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SUBJECT: Forwards info re status of Reg Guide 1.97 activities at facility,consisting of accident monitoring instrumentation program.Updated version of table of Reg Guide 1.97 variables,submitted on 850703,also encl.

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Iowa Electric Light and Power Company

JOHN F. FRANZ, JR.
VICE PRESIDENT, NUCLEAR

June 26, 1992
NG-92-2629

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No:DPR-49
Status of RG 1.97 Activities at DAEC

Reference: 1) Letter from D. Mineck (IELP) to T. Murley (NRC) dated January 8, 1992 (NG-92-0084)

2) Letter from R. McGaughy (IELP) to H. Denton (NRC) dated July 3, 1985 (NG-85-2423)

File: A-106

Dear Dr. Murley:

In Reference 1, we advised the Staff of the status of our Accident Monitoring Instrumentation Program developed to implement Regulatory Guide (RG) 1.97. Reference 1 also described certain activities which required further evaluation by Iowa Electric and a schedule for completion of these evaluations. The purpose of this letter is to inform the Staff of the results of these evaluations.

Attachment 1 to this letter provides a description of each of the intended activities and the results of our evaluations.

Attachment 2 to this letter provides an updated version of the Table of RG 1.97 variables originally submitted in Reference 2. It also discusses revisions to qualification categories for certain variables which have not been reviewed previously by the Staff. A table summarizing these revisions is included as

A003

Attachment 3. None of the proposed changes deviate from the minimum qualification criteria of RG 1.97.

Attachment 4 to this letter provides schedules for proposed plant modifications described in Attachments 1 and 2. The majority of modifications in Item 1 will be complete during the next (Cycle 12) refueling outage. The schedule for the remaining modifications, Items 2-4, is based on the availability of alternate instrumentation for the variables of concern and the low probability of Control Room instrument failure.

Based on the results of our evaluations, we believe that the accident monitoring instrumentation at the Duane Arnold Energy Center is fully functional and will permit Control Room personnel to monitor variables and systems during and following an accident.

If you have any further questions, please contact this office.

Very truly yours,

for John F. Franz, Jr.
Vice President, Nuclear

JFF/PMB/pjv~

Attachments:
1) RG 1.97 Activity Descriptions and Evaluations
2) RG 1.97 Variable Table
4) Summary of RG 1.97 Category Revisions
3) Schedule for Proposed Modifications

cc: P. Bessette
L. Liu
L. Root
R. McGaughy
C. Shiraki (NRC-NRR)
A. Bert Davis (Region III)
NRC Resident Office
Commitment Control 920057

RG 1.97 ACTIVITY DESCRIPTIONS AND EVALUATIONS

1. Primary Containment Isolation Valve List

On September 20, 1991, Iowa Electric submitted a request for an amendment to the DAEC Operating License which would remove the component lists from Section 3.7 of the DAEC Technical Specifications within the guidelines set forth in NRC Generic Letter 91-08, "Removal of Component Lists from Technical Specifications." As part of the TS amendment and implementation process, we evaluated which valves would be subject to the revised administrative controls for containment isolation valves. Upon completion of this activity, the RG 1.97 containment isolation valve position indication list was to be revised accordingly.

RESULTS OF EVALUATION

The engineering evaluation of the containment isolation valve list has been completed and the RG 1.97 Master Equipment List has been updated to include the additional valves which were identified as primary containment isolation valves. As a result of this effort, however, the position indication for certain containment isolation valves must be added to our equipment qualification (EQ) program. The addition of these components to the EQ program and any necessary upgrades will be performed in accordance with the schedule in Attachment 4.

2. Status of Standby Power

RG 1.97 requires that indication of the status of standby power and other energy sources important to safety (Type D, Category 2 variable) be available to Control Room operators during accident conditions. Our RG 1.97 instruments for this variable are located in the essential switchgear rooms along with the associated equipment. The essential switchgear rooms are within the Control Room/Control Building envelope and these instruments are accessible to operators during normal and post-accident conditions. Control Room indications consist of annunciators and alarms which are not included in our RG 1.97 program. An evaluation was performed to determine if the current RG 1.97 instrumentation for this variable is adequate to support Control Room operators during an accident.

RESULTS OF EVALUATION

The engineering evaluation of the adequacy of instrumentation for monitoring the status of standby and other power supplies has been completed. It concluded that the instrumentation currently available in the Control Room is satisfactory and meets the criteria of RG 1.97. The study, however, did recommend that certain of these annunciators and instruments be added to the RG 1.97 Master Equipment List. These additions to the Master Equipment List will be complete prior to July 15, 1992.

