

SEP 18 1972

Docket No. 50-331

Roger S. Boyd, Assistant Director for BWR's, I  
THRU: Walter R. Butler, Chief, BWR-1, I

Original signed by  
George E. Lear

*for Walt Butler*

QUESTIONS FOR IOWA ELECTRIC LIGHT AND POWER COMPANY (IELP) ON THE DUANE  
ARNOLD FSAR

Attached is a letter to IELP with questions for additional information in the following areas: thermal-hydraulic analysis, abnormal operational transients, control rod drop accident, gadolinia-UO<sub>2</sub> mix core performance, diesel generator room design, flooding of the turbine building, implementation of OSHA standards in Title 29 CFR Part 1910, quality assurance during plant operations, and the planning for increase in operating power level from rated to design value.

The new areas that are of significance are briefly described below:

- a. Thermal-hydraulic Analysis - Increased attention to fluid flow, power patterns, power peaking, and MCHFR for the new gadolinia-UO<sub>2</sub> mix core has been the basis for questions seeking a broader understanding of anticipated core performance.
- b. Abnormal Operational Transients - New information on these transients for a gadolinia-UO<sub>2</sub> mix core is requested.
- c. Control Rod Drop Accident - An update of this accident analysis is sought (the staff position has not been formally approved as yet, but should be available by the time the applicant responds).
- d. Occupational Health and Safety Administration (OSHA) Standards - Questions based on the OSHA standards have been asked for evaluation of crane operation and compressed gas storage. This represents a new area of enquiry related to Industrial Safety.
- e. Quality Assurance - A comprehensive enquiry into the QA planning for the operation of the Duane Arnold facility was prepared. Applicant response to the questions will provide ample material for understanding QA in the plant operating period.

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f. Increase in Power Level from Rated (1593 Mwt) to Design (1658 Mwt) -  
 The applicant initially requested a construction permit to build a plant to operate at 1593 Mwt rated power. In the amendment for issuance of an operating license (page 14 of IELP application received May 9, 1972), the applicant "requests license to operate this facility at a thermal power level of 1658 megawatts." Thus, the change from an application for rated power to one for design power operation permits the applicant to increase the power level from rated to design by an amendment of the Technical Specifications, after adequate operating data at the initial power level of 1593 Mwt has been collected and analyzed. In addition, a requirement exists for applicant validation or justification that the plant can safely operate not only at rated but at design power. Hence, questions of the applicant have been posed (Sections 1 and 14) to obtain further technical information and planning that will enable the staff to evaluate the increase in power level. Enquiry along the same line will be made in the review of Technical Specifications.

Original signed by  
 George E. Lear

George E. Lear, Project Manager  
 Boiling Water Reactors Branch 1  
 Directorate of Licensing

Enclosure:  
 As stated

|           |                  |                        |  |  |  |  |  |  |  |             |
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| OFFICE ▶  | <i>lrk</i> BWR-1 | <i>W. Butler</i> BWR-1 |  |  |  |  |  |  |  | <i>Memo</i> |
| SURNAME ▶ | GElear:lrk       | WButler                |  |  |  |  |  |  |  |             |
| DATE ▶    | 9/15/72          | 9/18/72                |  |  |  |  |  |  |  |             |

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AEC PDR  
 Local PDR  
 OGC

SEP 18 1972

Docket No. 50-331

Iowa Electric Light and Power Company  
 ATTN: Mr. Duane Arnold  
 President  
 Security Building  
 P. O. Box 351  
 Cedar Rapids, Iowa 52406

Gentlemen:

In order that we may continue our review of your application for a license to operate the Duane Arnold Energy Center, additional information as set forth in the enclosure is needed.

In order to maintain our licensing review schedule we will need a completely adequate response by October 20, 1972. Please inform us within 7 days after receipt of this letter of your confirmation of the schedule or the date you will be able to meet. If you cannot meet our specified date or if your reply is not fully responsive to our requests it is highly likely that the overall schedule for completing the licensing review for this project will have to be extended. Since reassignment of the staff's efforts will require completion of the new assignment prior to returning to this project, the extent of extension will most likely be greater than the extent of delay in your response.

