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September 22, 2006
Project No. 00037-000
File No. P-3

Proposed Revision to Sargent & Lundy (S&L)
Topical Report SL-TR-1A
Quality Assurance (QA) Program

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001
Attention: Ms. H. Cruz – Mail Stop 7 E1A

Gentlemen:

By letter dated December 30, 2005, Sargent & Lundy (S&L) submitted the proposed Revision 18 to our Nuclear Quality Assurance Program Topical Report, SL-TR-1A for review. Based on a conversation with Girija Shukla of the United States Nuclear Regulatory Commission (NRC) on April 6, 2006, S&L has made additional changes to the proposed Revision 18. Per Mr. Shukla's request, these responses were emailed to the NRC on May 4, 2006. Recently, Ms. H. Cruz of the NRC has requested these changes be formally submitted.

Below is a description of each additional change and the bases for concluding that the applicable changes continue to satisfy the requirements of 10 CFR 50.

NRC Comment No. 1: SL-TR-1A does not reference the requirements of 10 CFR Part 52, Early Site Permits, Standard Design Certifications and Combined Licenses. The topical report will need to reference these requirements for activities affecting future nuclear power plants under this regulation.

> *S&L Response: References to 10 CFR Part 52 were added to Chapters 00.00, 02.00 and 16.00.*
>
> *Specific changes are as follows:*
>
> - *references to 10 CFR 50.34 (f) (3) (ii) and 10 CFR 52.47 (a) (ii) were added after the reference to NRC Generic Letter 83-28 on Page 00-1,*
> - *references to 10 CFR 52.17 (a) (i) and 10 CFR 52.18 for early site permits, 10 CFR 52.47 (a) (i) and (ii) and 10 CFR 52.48 for standard design certifications, and 10 CFR 52.79 (a) (i), (b) and (c) and 10 CFR 52.81 for combined licenses all were added to Page 02-1,*

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- *references to 10 CFR 52.78, 10 CFR 50.120 and 10 CFR 55.4 were added at the bottom of Page 02-4, and*
- *reference to 10 CFR 52.37 were added to Page 16-2.*

NRC Request for Additional Information No. 1: On Page 00-4, Item 00.00c(4) states that: "For Supplement 2S-3 of NQA-1: S&L may apply a 90 day grace period to the requirements for documented annual evaluation of lead auditor proficiency. When the grace period is applied, the next due date for the activity is based upon the original scheduled date. However, in all cases the periodicity shall not exceed one year plus 90 days."

For justification to the exception, the S&L cover letter to this SL-TR-1A QAP change also states that "this exception, contained in the Florida Power and Light (FP&L) Company Quality Assurance (QA) Topical Report has been approved by the Nuclear Regulatory Commission (NRC)." We request additional information on the specific reference to the NRC staffs previous approval of this exception to the audit frequency in the FP&L Quality Assurance Topical Report. We also request additional information on whether this exception is acceptable for other nuclear power plants.

__S&L Response:__ This exception is contained in Florida Power & Light's Topical Quality Assurance Report (TQAR) 1 -76A, Appendix C dated April 1, 2004. However, this was derived from an exception that the NRC granted to Rochester Gas and Electric Corporation (RG&E) via letter from Guy Vissing (NRC) to Robert Mecredy (RG&E) dated July 22, 1998. The NRC Safety Evaluation is attached to this letter. The exceptions granted to FP&L and RG&E were to Regulatory Guide 1.146, but they are equally applicable to Section 6.3 of Supplement 2S-3 of ASME NQA-1-1994. This grace period would be used by S&L on FPL Group Projects (e.g., Duane Arnold, Seabrook, St. Lucie and Turkey Point) as well as for other nuclear projects.

As noted in the NRC Safety Evaluation for RG&E, the NRC staff's regulatory position on the required periodicity for the reevaluation of lead auditors was not aimed at preventing flexibility in the scheduled performance of the reevaluations but rather at providing an objective measure for ensuring suitable periodic intervals for activities affecting quality.

NRC Comment No. 2: On Page 11-3, Item 11.05, states, in part, that: "S&L may generate preoperational/startup test procedures for S&L or non-S&L design systems. Procedures are generated and reviewed by cognizant personnel in accordance with governing S&L procedures. Preparers, reviewers, and approvers of preoperational/startup test procedures meet the qualifications of ANSI/ANS 3.1-1987."

