of the turbine-generator and offsite power. Predictions are made for the worst case transients rather than low power transients. The

proper performance of the plant to a turbine trip at 100% power is demonstrated in ST-27.

22) See revised abstract for ST-32. Acceptance criteria is based upon containment air temperatures not concrete temperatures.

23) Recirculation flow calibrations are done at Test Conditions 3 & 6 where flow is sufficient to provide meaningful flow data. Additional data at TC 2&5 would not provide any additional meaningful data.

QUESTION 423.21

You state in Subsection 14.2.4.6 that the completion of Phase II on safety-related systems is a prerequisite for commencement of the Power Test Program. Describe any preoperational tests shown in Tables 14.2-1 and 14.2-2 that you consider need not be completed prior to the commencement of the Power Test Program.

RESPONSE:

The tests listed in Table 14.2-1 are a pre-requisite to commencement of the start-up test program. The test results and exceptions to the tests will be evaluated, reviewed and approved per Subsection 14.2.5. The tests listed in Table 14.2-2 may be conducted on non-safety-related equipment. Table 14.2-2 is not a pre-requisite to commencement of the start-up test program.

QUESTION 423.22

Describe any preoperational and startup tests that you will conduct on Unit No. 1 that you may not conduct on Unit No. 2.

RESPONSE:

See the response to Question 423.34.

QUESTION 423.23

Provide a test description to provide for the integrated testing of reactor vessel isolation on low water level.

RESPONSE:

Testing of reactor water level instrumentation will be done during the technical test program. The test will verify level instrument response and setpoints. The actual operation of the various isolation valves are tested in their respective systems and in the containment system preoperational test P59.1. An abstract of preoperational test P59.1 is found in Section 14.2. A brief abstract of the level setpoint test TP2.14 is found following preoperational test P59.1.

QUESTION 423.24

Your answer to parts (a) and (b) of Question 423.1 regarding the qualification requirements for persons performing the functions of preoperational test directors and startup test directors are not satisfactory. We consider that the minimum qualifications for persons that direct or supervise the conduct of preoperational tests include a bachelor's degree in engineering or the physical sciences or the equivalent and one year of applicable power plant experience. Included in the one year of experience should be at least 3 months of indoctrination/training in nuclear power plant systems and component operation in a nuclear power plant that is substantially similar in design to the type at which the individual will perform the function. We consider that the minimum qualifications for persons that direct or supervise the conduct of individual startup tests should include a bachelor's degree in engineering or the physical sciences or the equivalent and two years of applicable power plant experience, at least one year of which should be applicable nuclear power plant experience. Revise your FSAR to indicate conformance to the staff position.

RESPONSE:

See revised response to Question 423.1.

QUESTION 423.25

The response to item 423.8 stated that FSAR Subsection 14.2.7 would be revised to show a 60 day period for NRC review of test procedures. The revision was made in 14.2.3, not 14.2.7. Correct the item 423.8 response.

RESPONSE:

See revised response to Question 423.8.

QUESTION 423.26

The response to item 423.10 is incomplete. Provide abstracts for the following tests: A84.1; A85.2; A87.1; A99.2; and A99.6.

RESPONSE:

See Subsection 14.2.12.3 for test abstracts A84.1, A85.2, A99.2 and A99.6. Test A87.1 has been incorporated into test A98.1.

QUESTION 423.27

Your response to item 423.11 states that current readings of containment recirculation fans will be higher during ILRT than at accident conditions. Provide technical justification for this statement. Address such issues as air density, temperature, humidity, fan speed and blade angle.

RESPONSE:

See revised response to Question 423.11.

QUESTION 423.28

The response to item 423.14 indicates that testing described in Regulatory Guide 1.80 sections C.7 through C.10 will not be done since the testing will have already been done during "various system preoperational tests". Either provide test descriptions that show testing equivalent to that specified in regulatory positions C.8, C.9, and C.10 will be performed, or modify your preoperational test program to include an integrated loss of air test and provide an abstract of that test.

RESPONSE:

See revised response to Question 423.12.

Table 423.28-1 lists all operator valves/HVAC dampers which a re tested for loss of air Preoperational tests within which the loss of air testing is accomplished is also provided in Table 423.28-1.

Further testing is performed for the ADS/SRV valves as follows:

1. Verify minimum capacity of accumulator in acceptance criteria.

2. Verify ADS/SRV's are operated from their respective accumulator/supply with other supplies depressurized.

3. Record pressure at which an open valve begins to close for safety/relief valves and verify valve fails to closed on loss of air.

4. Verify an open ADS valve is maintained open at accumulator pressure of 75 + 0 - 2 PSIG and fails closed on loss of air.

TABLE 423.28-1

SYSTEM	VALVE NUMBER	PREOPERATIONAL NO.	INSTRUMENT AIR OR PRIMARY CONTAINMENT INSTRUMENT GAS
RHR	1-E11-F050A,B 1-E11-F122A,B 1-E11-F051A,B 1-E11-F052A,B 1-E11-F053A,B 1-E11-F111A,B 1-E11-F129A,B 1-E11-F132A,B 1-E11-F136, F137, F140	P49.1	Instrument Gas Instrument Gas Instrument Air
RCIC	HV-E51-1F008 HV-E51-1F025, 1F026 HV-E51-1F004, 1F005 HV-E51-1F054	P50.1	Instrument Gas Instrument Air
Core Spray	HV-E21-1F006A,B HV-E21-1F037A,B	P51.1	Instrument Gas Instrument Gas
HPCI	HV-E41-1F028,1F029 HV-E41-1F025,1F026 LV-E41-1F054,1F100	P52.1	Instrument Air 1F100 Gas Others Instrument Air
CRD	C12-F002A,B XV-1F010,1F011	P55.1	Instrument Air
RECIRC	HV-B31-1F019,1F020 Both*		F019-Instrument Gas F100-Instrument Air
Fire Protection	XV-12248,49 XV-02248 XV-02215	P13	Instrument Air
RBCCW	HV-11315	P14	Instrument Air