The questions in the enclosure correspond to the relevant section of the DAEC Final Safety Analysis Report. Some of these questions may have been addressed by applicants on other dockets. Your response to these questions may be made either by incorporating the information provided for other nuclear plants by reference, or you may amend your application by submitting revised pages and supplements.

Please contact us if you desire additional discussion or clarification  
of the material requested.

Sincerely,

Original Signed by  
Roger S. Boyd

Roger S. Boyd, Assistant Director  
for Boiling Water Reactors  
Directorate of Licensing

Enclosure:  
Request for Additional Information

cc: Mr. Charles Sandford  
Vice President  
Iowa Electric Light & Power Co.  
General Office  
Cedar Rapids, Iowa 52406

Mr. Jack Newman  
Lowenstein, Newman, and Reis  
1100 Connecticut Avenue, N.W.  
Washington, D.C. 20036

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| OFFICE ▶  | L: BWR-1   | L: BWR-1 | L: AD: BWR |  |  |  |
| SURNAME ▶ | GElear:lrk | WRButler | RSBoyd     |  |  |  |
| DATE ▶    | 9/14/72    | 9/18/72  | 9/18/72    |  |  |  |

REQUEST FOR ADDITIONAL INFORMATION  
IOWA ELECTRIC LIGHT AND POWER COMPANY  
DUANE ARNOLD ENERGY CENTER  
DOCKET NO. 50-331

1.0 INTRODUCTION AND SUMMARY

- 1.1 Provide a detailed and up-to-date Table of Contents for the Final Safety Analysis Report.
- 1.2 Provide a detailed description of the program for observation, collection, and analyses of operational data at the initial power level of 1593 Mwt that will be used to validate or support the "safety of operations at that power level and at a power level corresponding to 105 percent rated steam flow (or 1658 Mwt)." The discussion of the program must be sufficiently comprehensive and fundamental to demonstrate clearly that the data to be collected and analyzed will be adequate to support an increase in power level.

3.0 REACTOR

- 3.13 Provide a map of the core showing the relative power in each gadolinia-urania fuel assembly at BOC, EOC, and other critical times in the fuel cycle. Provide the relative power in each pin of the highest powered fuel assembly at the irradiation exposure which produces the worst peaking relative to (a) MCHFR during normal operation and (b) peak clad temperature following the design basis LOCA.
- 3.14 Provide a reactor heat balance at design power 1658 MW(t).
- 3.15 Provide the values of the parameters (bundle flow, inlet subcooling, exit quality, axial power shape, hot rod to bundle average peaking factor, and bundle power) used in calculating the safety limit of Figure 1.1-1 of the Technical Specification at 1020 psig and 20, 40, 80, and 100 percent of rated core flow.
- 3.16 What is the nominal calculated MCHFR at design and rated conditions?
- 3.17 Provide a table which lists the volumes of various portions of the reactor coolant system, including the volume of (a) each recirculation loop, (b) the downcomer region, (c) the lower plenum, (d) the guide tubes, (e) the fuel assemblies, (f) the space between fuel assemblies, (g) the steam dryers and separators, (h) the steam dome, and (i) the steam lines.
- 3.18 Provide additional data which defines the performance of the recirculation pumps and the jet pumps. Provide the usual centrifugal pump curves of head, horsepower, and net positive suction head versus flow and list the specific speeds. Provide curves of the jet pump M and N ratios and the dimensions of the pump which affect hydraulic performance.

- 3.19 Provide isometric drawings of the piping of the Reactor Coolant System and the Core Standby Cooling System which show the arrangement of piping, fittings, connections, valves, hangers, restraints, and penetrations.
- 3.20 Specify the diameters of the orifices in the fuel support pieces and the orifices integral with the fuel assembly inlet nose pieces. Provide a map of the core showing the size of each type of orifice in each fuel assembly.
- 3.21 Describe the environmental conditions (i.e., temperature, humidity, pressure, radiation, time) for which the relief valves have been designed. Describe the tests performed to assure that these valves will meet these requirements.