SL-TR-1A should reference NRC Regulatory Guide (RG) 1.68, Initial Test Programs, Revision 2, dated August 1978, and RG 1.8, Qualification and Training of Personnel for Nuclear Power Plants, Revision 3, dated May 2000, which endorses ANSI/ANS 3.1-1 993, "Selection, Qualification, and Training of Personnel in Nuclear Power Plants." RG 1.8 and ANSI/ANS 3.1-1993 should also be referenced in Section 2.0, QAP, Item 02.06 for qualification of Quality Assurance (QA) personnel. On page 11-1, RG 1.8 and ANSI/ANS 3.1-1993 should also be correctly referenced in place of ANSI/ANS 3.1-1987. RG 1.68 contains preoperational and startup test procedures for safety related and important to safety systems that are subject to QA requirements. RG 1.8 and ANSI/ANS 3.1-1993 contain current training qualification requirements for QA and quality control personnel who review, revise and approve preoperational and startup test procedures in nuclear power plants.

S&L Response: *S&L agrees that commitments to Revision 3 of RG 1.8 and ANSI/ANS 3.1-1993 with exceptions should be added to SL-TR-1A.*

Commitments to Revision 3 of RG 1.8 and ANSI/ANS 3.1-1993 were added to Pages 00-3 (in lieu of ANSI/ANS 3.1-1987 and -1978), 11-1 and 11-3.

The exceptions that S&L proposes to take to RG 1.8 and ANSI/ANS 3.1 are:

- *S&L commits to Part 1 and Appendix 2A-1 of the 1994 Edition of ANSI/ASME NQA-1 in lieu of the 1983 Edition, and*
- *Alternatives to the education and experience requirements, such as experience other than at a nuclear-fueled electric power production plant, shall be evaluated and documented by the Chief Executive Officer for the Quality Assurance Manager, by the Quality Assurance Manager for an individual providing quality assurance supervision and other members of the Quality Assurance Division, and by the responsible manager for other personnel in lieu of the applicable plant manager.*

Regarding Revision 2 of RG 1.68, Initial Test Programs, S&L suggests that the appropriate place for a licensee to take a position on RG 1.68 is in the SAR, Chapter 14 "Initial Plant Test Program." S&L will then follow whatever position the applicable client takes on this guide. This is consistent with Sections 14.2, VI of 17.1 and 17.2 (and also II.U and VI of draft 17.5) of NUREG-0800. Thus, we believe that a reference to RG 1.68 is neither desirable nor necessary.

NRC Comment No. 3: The statements on page 00-1 "The applicable criteria in this program shall be applied in a graded approach to radioactive material packaging and ISFSIs," is inappropriate.

We do not approve supplier/vendor QA programs. We require the COC holder (10 CFR 71.37) or the licensee (10 CFR 71.17) to apply their NRC approved 10 CFR Part 71 QA program to any quality activities of their contractors/fabricators/supplier etc.

S&L Response: *As noted in my December 30, 2005 cover letter, S&L is not requesting approval from the NRC for SL-TR-1A in accordance with 10 CFR 71 or 10 CFR 72. Nonetheless, the intent of the statement on Page 00-1 is that the graded application will be in a manner such as described in Appendix A of RG 7.10 or in NUREG/CR-6407 as stated in the following sentences on Pages 00-1 and 00-2. The purpose of these NRC documents is to describe acceptable methods for developing a QA Program with a graded approach. Work S&L performs under 10 CFR 71 or 10 CFR 72 is further controlled in project-specific documents such as Project Work Plans, project instructions and project quality plans. These project-specific documents are in compliance with whatever contractual requirements are imposed on S&L by the COC holder, licensee or other client.*

To facilitate your review, the change out pages for SL-TR-1A, Revision 18, that are referenced in/affected by the above responses are enclosed.