TABLE 423.28-1 (Continued)

SYSTEM	VALVE NUMBER		PREOPERATIONAL NO.	INSTRUMENT AIR OR PRIMARY CONTAINMENT INSTRUMENT GAS
RB HVAC	HD17534A,B,C,D,E,F,H HD17502A,B; HD17514A,B HD17564A,B; HD17524A,B HD17576A,B; HD17586A,B HD17508A,B HD17651	All* All* All All* Both*	P34.1	Instrument Air
RWCU	HV-14506A,B; 14507A,B HV-14508A,B; 14510A,B HV-14511A,B; 14512A,B HV-14513A,B; 14514A,B HV-14566A,B; 14522 HV-14523, 14528, 14516 HV-14518, 14519, 14520 HV-14521, G33-1F033		P61.1	Instrument Air
Liquid Radwaste	HV-16108A1, HV-16116A1 HV-16108A2, HV-16116A2	 Both*	P69.1	Instrument Air
Containment Recirculation	HV-15721, 23, 24, 22, 25 HV-15704, 05, 14 HV-15703, 13 HV-15711	All* All*	P73.1	Instrument Air
R.B. HVAC	PDD17501A,B; HD17511A,B HD17521A,B; HD17513A,B HD17518A,B; HD17516 HD17523A,B; HD17528A,B PDD17578A,B; HD17526 HD17566A,B; HD17588A,B HD17538A,B		P34.1	Instrument Air

TABLE 423.28-1 (Continued)

SYSTEM	VALVE NUMBER		PREOPERATIONAL NO.	INSTRUMENT AIR OR PRIMARY CONTAINMENT INSTRUMENT GAS
RB Chilled Water	TV-18726A1,A2,B1,B2		P34.2	Instrument Air
	TV-18741A,B,C,D			
	TV-18743A,B			
	TV-18751A,B,C,D			
	TV-18753A,B			
	TV-18764A,B			
	FV-18771A,B,C,D			
	HV-18781A1,A2,B1,B2	All*		Instrument Gas
	HV-18782A1,A2,B1,B2	All*		
	HV-18791A1,A2,B1,B2	All*		
	HV-18792A1,A2,B1,B2	All*		
Control Structure HVAC	HDM-07802A,B	Both*	P30.1	Instrument Air
	HDM-07833A,B; HDM-07824A2,B2; HDM-07824A1,B1			
	HDM-07824A4,B4; HDM-07881A,B			
	HDM-07872A,B; HDM-07873A,B	All*		
	TV-07813A,B			
	TV-08602A,B		P30.2	
Feedwater	FV-10604A,B,C; HV-10640; LV-10641		P45.1	Instrument Air
	HV-14107A,B; HV-10650			
	HV-10606A,B,C			
	TV-10663A1,A2,B1,B2,C1,C2			
	LV-10664A,B,C			

QUESTION 423.29

The response to item 423.15 is not complete. It is the staff's position that you (1) provide quantitative acceptance criteria for drywell floor bypass leakage and (2) modify your Reactor Protection System test to account for the delay time of interfacing hardware (e.g., sensing lines) on channel response time.

RESPONSE:

(1) The quantitative acceptance criteria for drywell floor bypass leakage is 176.4 scfm at a differential pressure of 4.3 psid.

(2) Per the NRC Standard Technical Specification (NUREG 0123), the Reactor Protection Response Time shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energized of the scram pilot valve solenoids. Therefore, no modification for the RPS is required. No account is made in the RPS test for delay times in sensing lines for pressure as delay time contribution to channel response in negligible.

QUESTION 423.30

Modify P51.1 to make it consistent with the response to item 423.16.

RESPONSE:

See revised response to Question 423.16, Preoperational Test P51.1 and new Preoperational Test P51.1A.

QUESTION 423.31

The response to item 423.17 states that Unit 1 and Unit 2 preoperational testing on the 4.16 kV system (P4.1) will be accomplished jointly in one month commencing 14 months prior to fuel load on Unit 1 (Figure 14.2 - 4a) and 11 months prior to fuel load on Unit 2 (Figure 14.2 - 4b). Section 14.2.11 states that because "the initial fuel loading of Unit 2 is scheduled to occur 18 months after Unit 1, the test programs will not overlap." Modify Chapter 14 and the response to item 423.17 as necessary to correct this discrepancy. In addition it will be necessary for you to provide the information requested in item 423.17 (i.e., electrical lineups) in enough detail for us to determine the following:

(1) That during the Regulatory Guide 1.41 testing on each unit, there will be no crossties from the other unit's electrical system that could compromise the validity of the test results.

(2) That if Unit 1 is licensed at the time the Unit 2 test is performed, there will be no crossties that could cause a loss of power to Unit 1 emergency bus.

RESPONSE:

As stated in the response to Question 423.17, the preoperational test encompassing the 4.16 kV systems for both units (P4.1) will be completed prior to fuel load of Unit 1.

4.16 kV preoperational testing to be performed prior to Unit 1 fuel load will include ES Transformers 101 and 201 and ES busses 1C, 1D, 1E, 1F, 2C, 2D, 2E and 2F including all feeder breakers. Since the 4.16 kV system is common to both units up to the feeder breakers for the unit ES busses, discussion of crossties affecting the validity of testing is not pertinent. The isolation points between Unit 1 and common and untested Unit 2 equipment are the feeder breakers for the Unit 2 ES busses. During testing of Unit 2 equipment with Unit 1 in operation, these feeder breakers protect against Unit 1 EA bus power failures originating in Unit 2.