#### 6.0 CORE STANDBY COOLING SYSTEMS

- 6.1 The automatic depressurization system (ADS) actuation can be delayed by the operator if, as stated in the FSAR on page 6.4-12, "control room information indicates the signal is false, or is not needed."
- (a) Provide the results of analyses using the Interim Acceptance Criteria evaluation model which show water level in the core and peak clad temperature for a range of break sizes, presuming the HPCI system failed and ADS actuation was delayed 2, 4, 6, and 8 minutes.
- (b) Specify the information which might indicate that the ADS signal was false or not needed. Specify the number and type of instrumentation in the control room that could provide this information.
- (c) Provide for the gadolinia-urania core the information described in Section 6.7.2 of the FSAR on conformance to the ECCS Interim Acceptance Criteria. Include curves of core flow and quality at the core inlet and exit for both small and intermediate breaks.
- 6.3 Provide the power peaking factors for each rod group that were assumed in the core heatup analyses.
- 6.4 The DAEC LPCI system logic system as described in the FSAR is designed such that both the suction and discharge valves in the unbroken recirculation loop are signaled to close. Provide the peak clad temperatures following the design bases loss-of-coolant accident which would be predicted by an analysis based on the Interim Acceptance Criteria if:
- (a) The LPCI logic incorrectly selected the broken loop and (1) closed both valves in the broken loop and (2) closed only the discharge valve in the broken loop.
- (b) The LPCI logic correctly selected the unbroken loop, but the suction valve in the broken loop closed.

- 6.5 Describe and provide the results of tests of the distribution of water from the Duane Arnold Energy Center core spray headers.
- 6.6 Provide a CSCS Performance Capability Bar Chart similar to Figure 6.3.1 based on the Interim Acceptance Criteria and evaluation models.
- 6.7 Provide the data which defines the performance of the ECCS (i.e., core spray, LPCI, and HPCI) pumps. Provide the usual centrifugal pump curves of head, horsepower, net positive suction head versus flow and list the specific speeds.
- 6.8 Describe the procedure used to calibrate the elbow taps which measure the RCIC and HPCI steam flows.
- 6.9 Provide a drawing which shows the location of all of the temperature switches within both the reactor and turbine buildings which provide isolation signals to the RCIC, HPCI or main steam line isolation valves. Also, show the location of the coolers and ventilation ducts which provide cooling to these spaces, and the location of all steam lines or other hot pipes which run through these spaces.
- 6.10 Describe additional information concerning the availability of net positive suction head for emergency core cooling and containment heat removal system pumps (Section G.1 of the FSAR). Specify the number of service water pumps, RHR pumps and RHR heat exchangers assumed to be in service. Provide a curve of suppression pool water temperature as a function of time including the initial temperature. Discuss the effect of operation of the RCIC system on suppression pool temperature and available NPSH. This question may be answered by an extension of your response to question QG1.1.

#### 8.0 ELECTRICAL POWER SYSTEMS

- 8.16 The location, arrangement and missile protection features of the diesel generator rooms should be further described:
  - a) Figure 12.1-3 is inadequate to illustrate the protective features from fires, seismic events, and missile hazards generated inside the area. Provide a detailed arrangement drawing and an elevation drawing of the diesel generator room details, equipment inside the room and construction details. Also include the nearby railroad entrance, main transformer bus duct, battery room, and essential switch gear room.
  - b) Details of the watertight features of the diesel generator room walls, doors, penetrations, and electrical services to and from the rooms are required.
  - c) Details of the stop-log watertight capability and sump pump capacity are required for those openings and pumps serving the immediate vicinity of the diesel generator rooms, essential switchgear room and battery rooms.