3. Single Point Failure Criteria for Reactor Pressure Instrument

Section 1.3.1.b of RG 1.97 states that no single failure within either the accident monitoring instrumentation or its power sources concurrent with the failures that are a condition or result of a specific accident should prevent the operators from being presented the necessary information. Our review of the RG 1.97 program identified a potential scenario wherein an instrument line failure within primary containment (as a result of a high energy line break) coincident with the loss of a single instrument power supply could result in the loss of Category 1 reactor pressure instrumentation in the Control Room. An engineering evaluation was performed to determine if the proposed scenario is credible and recommend any necessary corrective actions.

RESULTS OF EVALUATION

The engineering evaluation of the single point failure criteria for the reactor pressure instruments concluded that the scenario, while highly unlikely, would result in the loss of Category 1 reactor pressure instrumentation in the Control Room. However, the study also showed that numerous other indications of reactor vessel pressure, including HPCI and RCIC steam inlet pressure, are available as backups to the two qualified pressure instruments. These alternate indications are diverse and independent but do not meet the additional qualification criteria for RG 1.97, Category 1 variables. Therefore, the HPCI and RCIC steam inlet pressure instruments will be upgraded to meet the qualification criteria of RG 1.97, Category 1 variables. Attachment 4 to this letter provides the schedule for these modifications.

4. Availability of Instrumentation in Emergency Response Facilities

Items 8.2, "Technical Support Center (TSC)" and 8.4 "Emergency Operations Facility (EOF)" of Generic Letter 82-33, "Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability" state that variables which are essential for performance of TSC or EOF functions should be available in each of the facilities. Our response to Generic Letter 82-33 (Reference 2) stated that types B, C, D, and E variables necessary for TSC and EOF functions are provided in each facility. Our review of the RG 1.97 program noted that certain accident monitoring instruments are not monitored by the plant process computer and are therefore not directly indicated in the EOF or TSC. An engineering evaluation was performed to determine if the instrumentation currently available in these facilities is sufficient.

RESULTS OF EVALUATION

The engineering study of the availability of instrumentation in the TSC and EOF concluded that RG 1.97 instruments essential to the functions of the TSC and EOF are available to those facilities. While the plant process computer system is the preferred method for transmitting information from the instrumentation to the emergency response organization outside the Control Room, other acceptable methods of transmitting this information are in place including dedicated phone lines between the Control Room and the TSC and between the TSC and EOF. Hard copy data can also be transmitted between the TSC and EOF via facsimile. Instrumentation currently available in the TSC and EOF has received significant operational testing during emergency drills and determined to be satisfactory. Therefore, we concluded that the instrumentation currently available in the emergency response facilities meets the criteria of Items 8.2 and 8.3 of Supplement 1 to NUREG 0737.

5. Provisions for Class 1E Power for Category 1 Instruments

Section 1.3.1.c of RG 1.97 states that Category 1 instrumentation should be energized from station standby power sources and should be backed up by batteries where momentary interruption is not tolerable. We have identified certain primary containment isolation valve position indications (Type B, Category 1 variables) that are supplied by interruptible Reactor Protection System motor-generator sets. An engineering evaluation was performed to determine the acceptability of these power supplies.

RESULTS OF EVALUATION

An engineering study of the suitability of the power supplies for all RG 1.97 instruments was completed and concluded that the power supplies are adequate and meet the criteria of RG 1.97. However, we have identified certain Category 2 variable instruments (Standby Liquid Control flow and cooling water flow to ESF components) that should be enhanced by transferring their power supplies to more reliable power sources. A schedule for proposed modifications to enhance these power supplies is included in Attachment 4.

6. <u>Physical Independence of Electrical Systems - RG 1.75</u>

As described in References 1 and 2, the construction and licensing of the DAEC predated the issuance of RG 1.75, "Physical Independence of Electrical Systems." Physical and electrical separation of existing equipment was performed in accordance with specifications and design criteria in effect at the time of installation. These specifications are documented in Section 8.3, Onsite Power Systems, of the DAEC Updated Final Safety Analysis Report (UFSAR) and the DAEC Cable and Wire Installation Specification, Spec-E512. Instrumentation installed at the DAEC to meet RG 1.97 is in accordance with RG 1.75 within the constraints of the pre-existing structures, systems, and components. A position paper detailing how the design of the DAEC compares with the criteria of RG 1.75 was prepared. This position paper was provided to the NRC Staff during an inspection of the DAEC RG 1.97 program in December 1991. The following is a summary of the specific provisions of RG 1.75 from which the DAEC varies. In each case, the RG 1.75 provision is described and IELP's position stated.

- RG 1.75 Section 5.1.2 - Identification

  Exposed Class 1E raceways should be marked in a distinct permanent manner at intervals not to exceed 15 feet and at points of entry to and exit from enclosed areas.

  Cables installed in these raceways should be marked...at a sufficient number of points (intervals not to exceed 5 feet) to facilitate verification that the installation is in conformance with the separation criteria.