13.8 kV preoperational testing to be performed prior to Unit 1 fuel load will include Startup Transformers 10 and 20, Startup busses 10 and 20 and Auxiliary busses 1A, 1B, 2A and 2B including all feeder breakers except those from the Unit 2 Auxiliary Transformer. Since 13.8 kV system is common to both units, discussion of crossties affecting the validity of testing is not pertinent. The isolation points between Unit 1 and common and untested Unit 2 equipment are the feeder breakers at auxiliary busses 2A and 2B and Startup bus 20. During testing of Unit 2 equipment with Unit 1 in operation, these feeder breakers protect against Unit 1 power failures originating in Unit 2.

Figures 14.2-4a and 14.2-4b show typical preoperational test schedules and are not intended to be updated.

QUESTION 423.32

The response to item 423.19 states that: "Testing to verify that ESF pumps operate within their design pump head curves and with adequate NPSH will be done. This testing is committed to in the General Test Statement as part of the answer to Question 423.12." The general test statement says this will be done, but only "where possible". Modify the response to indicate that all ESP pumps will be completely tested.

RESPONSE:

The response to Question 423.12 has been revised to specify the scope of pump testing to be performed during the preoperational test program. The testing to be performed on the HPCI pump is described in the ST-15 abstract. Insufficient auxiliary steam capacity precludes preoperational HPCI testing. This also applies to the RCIC system.

QUESTION 423.33

The response to item 423.20 indicates that certain changes will be made to the initial test program. Some of these changes have not yet been reflected in Chapter 14.

1. Modify Figure 14.2-5 as stated in sub-item 7.

2. Revise the abstracts for HPCI and RCIC tests to include the demonstration of several successful cold starts as stated in sub-items 9 and 10.

3. Modify the PT-26 abstract to state that a review of factory test results (flow and opening times) is conducted as part of the overall test review program as described in the response to sub-item 16.

RESPONSE:

See revised Subsection 14.2.12 and Question 423.20.

QUESTION 423.34

Modify Figure 14.2-4b to make it consistent with the response to item 423.22. (Add or correct Tests P70.1, P30.2, P88.1, and P28.1.

RESPONSE:

As stated in the response to Question 423.31, Figure 14.2-4b is a typical preoperational test sequence. The following information reflects the testing to be performed:

- P30.2 and P28.1 will be performed on Unit 1 only.
- P88.1 will be performed on Unit 1 and Unit 2
- P70.1 will consist only of a negative leak rate test on Unit 2.

QUESTION 423.35

Expand or explain the following terms:

"Interlocks the RFPT alternate . . ." (P45.1)

"high-high temperature . . ." (A30.3)

RESPONSE:

(1) See revised acceptance criteria for P45.1

(2) For charcoal filters, high temperature actuates an alarm and high-high temperature actuates an alarm and the fire protection system.

QUESTION 423.36

Include testing of the communications system in the preoperational tests or provide assurance that the test procedure and results will be reviewed in a manner similar to the preoperational tests.

RESPONSE:

This acceptance test (A99.2) and its results will be reviewed in a manner similar to the preoperational tests.

QUESTION 423.37

The exception to Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units as Onsite Electric Power Systems at Nuclear Power Plants" (Revision 1), concerning the number of necessary consecutive valid tests per diesel is not acceptable. It is the staff's position that you perform the $\frac{69}{n}$ starts in accordance with Regulatory Position 2.a(9). Modify Subsection 14.2.7 to state that your test will be conducted in accordance with this position or provide a description of tests that you will perform to demonstrate the required reliability.

RESPONSE:

See revised Subsection 14.2.7.

QUESTION 423.38

Include Regulatory Guide 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants" (Revision 1), in Subsection 14.2.7. Provide justification for any exceptions to Regulatory Positions C.5 and C.6.

RESPONSE:

See revised Subsection 14.2.7.

QUESTION 423.39

Revise Table of Contents listing of Tables and Figures to reflect the current status of FSAR Section 14.

RESPONSE:

See revised Table of Contents.

QUESTION 423.40

Our review of recent licensee event reports disclosed that a significant number of reported events concerned the operability of hydraulic and mechanical snubbers. Provide a description of the inspections or tests that will be performed following system operation to assure that the snubbers are operable. These inspections or tests should be performed preoperationally if system operation can be accomplished prior to generation of nuclear heat.

RESPONSE:

Existing QA records on the construction installation and inspection of safety-related snubbers will be assembled into a package for review by the Superintendent of Plant. This package will provide assurance that the preoperational condition of the snubbers is acceptable and that they are installed in accordance with design.

After system preoperational testing and prior to fuel load, snubbers will be visually examined and manually tested for freedom of movement over the range of stroke in both compression and tension. This meets the requirement of IE Bulletin 81-01 Rev. 1. No hydraulic snubbers are utilized in safety applications at Susquehanna SES.

QUESTION 423.41

Revise acceptance test A39.1 (Condensate demineralizer Abstracts) to correct the following inconsistencies:

1) State whether the system will process water at 120% above rated capacity (Test Method) or at 120% of rated flow (Acceptance Criteria).

2) Ensure that monitored conditions are at least held at design specifications (Test Method).

RESPONSE:

See revised acceptance test A39.1.

QUESTION 423.42

Clarify the first acceptance criterion in P45.1 (Feedwater System Preoperational Test).

RESPONSE:

See revised preoperational test P45.1.

QUESTION 423.43

Modify the test method of P99.1 (Reactor Building Crane Preoperational Test) so that operation is completely checked in both directions vice "either direction" as stated.

RESPONSE:

See revised preoperational test P99.1.

QUESTION 423.44

Modify the Figure 14.2-3 references to refer to the proper figures.