Yours very truly,

Randall L. Kurtz

Randall L. Kurtz
Quality Assurance Director

RLK:tls
Enclosures
Copies:
H. Cruz - NRC
A. W. Wendorf
NRC Letter 9-22-06.doc

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 22, 1998

Dr. Robert C. Mecredy
Vice President, Nuclear Operations
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, NY 14649

SUBJECT: APPROVAL OF PROPOSED REVISION 25 TO THE ROCHESTER GAS AND ELECTRIC CORPORATION'S R. E. GINNA NUCLEAR POWER PLANT QUALITY ASSURANCE PROGRAM FOR STATION OPERATION (TAC NO. MA0391)

Dear Dr. Mecredy:

By letter dated December 17, 1997, you transmitted proposed Revision 24 to the R. E. Ginna Nuclear Power Plant Quality Assurance Program for Station Operation (QAPSO). Revision 24 to the QAPSO was submitted in accordance with the requirements of 10 CFR 50.54(a)(3) as reflecting changes that reduced commitments in the QAPSO description previously approved by the NRC. However, this submittal also included changes for which RG&E was not seeking NRC approval based on the licensee's conclusion that they had no impact on commitments in the QAPSO.

As a result of requests for additional information by the NRC staff and additional reorganization changes, you amended or clarified the original submittal via correspondence dated April 6, 1998. This submittal forwarded Revision 25 to the QAPSO which provided additional justification for changes previously identified as reductions in commitment in Revision 24 to the QAPSO, and also identified new organizational changes for which you were not seeking NRC approval. Therefore, Revision 25 to the QAPSO superseded Revision 24 in its entirety.

The enclosed safety evaluation documents the bases for our conclusion that the reductions in commitments identified in Revision 25 to the QAPSO continue to satisfy the requirements of Appendix B to 10 CFR Part 50 and are, therefore, acceptable.

Sincerely,

Guy S. Vissing

Guy S. Vissing, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosure: Safety Evaluation

cc w/encl: See next page

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cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

PROPOSED REVISION 25 TO THE ROCHESTER GAS AND ELECTRIC CORPORATION

QUALITY ASSURANCE PROGRAM FOR STATION OPERATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

By letter dated December 17, 1997, Rochester Gas and Electric Corporation (RG&E) transmitted proposed Revision 24 to the R. E. Ginna Nuclear Power Plant Quality Assurance Program for Station Operation (QAPSO). Revision 24 to the QAPSO was submitted in accordance with the requirements of 10 CFR 50.54(a)(3) as reflecting changes that reduced commitments in the QAPSO description previously approved by the NRC. However, this submittal also included changes for which RG&E was not seeking NRC approval based on the licensee's conclusion that they had no impact on commitments in the QAPSO.

As a result of requests for additional information by the NRC staff (Reference 2) and additional reorganization changes, RG&E amended or clarified its original submittal via correspondence dated April 6, 1998 (Reference 3). This submittal forwarded Revision 25 to the QAPSO which provided additional justification for changes previously identified as reductions in commitment in Revision 24 to the QAPSO, and also identified new organizational changes for which RG&E was not seeking NRC approval. Therefore, Revision 25 to the QAPSO superseded Revision 24 in its entirety. This evaluation only addresses changes in Revision 25 to the QAPSO which RG&E has deemed to be reductions in commitment pursuant to 10 CFR 50.54(a)(3).

2.0 EVALUATION

In its December 17, 1997, submittal (Reference 1), RG&E proposed to establish that a "grace period" of twenty five per cent (25%), not to exceed 90 days, be applied to frequencies for performance of periodic activities described in the QAPSO and the regulatory guides and standards listed in the QAPSO, Table 17.1.7-1, "Conformance of Ginna Station Program to Quality Assurance Standards, Requirements, and Guides."

In its request for additional information (RAI) dated April 6, 1998, the NRC requested that RG&E supplement its submittal to clarify which specific periodic activities described in Table 17.1.7-1 of the QAPSO would be affected by the (plus) 25% "grace period." NRC also requested that RG&E describe the impact of the proposed deferral on RG&E's audit activities and corresponding commitments to Regulatory Guide (RG) 1.33, "Quality Assurance Program Requirements (Operation)", and RG 1.144, "Auditing of Quality Assurance Programs for Nuclear Power Plants." RG&E incorporated its response to the NRC's RAI in Revision 25 to QAPSO which was transmitted via letter dated June 4, 1998. In this revision to the QAPSO, RG&E proposed to revise its commitments to RGs and standards as necessary to apply a grace period of 90 days for the performance of the following activities:

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- Annual Supplier Evaluations in accordance with RG 1.144, Revision 1 (Section C.3.b.2)
- Triennial Vendor Audits in accordance with RG 1.144, Revision 1 (Section C.3.b.(2))
- Recertification in accordance with ANSI N45.2.23-1978, "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants" (Sections 3.2 and 5.3)
- Annual Evaluations in accordance with ANSI N45.2.6-1978, "Qualifications of Inspection, Examination, and Testing Personnel for Nuclear Power Plants" (Section 2.3)
- Internal Audits in accordance with ANSI N18.7-1972, (Section 4.4)

Specifically, RG&E has proposed to modify its RG commitment as follows:

1. <u>RG 1.33, Revision 0</u>

   Internal Audits - Section C.3.a.(1) of RG 1.144 refers to RG 1.33 for requirements. Since RG&E is committed to RG 1.33, Revision 0, except for Appendix A, ANSI N18.7-1972 requirements are invoked. A grace period of 90 days will be applied to the 24-month frequency for internal audits described in Section 4.4 of ANSI N18.7-1972, which states that audits of safety related activities are completed "within a period of two years." RG&E noted that this grace period will not be applied to audits of the Nuclear Emergency Response Plan to satisfy the requirements of 10 CFR 50.54(t), and Station Security Plan to satisfy the requirements of 10 CFR 50.54(p)(3), 73.56 (g)(1) and (g)(2) and 10 CFR 73.55(g)(4). Audit frequency and further discussion of these audits are described in their respective plans.

2. <u>RG 1.58, "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel," Revision 1</u>

   Annual Evaluations - Section 2.3 of ANSI N45.2.6 -1978 states that "Any person who has not performed inspection, examination, or testing activities in his qualified area for a period of one year shall be reevaluated..." The 90-day grace period will be applied to this activity.

3. <u>RG 1.144, Revision 1</u>

   (a) Supplier Audits - Section C.3.b.(2) of Reg. Guide 1.144. Revision 1 states that audits be performed on a "triennial basis." The 90-day grace period will be applied to this activity. Section 17.2.5 of the QAPSO is being revised to allow for application of the grace period.

   (b) Supplier Evaluations - Section C.3.b.(2) of Reg. Guide 1.144 Revision 1 states that documented evaluations be performed "annually". The 90-day grace period will be applied to this activity.

   (c) Revised commitment to perform vendor audits from "at least every three years" to "on a triennial basis" to be consistent with the wording used in RG 1.144, Revision 1, Section C.3.b.(2).

4. RG 1.146, "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants," Revision 0

   Lead Auditor Recertifications - Sections 3.2 and 5.3 of ANSI N45.2.23-1978 require that an annual assessment be performed of each lead auditor's qualification and that each lead auditor's records be updated annually. The 90-day grace period will be applied to this a tivity.

Additionally, RG&E modified QAPSO Section 17.1.7, "Regulatory Commitments," to establish a commitment that for activities deferred in accordance the 90-day "grace period," the next performance due date for such activities will be based on their originally scheduled date, i.e., in all cases, the periodicity for these activities will not be allowed to exceed the original RG commitment plus 90 days.

Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires, in part, that the quality assurance program provide for indoctrination and training of personnel performing activities affecting quality as necessary to ensure that such personnel achieve and maintain suitable proficiency, and it also establishes that audits of the quality assurance programs for these facilities (including their suppliers) be conducted at regular intervals. As described above, RG&E relies on its commitments to RGs 1.33, 1.58, 1.144, anc 1.146 to satisfy these requirements.

While Appendix B to 10 CFR Part 50 provides that audits be performed "periodically," and that suitable personnel proficiency be maintained, it does not provide specific intervals for performing these activities. As a result, the NRC established nominal periodicity intervals for certain activities described in RGs 1.33, 1.58, 1.144, and 1.146. However, the NRC staff's regulatory position on the required periodicity for these activities was not aimed at preventing flexibility in the scheduled performance of such activities but rather at providing an objective measure for ensuring plant personnel proficiency and suitable periodic intervals for activities affecting quality as required by the regulations.

Since the 90-day grace period proposed by RG&E only aims to allow some limited additional flexibility in scheduling activities associated with the subject RGs, personnel proficiency standards and periodicity objectives in the QAPSO will remain unchanged. This is consistent with the provisions in Section 17.2 of NUREG-0800, "Standard Review Plan," (SRP) and is, therefore, acceptable.