IELP Position

Raceway identification at intervals not to exceed 15 feet was not an original design or installation requirement at the DAEC. The DAEC cabling identification requirements as described in UFSAR section 8.3 and SPEC-E512 include color coding of divisional cables to distinguish them from other cables and to identify their specific division. Similarly, marking cables every 5 feet was not an original design or installation requirement. The DAEC engineering specification for cable and wire installations requires that cables be marked with permanent markers at each cable end. Additional intermediate markers are installed where access is possible. The current raceway and cable identifications are deemed to be adequate.

- RG 1.75 Section 5.1.3 - Cable Spreading Area and Main Control Room

Where feasible, redundant cable spreading areas should be used.

IELP Position

The DAEC Cable Spreading Room and Main Control Room were designed, constructed and cabled prior to issuance of RG 1.75. Therefore, the layout of all major safety-related equipment and components had been accomplished prior to issuance of RG 1.75. A description of the original cable routing and separation criteria is contained in UFSAR section 8.3, Onsite Power Systems.

- RG 1.75 Section 3.0 - Isolation Device

Interrupting devices actuated only by fault current are not considered to be isolation devices...

IELP Position

IELP design criteria have historically allowed the use of fuses as isolation devices. This practice is in accordance with original design specifications and provides adequate fault protection and is, therefore, deemed to be adequate.

- RG 1.75 Section 4.7 - Documentation of Analyses

Analyses performed in accordance with 4.5.c (associated circuits) and 4.6.1.b (non-class 1E circuits) should be submitted as part of the Safety Analysis Report and should identify those circuits installed in accordance with these sections.

IELP Position

The DAEC has complied with the intent of RG 1.97 and RG 1.75 to provide for physical and electrical separation of equipment within the constraints of existing plant structures, systems and components as described in UFSAR section 8.3, Onsite Power Systems, and SPEC-E512, Cable and Wire Installation Specification. The DAEC Quality Assurance program and DAEC Administrative Control Procedures provide assurance that separation requirements in accordance with the design specifications are adequately implemented, enforced and documented.

- RG 1.75 Section 4.5 Associated Circuits

Associated circuits should comply with one of the following: a) They should be uniquely identified as such and should remain with or be separated the same as those Class 1E circuits with which they are associated.

IELP Position

The cable identification and separation criteria for electrical cabling is contained in UFSAR section 8.3, Onsite Power Systems, and SPEC-E512, Cable and Wire Installation Specification. These specifications require unique cable identifiers at each cable end and prohibit routing a cable associated with one division in a raceway with cables of the other division, but do not prevent routing an associated circuit cable in a non-divisional raceway which may contain cables associated with another division. This specification is in accordance with original design criteria and is deemed adequate.

- RG 1.75 Section 5.6.2 - Internal Separation

The minimum separation distance between redundant Class 1E equipment and circuits internal to the control boards can be established by analysis of the proposed installation. Where the control board materials are flame retardant and analysis is not performed, the minimum separation distance should be six inches.

IELP Position

The DAEC Cable and Wire Installation Specification requires that redundant engineered safeguards cables within equipment be separated from cables of redundant divisions or from non-divisional cables by a minimum distance of six inches or they must be physically isolated by a fire barrier or conduit. There are, however, certain conditions under which different divisions may be routed within 6 inches within equipment. These conditions are as follows.

a. Cable interconnections are in a common logic circuit between redundant sensors of different divisions.

b. Cables are not redundant counterparts of a system or interdependent system: i.e., loss of both cables will not prevent both of two redundant counterparts from implementing the protective functions.

c. Cables are in circuits that are interlocked so that cables are not energized simultaneously.

d. Cables are in circuits for the Standby Liquid Control System.

Because these deviations from the specified practice are documented in Specification E512 and based on good engineering practice, the design specifications are deemed adequate.

7. Seismic Qualification - RG 1.100

In Reference 1, we informed the Staff that a position paper detailing Iowa Electric's compliance with seismic qualification provisions of RG 1.97 and RG 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants," had been prepared. The paper correlates DAEC original design and construction (which pre-date RG 1.100) and subsequent seismic qualification design practices to the provisions of RG 1.100 and provides recommendations concerning variations from selected criteria. Additionally, the paper outlines the actions necessary to evaluate and resolve other seismic issues identified earlier in our review of our Accident Monitoring Instrumentation Program. This paper was provided to the NRC Staff during an inspection of the DAEC RG 1.97 program in December 1991. The results of this review are described below.