RESPONSE:

See revised Figure 14.2-3 references.

QUESTION 423.45

(1) The response to item 423.12 is not completely acceptable. Several acceptance test abstracts (A3.1, A13.1-A13.4, A15.1, A41.1, A45.1, A45.2, and A681.) are labeled as preoperational tests. Correct these inconsistencies

(2) The response to several sub-items (i.e., 1.i.2, 5.+, 5.u) does not address valve closure times. Modify the response to address them or provide technical justification for the deletions.

(3) If the factory testing of a component substitutes for inplant testing, then: 1) the method of testing, 2) the results of the testing, and 3) how these results are extrapolated to actual plant conditions should be reviewed and retained. Modify your response to provide commitment.

RESPONSE:

(1) Tests 13.1 through 13.4, 45.1 and 45.2 are preoperational tests. Tests 3.1, 15.1, 41.1 and 68.1 are acceptance tests. See also the revised response to Question 423.12.

(2) See revised response to Question 423.12 and preoperational test P59.1.

(3) See revised response to item 5.t of Question 423.12.

QUESTION 423.46

Revise the test method of A93.1 (Turbine Lube Oil System) so that it indicates the actual test method.

RESPONSE:

See revised acceptance test A93.1.

QUESTION 423.47

Explain the status of ST-84 (RPV Internals Vibration). It has been deleted as a startup test in section 14.2.12.2, is included as a startup test in figure 14.2-5 sheet 3, is also included in Section 14.2.12.1 with preoperational test abstracts, and yet is not included in Table 14.2-2 or Figure 14.2-4. Revise the applicable sections to address the internals vibration tests.

RESPONSE:

Reactor internals are tested in accordance with provisions of Regulatory Guide 1.20, Revision 2, for Non-prototype category I plants, as described in FSAR section 3.9.2.4. This testing is performed prior to fuel load in TP 2.16, "Reactor Internals Vibration and Inspection," an abstract of which is included in Section 14.2.12.1 as requested in an earlier item. The Startup Test "RPV Internals Vibration" was deleted since the testing is not repeated after fuel load. The number 34 was subsequently reassigned to the "Rod Sequence Exchange" Startup Test.

QUESTION 423.48

Your response to item 423.22 states that P30.2 and P28.1 are only test that will be conducted on Unit 1, and not on Unit 2. Modify test descriptions for P13.1 and ST-31 to indicate that testing will be accomplished on both units, or modify your response to item 423.22 to justify not conducting P13.1 and ST-31 on Unit 2.

RESPONSE:

See the revised abstract for ST-31 for testing of Unit 1 and Unit 2 on loss of turbine-generator and offsite power.

The abstract for P13.1 has been revised to discuss the reduced scope of testing to be performed on Unit 2 (deluge systems; dry pipe, wet pipe and preaction systems; hoses in Unit 2 areas).

QUESTION 423.49

Your response to several subitems of 423.12 are not acceptable. Provide the requested information:

(1) 64.79 - Modify preoperational test descriptions P81.1 and P99.1 to demonstrate that the refueling grapple and reactor building crane are statically tested at 125% rated load and dynamically tested at 100% rated load.

(2) 77.99 - Provide a startup test description that will demonstrate that concrete temperatures surrounding hot penetrations do not exceed design limits.

RESPONSE:

(1) The reactor building crane was tested at 125% of capacity by the vendor. Testing was performed on site by construction forces under the vendor's direction. Prerequisites to P99.1 require verification of the 125% test documentation. Testing at 100% of rated capacity is accomplished during the preoperational test program by TP2.23. An abstract of TP2.23 follows P99.1.

The refueling bridge main hoist (1200 pound capacity) will be tested to 125% of capacity utilizing a Technical Procedure. Preoperational Test P81.1 provides for load limit interlock testing and functional testing utilizing a dummy fuel assembly. The weight of the dummy fuel assembly and the grapple is approximately 950 pounds.

(2) The design of hot penetrations includes insulation on the exterior of the process pipe and an air gap between the inside surface of the penetration and outer surface of the pipe insulation. Analytical calculations have been performed to provide assurance that the present Susquehanna SES design of the hot penetrations will be able to maintain the concrete temperatures around these penetrations below the design limit ST-32, "Containment Atmosphere and Main Steam Tunnel Cooling," demonstrates that the temperature of the atmosphere inside the drywell is maintained within design limits.

With the reactor at rated temp. during the drywell inspection (described in ST-17) a check will be made to estimate the concrete temp. surrounding one of the main steamline penetrations by measuring the temperature at several accessible points on the containment liner plate or containment concrete surface.

QUESTION 423.50

Provide testing to verify that containment spray nozzles and headers, are free of debris by testing. If this testing is not performed with worker in conjunction with testing the pumps, verify that the flow path for this testing overlaps the flow path used when testing the pumps.

RESPONSE:

P49.1 (RHR System Preoperational Test) provides for testing of the containment drywell spray nozzles. This test consists of connecting a streamer to each spray nozzle and connecting a source of service air to the system and verifying that the nozzles are not plugged by observing air flow and streamer movement.

The containment wetwell spray nozzle test consists of directing RHR system water through the containment wetwell spray header and verifying that each nozzle is not plugged and is spraying water. System flow through the containment spray header was verified during TP 3.25 (RHR System flush) by connecting hoses between the two loop spray headers and flushing from one loop into the other and back to the suppression pool. Bench testing of a similar drywell spray nozzle has been accomplished in the factory as described in FSAR Subsection 6.2.2.2.

QUESTION 423.51

Provide or modify test descriptions that will verify that the emergency ventilation systems are capable of maintaining all ESF equipment within their design temperature range with the equipment operating in a manner that will produce the maximum heat load in the compartment. If it is not practical to produce maximum heat loads in a compartment, describe the methods that will be used to verify design heat removal capability of the emergency ventilation systems.