## 10.0 AUXILIARY SYSTEMS

- 10.12 FSAR Figure 10.10-1 and Amendment 1 response to Question 10.4 part f, should respond to the possibility of radioactive waste liquids reaching the wells through ground sources, and the potable water system. A discussion and P&I diagram for the potable water system should be provided to depict methods used to avoid contamination of the well supply by backflow through this system. Describe the number of wells, their location, and distance from possible pollution sources.
- 10.13 Amendment 1 response to Question 10.9 does not appear to describe a temporary connection from the plant heating system to the HPCI and RCIC systems for preoperational testing. Additional clarification is required.
- 10.14 Indicate the extent to which AEC Safety Guide No. 27, "Ultimate Heat Sink" will be followed. If the station design does not meet this guide, provide the basis and supporting evaluation regarding the acceptability of the present design. Your response may be incorporated into an appropriate part of Appendix G to the FSAR.
- 10.15 For all vessels that contain gas under pressure, i.e., nitrogen, chlorine, oxygen, CO<sub>2</sub> and air, provide the following:
- a) The design and operating pressure,
  - b) The maximum pressure of the gas supply,
  - c) The location of the vessel,
  - d) The total energy released if the vessel should rupture,
  - e) The possibility of the vessel or its parts to act as a missile,
  - f) The protective measures taken to prevent the loss of function of adjacent equipment essential for a safe and maintained reactor shutdown, and
  - g) For each vessel identify, discuss, and provide the basis for any exception or deviation from applicable portions of the Occupational Safety and Health Administration standards in Title 29, Code of Federal Regulations, Part 1910 as follows: Subpart H - "Hazardous Material", Sections 1910.101 (Compressed Gases), 1910.103 (Hydrogen), and 1910.104 (Oxygen); Subpart M - "Compressed Gas and Compressed Air Equipment", Sections 1910.166 (Inspection of Compressed Gas Cylinder), 1910.167 (Safety Relief Devices for Compressed Gas Cylinders), 1910.168 (Safety Relief Devices for Cargo and Potable Tanks Storing Compressed Gases), and 1910.169 (Air Receivers).



- 10.16 Discuss the potential for personnel to be overcome by automatically injected CO<sub>2</sub> in spaces provided with this type of fire protection system. Discuss methods for egress and instructions for this condition.
- 10.17 Discuss the accessibility of all areas which rely on manually operated fire protection equipment. Consider such problems as radiation exposure, toxic combustion products, and CO<sub>2</sub> asphyxiation.
- 10.18 Describe and discuss the plans and means provided to absorb the resulting impact should the spent fuel cask be dropped in the spent fuel cask pool. The discussion and analysis should include:
- a) An outline drawing of the cask showing cask dimensions and its center of gravity.
  - b) The cask weight, assumed drop height, deceleration distance, deceleration force versus distance, velocity at impact considering deceleration caused by the pool water.
  - c) The maximum possible drop height.
  - d) The means, aside from administrative control, to limit the drop height to that assumed in the analysis.
  - e) A description of any energy absorbing device and the vendor identification.
  - f) The possible modes of failure of the energy absorbing device, and the inspections and surveillance to be carried out prior and during spent fuel cask movement.
  - g) Information which demonstrates that the cask cannot be tipped before being dropped and/or that the energy absorbing system is adequate even if it is dropped in the tipped condition.
  - h) The individual and combined static and dynamic concrete and reinforcing steel stresses of the fuel cask pool structure when the fuel cask pool is subjected to its maximum normal anticipated loads as well as those experienced during impact. Discuss the dynamic properties of the fuel cask pool structure that influence the dynamic stresses.
- 10.19 Describe and discuss the operating practices, qualifications and training of the people who will operate and/or direct the operation of the reactor and turbine building cranes. As a guide, use Chapter 2-3.1, "Operation - Overhead and Gantry Cranes", ANSI-B-30.2-1967, as developed for the American National Standard Safety Code for Cranes, Derricks, Hoists, Jacks and Slings.