IELP Position

The construction and licensing of the DAEC predated the issuance of RG 1.100. Seismic qualification of originally installed equipment or equipment installed prior to the issuance of RG 1.97 was in accordance with the requirements applicable at the time of installation. The seismic qualification criteria is described in UFSAR section 3.7. Existing equipment that did not require seismic qualification at the time of installation, but which now requires seismic qualification as post-accident monitoring instrumentation, will be evaluated in accordance with the Seismic Qualification Utility Group (SQUG) methodology. Installation of new equipment and modification or replacement of existing equipment for RG 1.97 will meet the guidelines of RG 1.100, Revision 1 within the constraints of the existing structures. Additionally, we will utilize the SQUG methodology to review and assemble the seismic qualification documentation of all Category 1 equipment. The schedule for seismic evaluation of this equipment will be in accordance with the schedule to be submitted per GL 87-02, Supplement 1, Supplemental Safety Evaluation Report on SQUG Generic Implementation Procedure.

8. Type A Variable

In reference 1, we advised the Staff of our intention to review the basis for our decision regarding Type A variables. Type A variables are defined in RG 1.97 as those variables to be monitored that provide the primary information required to permit the Control Room operator to take specific manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis events. The results of this evaluation are described below.

Results of Evaluation

In our original submittal for RG 1.97, we stated that "automatic control is provided for all safety systems to accomplish their safety function for abnormal operational transients and accidents presented in Chapter 15 of the UFSAR. Therefore, no RG 1.97 Type A variables have been identified for the DAEC."

This original decision regarding Type A variables was based on the DAEC definition of design basis events (or design basis accidents). Design basis accidents, as defined in UFSAR Section 15.0, Plant Safety Analysis, are those accidents resulting in potential radiation exposures greater than any other accident considered under the same general accident assumptions. Design basis accidents are described in detail in Chapter 15 of the DAEC UFSAR.

We have subsequently reviewed events and transients other than those described in Chapter 15 of the UFSAR and have identified a potential Type A variable for the small break LOCA inside containment scenario which is described in Chapter 6 of the UFSAR. While the small break LOCA is less limiting radiologically than a design-basis LOCA, it has a greater potential to cause drywell temperatures to exceed the containment design temperature limit. Therefore, operators are currently required by the Emergency Operating Procedures to manually initiate containment spray at elevated drywell temperatures to mitigate this event. The drywell temperature instrumentation, however, does not meet the separation criteria of a Category 1 variable in that all eight channels of drywell temperature monitoring are routed through one drywell penetration.

We are presently working with General Electric to review the original containment analysis for the small break LOCA to better understand the necessity for manual actions. This review is expected to be completed by August 31, 1992. If the results of this evaluation indicate that drywell temperature is a type A variable, we will notify the NRC accordingly.

RG 1.97 VARIABLE TABLE

Identifier	Variable	Category	Range	EQ	Seismic	Redundant	Power	Indication
B01 (Note 2)	Neutron Flux	*	*	*	*	*	*	Continuous, Recorded
B02	Control Rod Position	3	Full-In/Not-Full-In	N/A	N/A	N/A	Cat 2	Continuous (Full Core Display)
B03	RCS Soluble Boron Concentration (Sample) (Note 3)	*	0 to 1000 ppm	N/A	N/A	N/A	N/A	None
B04	Coolant Level in the Reactor	1	Bottom of Core Support Plate (153 inches below TAF) to 182 inches above the centerline of the main steam lines (458 inches above TAF) (Note 4)	Cat 1	(Note 1)	Cat 1	Cat 1	Continuous, Recorded
B05	BWR Core Thermocouples (Note 5)	*	*	*	*	*	*	*
B06	RCS Pressure	1	0 psig to 1500 psig	Cat 1	(Note 1)	Cat 1 (Note 6)	Cat 1	Continuous, Recorded
B07	Drywell Pressure	1	-5 psig to 250 psig (Note 7)	Cat 1	(Note 1)	Cat 1 (Note 7)	Cat 1	Continuous, Recorded
B08	Drywell Sump Level	3	0 gpm to 120 gpm (Note 8)	N/A	N/A	N/A	Cat 3	Continuous, Recorded
B09	Primary Containment Pressure	1	-5 psig to 250 psig (Note 7)	Cat 1	(Note 1)	Cat 1 (Note 7)	Cat 1	Continuous, Recorded
B10	Primary Containment Isolation Valve Position Indication (excluding check valves)	1	Closed/Not Closed	(Note 9)	(Note 1)	Inboard/ Outboard valves provide redundancy	Cat 1	Continuous