Note that it is not apparent that post-accident design heat loads will be produced in ESF equipment rooms during the power ascension test phase; therefore, simply assuring that area temperatures remain within design limits during this period may not, in itself, demonstrate the design heat removal capability of these systems. It may be necessary to measure air and cooling water temperatures and flows and to extrapolate to verify that the ventilation systems can remove the postulated post-accident heat loads.

RESPONSE:

ESF equipment room coolers, Heat exchangers 1E230B, 1E230D, 1E217B, 1E217D, 1E218B, 1E218D, 1E257B, 1E231B, 1E231D, 1E229A and 1E229B were performance tested by the vendor to demonstrate conformance to design specifications. During the preoperational test program, the ESF equipment room coolers air flow and cooling water flows will be measured as part of hydronic balancing and air balancing procedures. These balancing procedures provides a comparison of design values and actual values for the heat load encountered. On the basis of meeting the design specification for heat removal the procedures will validate the vendor performance tests for design maximum heat removal.

QUESTION 423.52

Modify ST-30 to indicate that a simultaneous trip of both recirculation pumps will be performed at test condition 6 or provide technical justification in Subsection 14.2.7 for taking exception to Regulatory Guide 1.68 (revision 1, 1/77), Appendix A, 5.1.1.

RESPONSE:

On earlier plants, where MCHFR was used to determine reactor thermal margin, the two pump trip was performed since MCHFR was very sensitive to core flow. When GE developed the GEXL correlation, which establishes MCPR for determining reactor thermal margin for current plants, it was found that MCPR is relatively insensitive to core flow. When the effect of the two pump trip on the reactor thermal margin was determined to be minor, the test was generically deleted from BWR Startup Test Programs.

At Susquehanna, the two pump trip is done at Test Condition 3 (approximately 100% core flow and 75% power) not to determine the effects of core flow upon MCPR but to verify acceptable performance of the recirculation two pump circuit trip system and to demonstrate acceptable pump coastdown performance prior to high power turbine trips and generator load rejects.

QUESTION 423.53

Modify ST-31 to provide assurance that the loss of offsite power condition will be maintained for at least 30 minutes to demonstrate that necessary equipment, controls, and indication are available following station blackout to remove decay heat from the core using only emergency power supplies and distribution systems.

RESPONSE:

Test description for ST-31 has been modified to maintain the loss of offsite power condition for at least 30 minutes. See revised abstract for ST-31.

QUESTION 423.54

Include the test description (TP2.14) provided as a response to item 423.23 in the FSAR, Subsection 14.2.12.

RESPONSE:

Test description for TP2.14 has been removed from Question 423.23 and placed in FSAR Subsection 14.2.12 for the test abstract of Preoperational Test P59.1.

QUESTION 423.55

Revise Subsection 14.2.12 to incorporate responses to items 423.37 and 423.38.

RESPONSE:

Subsection 14.2.12 has been revised to incorporate the responses to Questions 423.37 and 423.38.

QUESTION 423.56

Your responses to items 423.32 and 423.45 reference a revised response to item 423.12. Provide this revised response, or revise your response to items 423.32 and 423.45 to provide the requested information.

RESPONSE:

Question 423.12 has been revised to incorporate the responses to Questions 423.32 and 423.45.

QUESTION 423.57

Modify ST-25 to address the following:

(1) The present method for determining MSIV closure times is inaccurate. Modify the test method to measure the full travel of the valves or provide technical justification for extrapolating the full closure time when only measuring 90 percent closed, plus the period from 10 percent closed to 90 percent closed times 1/8, or provide technical justification for the current method which "double-counts" delay time.

(2) Provide a description of a test which demonstrates that the MSIV-LCS components operate properly when handling steam and that the system can handle the amount of leakage that is present when the main steam system is at operating temperature.

RESPONSE:

(1) ST-25 provides for determination of MSIV closure times as described below:

MSIV closure time must meet divergent criteria. The valves must close fast enough to limit the release of reactor coolant, and they must close slow enough so that simultaneous closure of all steamlines will not induce transients that exceed the nuclear steam design limits. MSIV closure time is calculated using limit switches which actuate when valve stem travel indicates 10% and 90% valve closure. Extrapolations using this data assumes linear valve closure.

Two equations are necessary to accurately calculate elapsed times. The slow criteria equation must include the delay time from solenoid deenergization to valve stem movement, whereas the fast criteria equation, which is concerned only with valve movement, does not include this delay. The two equations are:

(1) for fast criteria

$$T_c = (T_{90} - T_{10}) + 0.25\,(T_{90} - T_{10})$$

$$T_c = 1.25\,(T_{90} - T_{10})$$

(2) for slow criteria

$$T_{cwd} = T_{90} + 0.1\,T_c$$

$$T_{cwd} = T_{90} + 0.125\,(T_{90} - T_{10})$$

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where:

T_c = valve closure time, excluding delays

T_{cwd} = valve closure time, with delays

T_{90} = elapsed time from solenoid deenergization to valve 90% closed

T_{10} = elapsed time from solenoid deenergization to valve 10% closed.

Start Historical Section

(2) The MSIV-LCS is designed to control and minimize the release of fission products which could leak through closed MISV's following a LOCA. The MSIV-LCS is initially verified operable in P83.1 using air and subsequently verified operable on a periodic basis in accordance with Technical Specifications. Complete system testing and isolation valve leak testing is performed only during reactor shutdown to preclude inadvertent steam discharge. Interlocks are provided to preclude system operation at excessive MSIV leak rates. No further Startup Testing is deemed necessary.