- 10.20 For both reactor building overhead and turbine building overhead cranes, describe the degree of compliance with each of the following portions of Title 29, CFR, Part 1910, Occupational Safety and Health Administration standards in Subpart N - "Material Handling and Storage", Section 1910.179 (Overhead and Gantry Cranes): b) General Requirements, c) Cabs, d) Foot Walks and Ladders, e) Stops, Bumpers, Rail Sweeps and Guards, f) Brakes, g) Electrical Equipment, h) Hoisting Equipment, i) Warning Device, j) Inspection, k) Testing, l) Maintenance, m) Rope Inspection, n) Handling the Load, and o) Other Requirements, General.
- 10.21 Assuming the maximum drop height, discuss with the aid of drawings where appropriate, the consequences of dropping the following: a) the reactor vessel head onto the opened reactor vessel, b) the dryer-separator assembly in an opened reactor vessel, and c) the largest segment of the removable missile shield onto the closed reactor vessel.
- 10.22 Section 10.7.5 of the FSAR provides a list of essential equipment cooled by the emergency service water system; however, Figure 10.7-1 does not confirm these design aspects. The pump capacities for the emergency and general service water systems shown on Figure 10.7-2 do not correspond with the FSAR text; thus, clarification is needed.
- 10.23 Figure 10.7-2 of the FSAR indicates that a cross-connection (a removable spool piece) has been provided between the emergency service water systems. Describe this cross-connection in detail including the bases for its incorporation and the procedure for determining its use.

#### 11.0 STEAM AND POWER CONVERSION SYSTEM

- 11.7 Provide a description of the in-service inspection plans for main steam lines, steam valves, and the turbine-generator. Describe the tests and nondestructive examinations planned for the highly stressed parts of the turbine-generator, and relate the detectable flaw size to the critical crack size for those parts.
- 11.8 Provide additional description of the condensate cleanup system including such information as: a) evaluation to justify the design basis, b) data to support system adequacy, c) percentage of feedwater to be treated under the abnormal water condition, d) evaluation of the demineralizer and filter arrangement, e) evaluation of resin traps, f) amounts of water required for backwashing, and g) data on the amount of resin or filter material to be transferred to the radwaste system for normal operation.
- 11.9 In the event of rupture of the condenser circulating water inlet or outlet expansion joint, describe the maximum level to which water would rise, the relation to cooling tower sump capacity, cooling tower makeup rate, operator alarms, operating procedures, and ability to recover safety systems should this event occur. Discuss cofferdams or watertight vaults around seismic

Category I equipment in the event of flooding caused by rupture of an in-plant system. This question may be answered as an extension of your response to question 11.6.

- 11.10 Describe precautions being taken or planned to minimize the effect of cooling tower vapor mist and drift on the main transformer and insulators. It is noted that the secondary maximum prevailing winds over a 12 month period are from the NNW which places the transformer in the path of the evaporative tower plume.

14.0 ACCIDENT ANALYSIS

- 14.6 Provide analyses of all abnormal operational transients of the reactor with a gadolinia-urania core. Confirm that the overpressure protection analysis presented in Amendment 3 page changes for Section 4 and in response to Question H1.1 is for a gadolinia-urania core.
- 14.7 Provide a block diagram of the version of the transient analysis model used in the analysis of abnormal operating transients. Identify which portions are digital and which are analogue.
- 14.8 Describe the models or subprograms of each portion of the transient analysis model block diagram. Provide a spatial description (i.e., the equations), and a code description (i.e., the program). Describe any supporting codes and specify the parameters used in the transient analysis model that are obtained from each supporting code.
- 14.9 List the values of the important nuclear, thermal, and hydraulic parameters (e.g., Doppler and void coefficients; power peaking factors; decay heat generation; the conductivity and specific heat of  $UO_2$ ,  $UO_2-GD_2O_3$ , and zircalloy; pellet gap conductance, convective heat transfer coefficients, friction and loss coefficients, two-phase flow friction multipliers) which:
- (a) were used in the abnormal transient analyses of the gadolinia-uranium.
  - (b) are expected at BOC, EOC, and other critical periods in the operating cycle.

Describe the results of the sensitivity studies in which these important parameters were varied.

- 14.10 Provide the functions used in the transient analysis model which represent the performance characteristics of equipment and controls. Provide the following functions used in the analyses of all abnormal operational transients:

- (a) The safety and relief valve capacity characteristics, if different from those shown in figures 3-H1.1-3 and 3-H1.1-2.
- (b) The reactor protection system scram settings if different from those listed in Table 3.1.1 of the proposed Technical Specifications.
- (c) The pressure regulator characteristics.
- (d) The main steam line isolation valve and turbine stop valve performance characteristics relating position, flow, and time.
- (e) The control rod characteristics relating reactivity, position, and time both for the scrammed rods if different from Figure 3.6.15 and for the rod being withdrawn.
- (f) The feedwater flow controller.