Identifier	Variable	Category	Range	EQ	Seismic	Redundant	Power	Indication
C01	Radioactivity Concentration or Radiation Level in Circulating Primary Coolant (Notes 3, 10)	3	½ Tech Spec Limit to 100 times Tech Spec Limit (R/hr)	N/A	N/A	N/A	N/A	Main Steamline Rad Monitors Continuous, Recorded
C02	Analysis of Primary Coolant (Gamma Spectrum) (Note 3)	*	10 µCi/gm to 10Ci/gm or TID-14844 source term in coolant volume	N/A	N/A	N/A	N/A	None
C03	BWR Core Thermocouples (Note 5)	*	*	*	*	*	*	*
C04	RCS Pressure	1	0 psig to 1500 psig	Cat 1	(Note 1)	Cat 1 (Note 6)	Cat 1	Continuous, Recorded
C05	Primary Containment Area Radiation	3	1 R/hr to 10^7 R/hr	N/A	N/A	N/A	Cat 3	Continuous, Recorded
C06	Drywell Sump Level	3	0 gpm to 120 gpm (Note 8)	N/A	N/A	N/A	Cat 3	Continuous, Recorded
C07	Suppression Pool Water Level	1	1.5 ft to 16 ft	Cat 1	(Note 1)	Cat 1	Cat 1	Continuous, Recorded
C08	Drywell Pressure	1	-5 psig to 250 psig (Note 7)	Cat 1	(Note 1)	Cat 1 (Note 7)	Cat 1	Continuous, Recorded
C09	RCS Pressure	1	0 psig to 1500 psig	Cat 1	(Note 1)	Cat 1 (Note 6)	Cat 1	Continuous, Recorded
C10	Primary Containment Pressure	1	-5 psig to 250 psig (Note 7)	Cat 1	(Note 1)	Cat 1 (Note 7)	Cat 1	Continuous, Recorded
C11	Containment and Drywell Hydrogen Concentration	1	0% to 20%	Cat 1	(Note 1)	Cat 1	Cat 1	Continuous, Recorded
C12	Containment and Drywell Oxygen Concentration	1	0% to 10%	Cat 1	(Note 1)	Cat 1	Cat 1	Continuous, Recorded
C13	Containment Effluent Radioactivity	3	10^{-7} µCi/cc to 10^5 µCi/cc	N/A	N/A	N/A	Cat 3	Continuous, Recorded

Identifier	Variable	Category	Range	EQ	Seismic	Redundant	Power	Indication
C14	Radiation Exposure Rate Inside Buildings	3	0.1 mR/hr to 1000 mR/hr (Note 11)	N/A	N/A	N/A	Cat 3	Continuous, Recorded
C15	Effluent Radioactivity - Noble Gases	3	10^{-7} µCi/cc to 10^{5} µCi/cc	N/A	N/A	N/A	Cat 3	Continuous, Recorded
D01	Main Feedwater Flow	3	0 to $4x10^{6}$ lb/hr each (2 channels) Rated feedwater flow rate is ~ $7.2x10^{6}$ lb/hr	N/A	N/A	N/A	Cat 3	Continuous
D02	Condensate Storage Tank Level	3	0 ft to 24 ft	N/A	N/A	N/A	Cat 3	Continuous
D03	Suppression Chamber Spray Flow	2	0 gpm to 15,000 gpm (Note 12)	Cat 2	N/A	N/A	Cat 2	Continuous, Recorded
D04	Drywell Pressure	1	-5 psig to 250 psig (Note 7)	Cat 1	(Note 1)	Cat 1 (Note 7)	Cat 1	Continuous, Recorded
D05	Suppression Pool Water Level	1	1.5 ft to 16 ft	Cat 1	(Note 1)	Cat 1	Cat 1	Continuous, Recorded
D06	Suppression Pool Water Temperature	2	20 °F to 220 °F (Note 13)	Cat 2	N/A	N/A	Cat 2	Continuous, Recorded
D07	Drywell Atmosphere Temperature	2	0 °F to 350 °F (Note 14)	Cat 1	N/A	N/A	Cat 1	Continuous, Recorded
D08	Drywell Spray Flow	2	0 gpm to 15,000 gpm (Note 12)	Cat 2	N/A	N/A	Cat 2	Continuous, Recorded
D09	MSIV Leakage Control System Pressure	2 (Note 15)	-5 psig to 45 psig	Cat 2	N/A	N/A	Cat 2	Continuous
D10	Primary System SRV Position	2 (Note 16)	Closed/Open > 25 psig	Cat 2	(Note 1)	N/A	Cat 1	Continuous
D11	Isolation Condenser Shell Side Water Level (Note 17)	*	*	*	*	*	*	*