End Historical Section

QUESTION 423.58

Update Table 14.2-3 (Startup Test Procedures) and Figure 14.2-5 (Individual Startup Test Sequence) to reflect the current status of Subsection 14.2.12.2.

RESPONSE:

Table 14.2-3 and Figure 14.2-5 has been revised to reflect the current status of Subsection 14.2.12.2.

QUESTION 440.1

Provide the fire protection program information in accordance with the guidelines of SRP Section 13.2 and 13.5 Revision 1.

RESPONSE:

Please refer to revised FSAR Subsections 13.2.1.1.7, 13.2.1.1.7.1, 13.2.1.1.7.2 and 13.5.2.2.18.

QUESTION 441.1

FSAR Figure 13.2-1, Revision 1, shows that 16 people will be licensed prior to fuel loading. Provide the number of people who will be trained in your licensed operator training program. This number should not only meet Technical Specification requirements but should also allow for examination contingencies and avoidance of planned overtime during the startup phase. We recommend the training of at least 25 people.

RESPONSE:

Figure 13.2-1 shows the minimum numbers in each position. The cold license training program has been layed out to handle 45 license candidates. Training is not planned on an overtime basis.

QUESTION 441.2

What are your plans for additional training in the event that fuel loading is substantially delayed? An acceptable method of maintaining the required level of training if fuel loading is significantly delayed would be to initiate the requalification program.

RESPONSE:

Requalification training is intended to commence 4 1/2 months after the completion of the cold license training program as presently scheduled. In the event fuel loading is delayed significantly, one alternative would be to move the requalification program up to commence within 3 months of the end of the cold license training program, or reschedule a Condensed Form or Refresher Training.

QUESTION 441.3

FSAR Figure 13.2-1, Revision 1, indicates that a Pre-license Refresher Course will be conducted six months prior to fuel loading. We consider it highly desirable that license applicants participate in a short simulator course immediately prior to the examinations. Is it PP&L's intention to provide such a course?

RESPONSE:

The Refresher Course commences 5 months prior to license examinations and is composed of a 4 week session and 3 week session. The license candidates will be divided into 6 groups; the first group starts the 4 week session 5 months prior to examination with the 6th group finishing the 4 week session 2 months prior to examination. The 6 groups then rotate through the 3 week session such that the longest any group will be away from the Simulator will be 6 weeks before the examination is administered.

QUESTION 441.4

State the methods used to evaluate the training program effectiveness for Phase I, Phase II, Phase V, and Phase VI training.

RESPONSE:

The effectiveness of Phase I thru Phase V has been documented thoroughly via written exam, as well as, formal certification exams administered at the conclusion of vendor supplied courses approved by the NRC.

The performance of the license candidates in succeeding courses further attest to the effectiveness of prior training.

Phase VI training will be evaluated by both oral and written exams, as well as demonstrated on the job performance.

QUESTION 441.5

The BWR simulator course is taught at the General Electric Training Center in Morris, Illinois. Our position is that individuals seeking licenses for the Susquehanna Plant will have to participate in training programs that utilize a Nuclenet simulator, if such a simulator is operational. Provide a commitment that simulator training will be conducted at a Nuclenet simulator, if operational prior to fuel loading, and identify the simulator to be used for this training.

RESPONSE:

It is PP&L's position that training for NRC license candidates will be conducted on the Susquehanna BWR Simulator which uses computer generated graphic displays in the control room if it is available prior to fuel loading. It is expected that the Susquehanna Simulator will be ready-for-training during the Refresher Training Program and thus will provide the necessary operator familiarization.

QUESTION 441.6

The Susquehanna Fire Safety Training program is unacceptable. Provide a detailed description of the fire protection training and retraining for the critical plant staff and replacement personnel which meets the following acceptance criteria:

A. Fire Brigade Training

(1) Instruction

(a) Instruction in all the topics listed in (d) below should be administered to individuals prior to assignment as a fire brigade member.

(b) Refresher instruction should be provided to all fire brigade members on a regularly scheduled basis of not less than four sessions a year. The sessions shall be repeated at a frequency of not more than 2 years.

(c) The instruction shall be provided by qualified individuals, knowledgeable and experienced in fighting the types of fires that could occur in the plant and in using the types of equipment available in a nuclear power plant. Members of the Fire Protection Staff and fire brigade leaders may also conduct this training.

(d) The scope of the instruction should include the following items:

(i) An identification of the fire hazards and associated types of fires that could occur in the plant, and an identification of the location of the hazards, including areas where breathing apparatus is required, regardless of the size of the fire.

(ii) Identification of the location of installed and portable fire fighting equipment in each area, and familiarization with layout of the plant including access and egress routes to each area.

(iii) The proper use of available equipment, and the correct method of fighting each type of fire. The types of fires covered should include electrical fires, fires in cables and cable trays, hydrogen fires, flammable liquids, waste/debris fires, and record file fires.

(iv) Indoctrination in the plant fire fighting plan, with coverage of each individual's responsibilities, including changes thereto.

(v) The proper use of breathing equipment, communication, lighting and portable ventilation equipment.

(vi) A detailed review of the procedures with particular emphasis on what equipment must be used in particular areas.

(vii) A review of latest modifications, additions or changes to the facility, procedures, fire fighting equipment or fire fighting plan.

(viii) The proper method of fighting fires inside building and tunnels.

(ix) In addition, special instruction should be provided for fire brigade leaders in directing and coordinating fire fighting activities.

(2) Practice

Practice sessions should be held for fire brigade members on the proper method of fighting various type of fires. These sessions should provide brigade members with practice in extinguishing actual fires, except in the case of energized cables. Practice sessions should be conducted at facilities sufficiently remote from the nuclear power plant so as not to endanger safety-related equipment. These practice sessions should be provided at regular intervals, but not to exceed one (1) year.