3.0 CONCLUSION

While the proposed 90-day deferral period (grace period) proposed by RG&E for the RG activities described above constitute a reduction in commitments in the QA program description previously approved by the NRC, such exceptions continue to satisfy the provisions of Section 17.2 of the SRP. Therefore, proposed Revision 25 to RG&E's QAPSO, dated June 4, 1998, continues to comply with the quality assurance criteria of Appendix B to 10 CFR Part 50 and is acceptable.

4.0 REFERENCES

1.0 Robert C. Mecredy (RG&E) letter to USNRC, "Revised Submittal of Quality Assurance Program for Station Operation - R.E. Ginna Nuclear Power Plant - Docket No. 50-244," dated December 17, 1997.

2.0 USNRC Letter to RG&E, "Request for Additional Information Concerning Revision 24 of the Quality Assurance Plan for the R.E. Ginna Nuclear Power Plant (TAC No. MA0391)," dated April 6, 1998.

3.0 Robert C. Mecredy (RG&E) letter to USNRC, "Revised Submittal of Quality Assurance Program for Station Operation - R.E. Ginna Nuclear Power Plant - Docket No. 50-244," dated June 4, 1998.

Principal Contributor: J. Peralta

Date: July 22, 1998

00.00 INTRODUCTION

This Sargent & Lundy LLC (S&L) Nuclear Quality Assurance Program was established by management policy. It is intended to be used ~~primarily~~ to assure the quality of modifications and design analyses for ~~operating~~ *the construction, operation or decommission of* nuclear plants and gaseous diffusion plants, and of the design and construction of radioactive material packaging and of independent spent fuel storage installations (ISFSIs). ~~It is, however, written to also assure the quality of design analyses and modifications for nuclear plants that are under construction or are being decommissioned.~~ The program is employed where the structures, systems and/or components are classified as important to safety insofar as they prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Safety-related structures, systems and components of nuclear power plants controlled by this Quality Assurance Program are identified in the Safety Analysis Report (usually Section 3.2) and in more detailed lists developed in response to NRC Generic Letter 83-28 *or 10 CFR 50.34 (f) (3) (ii) referenced by 10 CFR 52.47 (a) (ii)*. Quality assurance commitments for other types of important to safety items, as found in licensees' or U.S. Department of Energy contractors' quality assurance programs and other licensing basis documents, are specified to S&L in contract documents. Project instructions or project work plans shall delineate the applicability of this program to these other types of items.

The applicable criteria in this program shall be applied in a graded approach to radioactive material packaging and ISFSIs. The application shall be to an extent that is commensurate with the importance to safety, such as described in Appendix A of Regulatory Guide 7.10 (see item i in this chapter), or its equivalent for ISFSIs, such as the classification system described in

NUREG/CR-6407 titled "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety."

The applicable criteria in this program shall be applied in a graded approach to operating gaseous diffusion plants to an extent that is commensurate with the importance to safety and is consistent with the quality assurance program implemented by the United States Enrichment Corporation (USEC), or its successor, in accordance with 10 CFR 76.93.

To implement the program, standard operating procedures have been prepared. Revisions to the Nuclear Quality Assurance Program and the standard operating procedures will be made, in accordance with a standard operating procedure, for any of the following reasons:

a. the program or standard operating procedures may be incomplete, unclear or incorrect;

b. the resolution of a nonconformance may require change to some portion of the program or standard operating procedures;

c. the personnel implementing or auditing the program or standard operating procedures determine that the program and/or procedures do not effectively control a work function;

d. the standards, codes, regulatory requirements, or organization may be changed.

S&L policy makes compliance with the S&L Nuclear Quality Assurance Program and implementing procedures mandatory for all personnel performing activities relating to safety.

For limited scope projects, such as modification work for operating plants, implementation of various elements of this Nuclear Quality Assurance Program will depend on S&L's assigned responsibilities on the project.