Identifier	Variable	Category	Range	EQ	Seismic	Redundant	Power	Indication
D12	Isolation Condenser Valve Position (Note 17)	*	*	*	*	*	*	*
D13	RCIC Flow	2 (Note 18)	0 gpm to 500 gpm	Cat 2	N/A	N/A	Cat 2	Continuous
D14	HPCI Flow	2 (Note 18)	0 gpm to 3500 gpm	Cat 2	N/A	N/A	Cat 2	Continuous
D15	Core Spray Flow	2 (Note 19)	0 gpm to 5000 gpm	Cat 2	N/A	N/A	Cat 2	Continuous
D16	LPCI System Flow	2	0 gpm to 15,000 gpm (Note 12)	Cat 2	N/A	N/A	Cat 2	Continuous, Recorded
D17	SLCS Flow	2	0 gpm to 60 gpm	Cat 2	N/A	N/A	(Note 20)	Continuous
D18	SLCS Storage Tank Level	2 (Note 21)	0 to 100%	Cat 2	N/A	N/A	Cat 2	Continuous
D19	RHR System Flow	2	0 gpm to 15,000 gpm	Cat 2	N/A	N/A	Cat 2	Continuous, Recorded
D20	RHR Heat Exchanger Outlet Temperature	3	40 °F to 500 °F	N/A	N/A	N/A	Cat 3	Continuous, Recorded
D21	Cooling Water Temperature to ESF System Components	3	0 °F to 100 °F	N/A	N/A	N/A	Cat 3	Continuous
D22	Cooling Water Flow to ESF System Components	2 (Note 22)	0 gpm to 1500 gpm	Cat 2	N/A	N/A	(Note 20)	Continuous
D23	High Radioactivity Liquid Tank Level (Note 23)	*	*	*	*	*	*	*
D24	Emergency Ventilation Damper Position	2 (Note 24)	Open/Closed	Cat 2	N/A	N/A	Cat 2	Continuous
D25	Status of Standby Power and Other Energy Sources Important to Safety	2 (Note 25)	Current, Voltage, Alarm Conditions	Cat 2	N/A	N/A	Cat 2	Continuous or Annunciated

Identifier	Variable	Category	Range	EQ	Seismic	Redundant	Power	Indication
E01	Primary Containment Area Radiation - High Range	1	1 R/hr to 10^7 R/hr	N/A	N/A	N/A	Cat 3	Continuous, Recorded
E02	Reactor Building Area Radiation (Note 11)	3	100 mR/hr to 10^6 mR/hr	N/A	N/A	N/A	Cat 3	Continuous, Recorded
E03	Radiation Exposure Rate Inside Buildings (Note 11)	3	100 mR/hr to 10^6 mR/hr	N/A	N/A	N/A	Cat 3	Continuous, Recorded
E04	Noble Gases and Vent Flow Rate	3	0 cfm to 71,000 cfm	N/A	N/A	N/A	Cat 3	Continuous, Recorded
E05	Particulates and Halogens (Note 26)	*	*	*	*	*	*	*
E06	Radiation Exposure Meters (Note 5)	*	*	*	*	*	*	*
E07	Airborne Radiohalogens and Particulates (Note 26)	*	*	*	*	*	*	*
E08	Plant and Environs Radiation (Note 27)	*	*	*	*	*	*	*
E09	Plant and Environs Radiation (Note 27)	*	*	*	*	*	*	*
E10	Wind Direction	3	0 ° to 360 °	N/A	N/A	N/A	Cat 3	Continuous, Recorded
E11	Wind Speed	3	0 mph to 100 mph	N/A	N/A	N/A	Cat 3	Continuous, Recorded
E12	Estimation of Atmospheric Stability	3	-10 °F to 18 °F	N/A	N/A	N/A	Cat 3	Continuous, Recorded
E13	Primary Coolant and Sump Sample (Note 3)	*	*	*	*	*	*	*
E14	Containment Air Sample (Note 3)	*	*	*	*	*	*	*

NOTES TO RG 1.97 VARIABLE TABLE

1. As described in Item 7 of Attachment 1 to this letter, the DAEC was constructed and licensed prior to the issuance of RG 1.97 and RG 1.100. Post-accident monitoring instrumentation installed to meet NUREG-0737 was procured and installed as seismic class 1. IELP will evaluate the seismic qualification of pre-existing RG 1.97 equipment using the SQUG methodology in accordance with the schedule to be submitted per GL 87-02, Supplement 1. Additionally, we plan to use the SQUG methodology to review and assemble the seismic qualification documentation of all Category 1 equipment.

2. Category 1 instrumentation for monitoring neutron flux has not been implemented in the DAEC RG 1.97 program. The NRC approved the existing instrumentation for interim operation in the Safety Evaluation dated 5/9/90. RG 1.97 guidelines call for Category 1 qualifications and an instrument range of 10^{-6}% to 100% power. IELP has been a participant in efforts by the BWR Owners' Group to obtain NRC approval of alternate qualification guidelines. IELP will inform the staff of our plans for implementing this variable after the NRC issues a final decision on these alternate requirements.