- 14.11 In the core flow thermal-hydraulic analysis of abnormal operational transients, the assumption is made that the relative flow between fuel assemblies remains constant throughout a transient. Provide analyses that show the effect that flow redistribution would have on the flow and MCHFR of the hot fuel assembly.
- 14.12 In the analyses of abnormal operational transients the time from initiation of any channel trip to the de-energization of the scram relay (i.e., the protection system response time) is assumed to be 0.050 seconds. The Technical Specification 3.1.B only requires this time to be not more than 0.100 seconds. Justify this difference, modify the Technical Specification, or provide analyses of all abnormal transients based on the longest response time permitted by the Technical Specification.
- 14.13 Provide a turbine cycle heat balance showing operation at design power (1658 MWt) with the largest reduction on feedwater temperature resulting from the failure of a feedwater heater.
- 14.14 The results of the analyses of operational transients presented in the FSAR differ from those presented in the PSAR. Describe the changes which were made in the analyses which caused these changes in the results. Specifically discuss the reasons for the changes in the (a) neutron flux and core flow, following a turbine trip without bypass (FSAR Figure 14.5.3 and PSAR Figure 14.0.3) and (b) the feedwater flow, steam flow, level and pressure following closure of all main steamline isolation valves (FSAR Figure 14.5-5 and PSAR Figure 14.0.5).

- 14.15 Describe and provide the results of the analyses of a control rod drop accident in the DAEC gadolinia-urania core. If the analyses are the same as described in NEDO-10527 and its Supplement 1, compare the distribution of gadolinia and neutron flux shapes used in the NEDO-10527 Supplement 1 analysis, with those described in the DAEC FSAR (i.e., Figure 2.3.1.1 through 2.3.1.11). Provide histograms showing fuel enthalpy versus number of rods following a control rod drop accident assuming that the dropped rod is (a) the highest worth in-sequence rod and (b) the highest worth out-of-sequence rod. Provide appropriate page changes to update description and analyses of the control rod drop accident.
- 14.16 An apparent inconsistency between entries on FSAR page 14.5-5 and (new) page 4.4-9 requires clarification to identify accurately the "most severe nuclear system pressure increase".

APPENDIX D - QUALITY ASSURANCE

QD1.3 With regard to Section D.7, Operating QA Program:

- a. Provide a list or tabulation of the principal titles of the QA procedures, policies, and instructions contained in the Iowa Electric Light and Power (IELP) Company's QA Manual for Operations. How will IELP familiarize operations and contractor personnel with these QA procedures, policies and instructions and, in general, with the requirements of Appendix B to 10 CFR 50.
- b. For post-construction phases, describe how the IELP, Bechtel, and GE documents set forth in their QA Manuals are reviewed, approved, revised, distributed and controlled. Indicate how IELP, Bechtel, and GE assure that the appropriate departments and organizations properly implement these documents.
- c. Describe your system for communicating information concerning abnormal experiences at other facilities, including AEC's "Reactor Operating Experience Reports" (ROE's) and "Reactor Construction Experience Reports" (RCE's) to the appropriate design and construction organizations and for assuring that the experience embodied in these reports are considered in the program efforts.
- d. Discuss the quality assurance programs and quality control checks that are designed to assure the mechanical integrity of your fuel over its anticipated lifetime including the design review effort, review and audit of quality assurance measures, and your planned inspections of the fuel upon delivery. Indicate how your QA program with respect to fuel design and fabrication will minimize possible failures from clad hydriding, clad collapse and  $UO_2$ -clad interaction. Describe the efforts to apply the principles and practices of statistical quality control, reliability, and other recognized good practice in this area.
- e. Recent experience indicates that the bodies of valves and other cast components important to nuclear safety may have areas where the wall thickness may be less than the specified value. Describe the Quality Control procedures that you have and are using to verify wall measurements to demonstrate that the components meet design requirements.