Practice sessions should also be conducted that require the brigade members to don protective equipment, including emergency breathing apparatus. These practice sessions need not include fire fighting. These practice sessions should be provided at regular intervals, but not to exceed one (1) year.

(3) Drills

Fire brigade drills should be performed in the plant so that a fire brigade can practice as a team. Drills should include the following.

(a) The simulated use of equipment for the various situations and types of fires which could reasonably occur in each safety-related area.

(b) Conformance, where possible, to the established plant fire fighting plans.

(c) Operating fire fighting equipment where practical. This would also include self-contained breathing apparatus, communication equipment and portable and/or installed ventilation equipment.

(d) The drills should be performed at regular intervals, but not to exceed three months for each fire brigade. The minimum number of fire brigade drills conducted within a period of three months shall be equal to the number of operating shifts at the station. Each individual member of the fire brigades shall participate in at least two drills per year. At least one drill per year for each operating shift shall be unannounced.

(e) Periodically (at least annually), these drills should include off-site fire department personnel. These drills should also conform with the facility plan for coordination with off-site fire departments.

(f) The drills should be preplanned to establish the training objectives of the drills. The drills should be critiqued to determine how well the training objectives have been met. At a minimum, the critique should assess:

(i) Fire alarm effectiveness, response time, selection, placement and use of equipment.

(ii) The leader's direction of the effort and each member's response.

B. Other Station Employees

(1) Instruction

(a) Instruction shall be provided for all employees. It shall be repeated on an annual basis. The instruction shall be given, as appropriate, on (i) the fire protection plan (ii) evacuation routes and (iii) procedure for reporting a fire.

(b) Instruction shall be provided for security personnel that addresses (i) entry procedures for outside fire departments (ii) crowd control for people exiting the station, and (iii) procedures for reporting potential fire hazards observed when touring the facility.

(c) Instruction should be provided to all shift personnel that complements that provided members of the fire brigade.

(d) Instruction shall be provided to a temporary employees, so that they are familiar with (i) evacuation signals, (ii) evacuation routes and (iii) procedure for reporting fires.

(2) Drills

All employees should participate in an annual evacuation drill.

C. Fire Protection Staff

All employees should participate in an annual evacuation drill.

(1) design and maintenance of fire detection, suppression and extinguishing systems.

(2) fire prevention techniques and procedures,

(3) training and manual fire fighting techniques and procedures for plant personnel and the fire brigade.

D. Off-Site Fire Departments

Training for the off-site fire departments include courses in basic radiation principles and practices, typical radiation hazards that may be encountered when fighting fires and related procedures.

E. Construction Personnel

Training for construction personnel clearance should include instruction in reporting fires, alarm responses and evacuation routes.

<u>RESPONSE:</u>

The response to this question is provided in Section 1.4 of the Susquehanna SES Fire Protection Review Report.

QUESTION 441.7

Revise paragraph 13.2.2.1.3 in the FSAR to indicate the following:

(1) An individual who prepares, administers, or grades a written examination need not take the examination. a maximum of three licensed personnel may be exempted under this condition.

(2) Retraining lectures may use training aids such as video tapes and films in lieu of an instructor. However, no more than 50% of the lectures may be supplemented by use of training aids.

RESPONSE:

FSAR subsection has been revised accordingly.

QUESTION 441.8

Oral exams are acceptable for determining whether or not an individual resumes licensed duties after receiving a grade of 70% or less on an annual exam. However, the individual must remain on a Performance Review Program until a grade of 70% or better is obtained on a written examination.

RESPONSE:

A revision to FSAR subsection 13.2.2.1.4 has been made to reflect the necessity of passing an annual written examination with a grade >70% prior to being removed from the PERFORMANCE Review Program.

QUESTION 441.9

As a minimum, refresher instruction on administrative, radiation protection, emergency, and security procedures should be provided to all non-licensed personnel.

RESPONSE:

Subsection 13.2.2.2 of the FSAR has been revised to reflect the desirability of providing refresher instruction or administrative, radiation protection, emergency, and security procedures to all non-licensed personnel.

QUESTION 441.10

The Susquehanna Fire Safety Training program remains unacceptable. The PP&L response meets some, but not all of the acceptance criteria required by Question 441.6. As an example, using the numbering scheme of Question 441.6, the NRC acceptance criteria and the NRC position is as follows:

NRC Acceptance Criteria A.1.d(i)

An identification of the fire hazards and associated types of fires that could occur in the plant, and an identification of the location of the hazards, including areas where breathing apparatus is required, regardless of the size of the fire.

NRC Position A.(1)a.

Identification of fire hazards.

Many other instances of oversimplification and material deletion exist in the applicants response. PP&L is required to meet the acceptance criteria of first-Round Question 441.6 or provide adequate justification for deviations. Additionally, the applicant must include the response to this question as part of FSAR Section 13.2. Also, PP&L must commit to not using Shift Supervisor as the individual responsible for directing the actual fire fighting forces.

RESPONSE:

Please see the revised response to Question 441.6.

QUESTION 441.11

SSES must commit to following the guidance of Section 5.5.2 of ANSI N18.1-1971 in the training conducted for replacement personnel.

RESPONSE:

Replacement Training programs have been defined as being the individual's Cost Area Head's responsibility per Nuclear Department Instruction (NDI) QA 4.1.2. The requirements of this NDI include the requirements of ANSI 18.1-1971.

QUESTION 441.12

The staff requires that the applicant develop a program to ensure that all operating personnel are trained in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged. The training program shall be completed prior to Full Power Operation and shall include the following topics:

(1) Incore Instrumentation

(a) Use of fixed or movable incore detectors to determine extent of core damage and geometry changes.