The S&L Nuclear Quality Assurance Program, as represented herein, complies with Title 10 of the Code of Federal Regulations, Part 50, Appendix B, titled "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." S&L is committed to meeting and implementing the applicable provisions of the following requirements except as indicated below and/or as these provisions may be modified by a commitment in an applicable SAR:

a. *Regulatory Guide 1.8, May 2000 – Qualification and Training of Personnel for Nuclear Power Plants* (ANSI/ANS-3.1– ~~1987~~ *1993 - Selection* ~~Selection,~~ Qualification, and Training of Personnel for Nuclear Power Plants*) with the following exceptions:*

~~*. For qualifications of the Quality Assurance Manager, S&L is committed to ANSI/ANS-3.1 - 1978.*~~

(1) *S&L commits to Part I and Appendix 2A-1 of the 1994 Edition of ANSI/ASME NQA-1 in lieu of the 1983 Edition (see Regulatory Guide 1.28 below).*

(2) *Alternatives to the education and experience requirements, such as experience other than at a nuclear-fueled electric power production plant, shall be evaluated and documented by the Chief Executive Officer for the Quality Assurance Manager, by the Quality Assurance Manager for an individual providing quality assurance supervision and other members of the Quality Assurance Division, and by the responsible manager for other personnel in lieu of the applicable plant manager.*

Qualification requirements for the Quality Assurance Manager are established in a position description which includes the following prerequisites:

(1) Management experience through assignments to responsible positions.

(2) Knowledge of QA regulations, policies, practices, and standards.

(3) Experience working in QA or related activity in reactor design, construction, or operation or in a similar high technological industry.

b. Regulatory Guide 1.26, February, 1976 - Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants.

c. Regulatory Guide 1.28, August 1985 - Quality Assurance Program Requirements (Design and Construction) (ANSI/ASME NQA-1 - Quality Assurance Requirements for Nuclear Facility Applications) with the following exceptions and clarification:

(1) S&L commits to Part I, Subparts 2.4, 2.5, 2.7, and 2.8 of Part II, and Appendix 2A-1 of the 1994 Edition of ANSI/ASME NQA-1.

(2) S&L deviates from the Introduction to Part I of NQA-1 in the following definitions:

(a) Commercial Grade item – See the current definition in 10 CFR 21.3.

(b) Nonconformance – A condition of, or affecting, a structure, system, or component in which there is a failure to meet

02.00 QUALITY ASSURANCE PROGRAM

02.01 This Quality Assurance Program has been established in accordance with the requirements of 10 CFR Part 50, Appendix B.

This program has also been established to meet the relevant requirements of 10 CFR 52 titled "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants." These requirements are 10 CFR 52.17 (a) (i) and 10 CFR 52.18 for early site permits, 10 CFR 52.47 (a) (i) and (ii) and 10 CFR 52.48 for standard design certifications, and 10 CFR 52.79 (a) (i), (b) and (c) and 10 CFR 52.81 for combined licenses.

During the preparation of the Program and the standard operating procedures, steps are taken to verify that the S&L Nuclear Quality Assurance Program and procedures responds to each of the applicable criteria of 10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants; 10 CFR 71, Subpart H, Quality Assurance; 10 CFR 72, Subpart G, Quality Assurance; 10 CFR 76.93, Quality Assurance; and to the requirements of the applicable Regulatory Guides, Regulatory Issue Summary, and NRC Generic Letters referenced in Chapter 00.00, Introduction (except as noted therein). NRC Regulatory Guides are reviewed for suitability and used as appropriate for S&L activities. The Generic Letters are used in conjunction with current regulations.

Those responsible for defining the content of the Nuclear Quality Assurance Program are the Chief Executive Officer and the Quality Assurance Manager. The Quality Assurance Manager is responsible for approval of this Quality Assurance Program and implementing procedures. The Chief Executive Officer provides senior management approval of this Quality Assurance Program and the standard operating procedures.

online configuration management database. To the extent appropriate, controls are established to prevent unauthorized changes to verified and validated program files. Temporary changes to listed programs may be authorized in special circumstances. However, all such changes are required to be validated and documented.

02.06 To assure that appropriate skills are utilized in the performance of quality-related activities:

a. Personnel responsible for performing quality-affecting activities are instructed as to the purpose, scope, and implementation of this Quality Assurance Program, project instructions, and procedures.

b. Personnel in the Quality Assurance Division, as well as technical specialists who assist with audits, are trained and qualified in the principles, techniques, and requirements of the activity being performed.

c. The technical and administrative processes used to produce deliverables have been defined. Each of these processes has a formal description.