QD1.4 With regard to Sections D.7.1 and D.7.2:

- a. Provide a discussion of the duties of each group denoted in Figure D.2-1. Specify the organizational location and duties of the Quality Assurance personnel relative to activities carried out by the groups on Figure D.2-1.

- b. Describe the role of the Quality Assurance personnel relative to membership on and/or involvement with the Plant Safety Committee and other Committees described in the FSAR. Will QA be involved in verifying the implementation of these Committee's recommendations?
- c. Provide the projected number and location of QA/QC personnel required and assigned by Iowa Electric Light and Power Company (IELP), Bechtel, and GE during the remainder of design and construction activities and for all other subphases and efforts of the project up to and including plant operation.
- d. Describe the qualification and training requirements for each QA/QC position in IELP.
- e. Provide a discussion of the role of Bechtel and GE-QA personnel (site and home office) during the phaseout of plant design and construction, plant tests checkout, initial fuel loading, initial startup tests and initial operations.
- f. Figure D.2-2 of the FSAR indicates an organization arrangement, relative to the General Electric Co. (GE), whereby Nuclear Energy Division's Quality Assurance Manager reports directly to NED's Manufacturing Manager. Describe the measures invoked by GE to assure compliance with the following portion of 10 CFR 50, Appendix B, Criterion I "Organization":

"In general, assurance of quality requires management measures which provide that the individual or group assigned the responsibility for checking, auditing, inspecting, or otherwise verifying that an activity has been correctly performed is independent of the individual or group directly responsible for performing the specific activity."

QD1.5 With regard to Section D.7.3, "Design Control":

- a. Describe the role of IELP's QA personnel relative to review of designs, specifications, procurement documents, as-built drawings, and changes. Provide flow diagrams to depict the routing, review, coordination, approval, revision, and other functions or acts involved in the preparation and updating of these documents. What system is in effect for IELP review and approval of "as built" drawings and for assuring "as-built" drawing files are updated on a timely basis?
- b. Describe the roles of the Plant Operating Review Committee and Plant Safety Committee relative to the design control role specified for these Committees in Section D.7.3.

- c. Describe the role and system of IELP in complying with AEC's Codes and Standards Rule and AEC's Deficiencies Reporting Rule.
- d. Describe the procedural measures implemented by GE, Bechtel, and IELP to assure that design drawings, specifications, and field changes are reviewed by QA/QC personnel. Describe the scope and depth of the review performed by QA/QC personnel. Does this review include the evaluation of design characteristics of changes made to determine whether they can be inspected and controlled?

QD1.6 With regard to Section D.7.4, "Procurement Document Control":

Describe the QA review of procurement documents for plant maintenance, and for the modification, repair, and replacement of material, equipment, or components for safety related systems.

QD1.7 With regard to Section D.7.5, "Instruction, Procedures, and Drawings":

Expand the discussion to describe the role of the QA staff and others relative to assuring that activities affecting quality (per 10 CFR 50) are defined by documented instructions, procedures, and drawings.

QD1.8 With regard to Section D.7.6, "Document Control":

Describe the policy and process implemented for the review and approval of changes to documents, including proposed changes to documents such as logbooks made at the plant site.

QD1.9 With regard to Section D.7.7, "Control of Purchased Material, Equipment and Services":

Expand the discussion to address fully each element of Criterion VII of Appendix B to 10 CFR 50. Discussion should include source evaluation and selection, along with a discussion of the documentary evidence required at the site (prior to installation or use) to assure that delivered product conforms to procurement requirements.

QD1.10 With regard to Section D.7.8, "Identification and Control of Material, Parts, and Components":

Describe the role of the QA staff in assuring that adequate identification and control procedures are used by site contractors. What policy and procedures are invoked to provide assurance that unacceptable items will not be incorporated into the plant?



QD1.11 With regard to Section D.7.9, "Control of Special Processes":

Describe the role of IELP's QA staff relative to Section D.7.9 of the FSAR.