(b) Use of thermocouples in determining peak temperatures; methods for extended range readings; methods for direct readings at terminal junctions.

(2) Excore Nuclear Instrumentation (NIS)

(a) Use of NIS for determination of void information; void location basis for NIS response as a function of core temperatures and density changes.

(3) Vital Instrumentation

(a) Instrumentation response in an accident environment; failure sequence (time to failure, method of failure); indication reliability (actual vs. indicated level),

(b) Alternative methods for measuring flows, pressures, levels, and temperatures.

(i) Determination of pressurizer level if all level transmitters fail.

(ii) Determination of letdown flow with a clogged filter (low flow)

(iii) Determination of other Reactor Coolant System parameters if the primary method of measurement has failed.

(4) Primary Chemistry

(a) Expected chemistry results with severe core damage; consequences of transferring small quantities of liquid outside containment; importance of using leak tight systems.

(b) Expected isotopic breakdown for core damage; for clad damage.

(c) Corrosion effects of extended immersion in primary water; time to failure.

(5) Radiation Monitoring

(a) Response to Process and Area Monitors to severe damages; behavior of detectors when saturated; method for detecting radiation readings by direct measurement at detector output (overanged detector); expected accuracy of detectors at different locations; use of detectors to determine extent of core damage.

(b) Methods of determining dose rate inside containment for measurements taken outside containment.

(6) Gas Generation

(a) Methods of H_2 generation during an accident; other sources of gas (Xe, Kr); techniques for venting or disposal of non-condensables.

(b) H_2 flammability and explosive limit, sources of O_2 in containment or Reactor Coolant System.

The operating personnel who receive this training must include the Station Superintendent, his assistant, Shift Technical Advisors, licensed operators, and all other members of the operating staff whose skills would be utilized during degraded core conditions. Chemistry, Health Physics, and ICS personnel should receive training in those areas applicable to their duties.

RESPONSE:

The requested information was provided in response to TMI related requirements transmitted to the NRC on January 22, 1981. Refer to PLA-614, N. W. Curtis to B. J. Youngblood.

QUESTION 441.13

(1) Training instructors who teach systems, integrated responses, transients and simulator courses shall successfully complete a SRO examination.

(2) Instructors shall attend appropriate retraining programs that address, as a minimum, current operating history, problems and changes to procedures and administrative limitations. In the event an instructor is a licensed SRO, his retraining shall be the SRO requalification program.

RESPONSE:

See Subsection 18.1.

QUESTION 441.14

(1) Applicants for SRO license shall have 4 years of responsible power plant experience, of which at least 2 years shall be nuclear power plant experience (including 6 months at the specific plant) and no more than 2 years shall be academic or related technical training.

(2) Certifications that operator license applicants have learned to operate the controls shall be signed by the highest level of corporate management for plant operation.

(3) Revise training programs to include training in heat transfer, fluid flow, thermodynamics, and plant transients.

RESPONSE:

The above has been addressed in response to TMI related requirements transmitted to the NRC on January 22, 1981. Refer to PLA-614, N. W. Curtis to B. J. Youngblood.

QUESTION 441.15

(1) Applicants for operator licenses will be required to grant permission to the NRC to inform their facility management regarding the results of examinations.

(2) Content of the licensed operator requalification program shall be modified to include instruction in heat transfer, fluid flow, thermodynamics, and mitigation of accidents involving a degraded core.

(3) The criteria for requiring a licensed individual to participant in accelerated requalification shall be modified to be consistent with the new passing grade for issuance of a license.

(4) Requalification programs shall be modified to require specific reactivity control manipulations. Normal control manipulations, such as plant or reactor start-ups, must be performed. Control manipulations during abnormal or emergency operations shall be walked through and evaluated by a member of the training staff. An appropriate simulator may be used to satisfy the requirements for control manipulations.

RESPONSE:

See Subsections 18.1 and 18.2.

QUESTION 442.1

Provide a diagram of the control area that indicates the area designated "at the controls."

RESPONSE:

Figure 442.1-1 illustrates the area designated "at the controls."

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV. 46, 06/93

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
MAIN CONTROL ROOM FSAR FIGURE 442.1-1

PP&L DRAWING

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
MAIN CONTROL ROOM FIGURE 442.1-1, Rev 47

AutoCAD: Figure Fsar 442_1_1.dwg

QUESTION 442.2

The description of administrative procedures should include Fire Protection and Temporary Procedures of an administrative nature.

RESPONSE:

A revision to FSAR Subsection 13.5.1.3 has been made to include fire protection and temporary procedures as topics addressed by administrative procedures.

QUESTION 442.3

Figure 442.1-1, submitted by Revision 4, does not adequately define the "at the controls" area. For example, an operator should not routinely enter areas behind control panels, yet Figure 442.1-1 shows the area behind panel 1Z651 (2Z651) as being "at the controls." It does not appear that an operator would have an unobstructed view of the control panels from this position. Revise Figure 442.1-1 to indicate the surveillance area (an area where continuous attention can be given to reactor operator conditions and to the manipulations at reactor controls) and areas which may be entered by the operator at the controls to verify receipt of an annunciator alarm or initiate corrective action in the event of an emergency. Please note that Regulatory Guide 1.114, "Guidance on Being Operator at the Controls of a Nuclear Power Plant," provides information on this subject.

RESPONSE:

Figure 442.1-1 is an accurate description of the control room area defined as "at the controls." Panels 1Z651 and 2Z651 are in fact typers that stand approximately waist-high and in no way obstruct the operator's view at the control panels. The operator has a significant need to have access to the 4 typers in this area to obtain information vital to the safe operation of the plants. Regulatory Guide 1.114 was used extensively in the preparation of the FSAR, and it is our opinion that we are in complete compliance with this publication.