3. This variable will be monitored using the Post-Accident Sampling System (PASS) installed in response to NUREG-0737 Item II.B.3. The NRC found this alternate methodology acceptable in their safety evaluation of 5/9/90.

4. Reactor Water Level Instrumentation meets Category 1 requirements over the accident range from 153 inches below the top of active fuel (TAF) to 218 inches above TAF. A single channel of non-safety Reactor Floodup Level instrumentation is provided for the range from 158 inches above TAF to 458 inches above TAF. This configuration was found acceptable in the NRC safety evaluation of 5/9/90.

   Additionally, two channels of Yarway level instruments which provide Category 1 qualified indication during power operation have been added to the RG 1.97 program to supplement the accident range instruments. The cabling and location of indicators on the Control Room front panels, however, does not meet DAEC Category 1 separation criteria. Attachment 4 to this letter provides the schedule for modification of these instruments.

5. The requirement to monitor this variable was withdrawn.

6. As described in Item 3 of Attachment 1 to this letter, a scenario has been identified under which a HELB inside containment could result in damage to the instrument piping of one channel of RG 1.97 pressure instrumentation. With a single failure in the other channel, both qualified channels of reactor pressure indication could be lost. To ensure that qualified reactor vessel pressure instrumentation is available during the postulated scenario, the HPCI and RCIC steam inlet pressure instruments will be upgraded to Category 1. (Refer to Attachment 4 for schedule.)

7. Drywell and containment pressure monitoring is provided in three, overlapping instrument ranges with redundant instruments provided for each range. Maximum design drywell pressure for the DAEC is 62 psig. The following ranges are provided:

   - 5 psig to 5 psig
   0 psig to 100 psig
   0 psig to 250 psig

8. IELP provided drywell floor drain and drywell equipment sump flow integration instruments in place of level instruments as recommended by RG 1.97. The flow integrators provide a means of evaluating increasing leakage prior to containment isolation. Other instrumentation, including drywell pressure, drywell temperature, and primary containment area radiation provides indication of leakage. The NRC found the alternate instrumentation acceptable in the Safety Evaluation of 5/9/90.

9. As described in Item 1 of Attachment 1 to this letter, a revised list of primary containment isolation valves has been developed for the DAEC. Additional valves have been included in the RG 1.97 program. As a result, modifications to the position indication of certain containment isolation valves are required to meet EQ criteria. (Refer to Attachment 4 for schedule.)

10. IELP has provided main steamline radiation monitors, drywell high range radiation monitors, primary containment area radiation monitors and a post accident sampling system as indications of increasing radioactivity levels. The NRC found the alternate instrumentation acceptable in the Safety Evaluation of 5/9/90.

11. The instrumentation for this variable covers a smaller range than recommended by RG 1.97. Extended range airborne effluent monitors and atmospheric sampling are more suitable

for detection of breaches in containment, for detecting significant releases and for assessment and long term surveillance of releases. Access is not required to service safety-related equipment. Should access be required, adequate coverage is provided through use of portable radiation survey instruments. The NRC found the range and alternate instrumentation for this variable to be acceptable in the Safety Evaluation of 5/9/90.

12. Torus and drywell spray flow is obtained from the RHR system. The RHR LPCI flow instrument together with suppression pool and drywell temperature and pressure instruments accurately and reliably measure the effectiveness of containment sprays. The NRC found this instrumentation to be acceptable in the Safety Evaluation of 5/9/90.

13. The calculated maximum bulk suppression pool temperature is 197°F. Based on this the NRC accepted the existing instrument range in the Safety Evaluation of 5/9/90.

14. The 0°F to 350°F range of existing drywell atmosphere temperature instrumentation at the DAEC covers the temperature limits for design basis accidents at the DAEC and was determined to be acceptable in the NRC Safety Evaluation of 5/9/90.

15. In Reference 2, IELP committed to provide Category 1 instrumentation for this variable although RG 1.97 recommended Category 2. We have subsequently determined that the MSIV LCS pressure instrumentation does not meet RG 1.97 Category 1 requirements. Only one instrument channel is provided for each Main Steam Line and the instrumentation does not meet the Category 1 physical separation requirements. This instrumentation does, however, meet RG 1.97 Category 2 requirements. Therefore, we propose to provide Category 2 qualified instrumentation for this variable as recommended by RG 1.97.

16. In Reference 2, IELP committed to provide Category 1 instrumentation for this variable although RG 1.97 recommended Category 2. We have subsequently determined that the Primary System Safety/Relief Valve Position indication does not meet RG 1.97 Category 1 requirements. Only one indication is available for each valve and the instrumentation does not meet the Category 1 physical separation criteria. This instrumentation does, however, meet RG 1.97 Category 2 requirements. Therefore we propose to provide Category 2 qualified instrumentation for this

variable as recommended by RG 1.97.