   Personnel who perform quality-related activities are required to be qualified in the applicable process. A standard operating procedure describes the different qualification levels and what activities each level authorizes the person to perform. Records are maintained of the process description and personnel qualifications.

d. Proficiency of personnel performing and verifying activities affecting quality is maintained by retraining, re-examining, and/or re-certifying as determined by management or program commitment.

In accordance with 10 CFR 52.78, personnel associated with the operating phase of combined licenses shall be trained and qualified in accordance with 10 CFR 50.120 titled "Training and Qualification of Nuclear Power Plant

Personnel" derived from a systems approach to training as defined in 10 CFR 55.4, as these are applicable to contractor personnel. The NRC has determined that, based on a sample review of process descriptions, individual training records, and standard operating procedures that S&L's personnel qualification certification and training program is consistent with the requirements of 10 CFR 50.120 and 10 CFR 55.4, as well as the guidance in ANS 3.1–1993 (Section 00.00 of this program) regarding qualification of contractor personnel and establishment of a program based on the five elements of a systems approach to training, as defined in 10 CFR 55.4 (reference letter from S. Dembek (NRC) to R. Kurtz (S&L) dated January 2, 2004).

02.07 Differences of opinion between Quality Assurance and other S&L organizations are resolved by the Chief Executive Officer. Resolution is documented.

02.08 Management annually assesses the adequacy of this QA Program's overall implementation. This assessment is initiated by the Chief Executive Officer. The management team is led by an S&L owner and consists of senior level personnel, such as Project Managers and Senior Project Engineers, with expertise in the engineering disciplines. The report of the assessment is approved by the Chief Executive Officer and is distributed to the responsible management for action.

11.00 TEST CONTROL

11.01 S&L does not conduct tests other than of computer software. However, on request, S&L suppliers may test safety-related items and S&L provides guidance to clients on formulation of their test programs. S&L provides the following services in connection with test activities performed by non-S&L organizations:

a. surveillance of tests in progress;

b. inclusion of test requirements, parameters and acceptance criteria in design and procurement documents in accordance with applicable codes, standards, and regulatory documents;

c. development of preoperational, startup, and other test procedures; review of test procedures submitted by clients or suppliers. Personnel who prepare or review test procedures or evaluate the adequacy of such procedures to accomplish the test objectives are certified as Level III testers in accordance with NRC Regulatory Guide 1.28, as delineated in Chapter 00.00, Introduction, or as Preoperational Test Engineers or Startup Test Engineers in accordance with *Regulatory Guide 1.8 (Section 00.00 of this program)* ~~ANSI/ANS-3.1-1987~~, as appropriate;

d. review of test reports, evaluation of test results.

11.02 If an S&L supplier will be installing safety-related items, procedures provide criteria for determining the accuracy requirements of test equipment and criteria for determining when a test is required or how or when testing activities are performed.

11.04 Inclusion of test criteria, instructions, and specifications in design and procurement documents is governed by Chapters 03.00 and 04.00 of the program and implementing procedures. Procurement documents specify witness points, acceptance limits, test environments, personnel certification, and other requirements to be included in procedures submitted by the supplier.

11.05 S&L may generate preoperational/startup test procedures for S&L or non-S&L designed systems. Procedures are generated and reviewed by cognizant personnel in accordance with governing S&L procedures. Preparers, reviewers, and approvers of preoperational/startup test procedures meet the qualifications of Regulatory Guide 1.8 (Section 00.00 of this program). Preoperational/startup test procedures include test instructions, acceptance/rejection criteria, text prerequisites, mandatory witness points, documentation of data and results, and related items. Procedures are consistent with design criteria and project requirements, and with codes, standards, and regulatory documents.

Vendor and client test procedures submitted to S&L are reviewed for compliance with procurement documents and inclusion of the above items. Reviews are performed and documented by qualified persons.

action necessary to correct the condition and to preclude its recurrence is taken. This is verified and the corrective action documented.

16.03 S&L complies with 10 CFR 21 and 10 CFR 50.55(e) as part of its corrective action program, including the control of nonconforming items in accordance with Chapter 15.00 of this program. *In accordance with 10 CFR 52.37, an early site permit is a construction permit for the purposes of 10 CFR 21.*