QD1.12 With regard to Section D.7.11, "Test Control":

- a. Describe the role of QA/QC personnel from IELP, Bechtel, and GE to assure that testing of materials, components, systems and assemblies at vendor and supplier shops are performed in accordance with documented and approved test procedures. What involvement do these QA personnel have with the planning, including procedures, and implementation of these tests?
- b. Indicate whether test procedures are required to include the requirements and acceptance limits contained in applicable design documents, the test prerequisites, the kind of test instrumentation to be utilized, the environmental test conditions under which the test is to be made, and the assignment of individuals specifically responsible for performing the test. To illustrate your answer to the previous questions, provide a sample format for test documents.
- c. Provide detailed flow diagrams to illustrate control exercised over test procedures for their preparation, review, approval, implementation, and for final review of test results. Separately, depict this control for the acceptance, pre-operation, and start-up tests.

QD1.13 With regard to Section D.7.12, "Control of Measuring and Test Equipment":

Describe the calibration policy, schedule, and system now in effect at the plant site and that planned for the plant operating phase. What is the role of the QA personnel relative to the calibration program? Provide examples of the known and widely accepted standards that will be implemented for the calibration of plant instrumentation.

QD1.14 With regard to Section D.7.13, "Handling, Storage, Shipping, Preservation":

Describe the on-site role of the IELP, Bechtel, and GE personnel relative to this function. For the operating phase, describe the role of IELP's QA personnel relative to IELP's storekeeper function described in D.7.13.

QD1.15 With regard to Section D.7.14, "Inspection, Testing and Operating Status":

- a. Describe the tagging and other measures invoked for meeting Criterion XIV of Appendix B to 10 CFR 50. Describe the responsibilities of the IELP QA staff and the Assistant Chief Engineer with respect to this activity.
- b. Describe what will be done in terms of logging and tagging the status of inoperative or malfunctioning components in such a manner that their status cannot be overlooked in operating the plant.

QD1.16 With regard to Section D.7.15, "Nonconforming Material, Parts, and Components":

Describe the role of IELP's QA staff relative to Section D.7.15. What mechanism exists to assure timely notification of all affected parties of those cases where repair, rework, and/or reduction of requirements is anticipated. Describe the policies and steps established to assure that appropriate organizations evaluate discrepant or unacceptable materials or components and decide proper disposition. Describe the organizational arrangements for evaluation, the membership of review boards (if these are to be such), and the level of management which is to be made cognizant of the actions taken in this area.

QD1.17 With regard to Section D.7.16, "Corrective Action":

Describe the IELP policy and measures in terms of what will be done, how, and by whom relative to D.7.16. Describe whether documented procedures exist to cover this area. Discuss the involvement of IELP's line organizations, QA staff, top management in the review and approval functions pertinent to corrective action measures.

QD1.18 With regard to Section D.7.17, "Records":

- a. Summarize the records retention policy for both the construction and operational periods. Include by general category the duration of record retention and location of records. Indicate whether IELP will conform in all respects to Criterion XVII of Appendix B to 10 CFR 50.
- b. Describe the systems in effect and planned for the plant operation period for ready identification and timely retrieval of those detailed information records which support the above records and which may be maintained by a contractor or subcontractor.

QD1.19 With regard to Section D.2.18, "Audits":

Describe the nature and extent of the audit program planned for plant testing, fuel loading, initial startup, operations, maintenance, refueling, purchase of material and services, and in-service inspection efforts. Describe the estimated frequency of audits over various home office and plant activities, and describe those audits which are to be performed by the Plant Safety Committee versus those to be performed by IELP's Quality Assurance group. Describe the audit reporting policy, followup responsibility, and provisions for review of audit reports by top management of IELP.

APPENDIX G - RESPONSE TO AEC SAFETY GUIDES

QG1.0 As you have for AEC Safety Guides 1 through 17, discuss your acceptance (or nonacceptance) and provide a detailed description of your response to Safety Guides 18 through 29. Where deviation exists, provide a detailed explanation for your decision to deviate, including the basis and evaluation, that justify the decision.