17. The reference to an Isolation Condenser is not applicable to the DAEC.

18. In Reference 2, IELP committed to provide Category 1 instrumentation for this variable although RG 1.97 recommended Category 2. We have subsequently determined that this instrumentation does not meet RG 1.97 Category 1 criteria. Only one channel of flow instrumentation is provided for this system at the DAEC. This instrumentation does, however, meet RG 1.97 Category 2 requirements. Therefore, we propose to provide Category 2 qualified instrumentation for this variable as recommended by RG 1.97.

19. In Reference 2, IELP committed to provide Category 1 instrumentation for this variable although RG 1.97 recommended Category 2. We have subsequently determined that the Core Spray Flow indication does not meet the RG 1.97 Category 1 requirements. Core Spray Flow indication is not recorded at the DAEC. This instrumentation does, however, meet RG 1.97 Category 2 requirements. Therefore, we propose to provide Category 2 qualified instrumentation for this variable as recommended by RG 1.97.

20. This instrument is currently powered from a lighting panel. A modification to transfer this instrument to a more reliable power source will be implemented in accordance with the schedule provided in Attachment 4.

21. In Reference 2, IELP committed to provide Category 3 instrumentation for this variable although RG 1.97 recommended Category 2. IELP took this exception because the level instrumentation was not qualified in accordance with the DAEC EQ program. The SLCS level instrumentation is located in a mild environment and therefore, equipment qualification is not required. This instrumentation, therefore, meets Category 2 requirements. We propose to provide Category 2 qualified instrumentation for this variable as recommended by RG 1.97.

22. In Reference 2, IELP committed to provide Category 3 instrumentation for this variable although RG 1.97 recommended Category 2. IELP took exception because the flow instrumentation was not qualified in accordance with the DAEC EQ program. The ESF cooling water flow instrument is located in a mild environment and therefore, equipment qualification is not required. This instrumentation, therefore, meets Category 2 requirements. We propose to provide Category 2 qualified instrumentation for this

variable as recommended by RG 1.97.

23. This variable is not required for the DAEC. The NRC accepted this position in the Safety Evaluation of 5/9/90.

24. In Reference 2, IELP committed to provide Category 1 instrumentation for this variable although RG 1.97 recommended Category 2. We have subsequently determined that this instrumentation does not currently meet seismic qualification criteria and recording of position indication is not provided in the Control Room. This instrumentation does, however, meet RG 1.97 Category 2 requirements. Therefore, we propose to provide Category 2 qualified instrumentation for this variable as recommended by RG 1.97.

25. In Reference 2, IELP committed to provide Category 1 instrumentation for this variable although RG 1.97 recommended Category 2. We have subsequently determined that certain of these instruments are not seismically qualified and other instruments are located in areas other than the Control Room (but within the Control Building envelope.) This instrumentation does, however, meet RG 1.97 Category 2 requirements. Therefore, we propose to provide Category 2 qualified instrumentation for this variable as recommended by RG 1.97.

26. The function of this variable is accomplished by laboratory analysis of samples. The NRC accepted this position in the Safety Evaluation of 5/9/90.

27. The function of this variable is accomplished by using portable equipment which satisfies the specified range. The NRC accepted this position in the Safety Evaluation of 5/9/90.

SUMMARY OF RG 1.97 CATEGORY REVISIONS

VARIABLE	VARIABLE TITLE	PREVIOUS IELP CATEGORY	RG 1.97 CATEGORY	NEW CATEGORY
D-09	Main Steam Isolation Valve Leakage Control System Pressure	1	2	2
D-10	Primary System Safety Relief Valve Position	1	2	2
D-13	Reactor Core Isolation Cooling Flow	1	2	2
D-14	High Pressure Coolant Injection Flow	1	2	2
D-15	Core Spray Flow	1	2	2
D-18	Standby Liquid Control System Tank Level	3	2	2
D-22	Emergency Service Water Flow	3	2	2
D-24	Emergency Ventilation Damper Position	1	2	2
D-25	Status of Standby Power and Other Power Sources Important to Safety	1	2	2

SCHEDULE FOR PROPOSED MODIFICATIONS

1) Incorporation of Primary Containment Isolation Valve Position Indication into DAEC Equipment Qualification Program

   Date: May 30, 1994

2) Upgrade HPCI and RCIC Turbine Inlet Pressure Instrumentation

   Date: Prior to Startup from the Cycle 13* Refueling Outage

3) Upgrade Power Supplies for Category 2 Instrumentation.

   Date: Prior to Startup from the Cycle 13* Refueling Outage

4) Modification of Yarway Reactor Level Instrumentation in Control Room Panels.

   Date: Prior to Startup from the Cycle 13* Refueling Outage

* Startup from the Cycle 13 Refueling Outage is currently scheduled for March 1995.