UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
COMMONWEALTH EDISON COMPANY	)	Docket Nos. 50-454
	)	50-455
(Byron Nuclear Power Station, Units 1 and 2)	)	

COMMONWEALTH EDISON COMPANY'S
RESPONSE TO THE LEAGUE OF WOMEN VOTERS' FIRST
INTERROGATORIES AND ACCOMPANYING REQUEST FOR DOCUMENTS

Pursuant to the Nuclear Regulatory Commission's Rules of Practice, and in accordance with the stipulation dated August 18, 1982 among all parties to this proceeding, Commonwealth Edison Company ("Edison") provides the following responses to "League of Women Voters of Rockford, Illinois' First Interrogatories to, and Accompanying Request for Documents from, Commonwealth Edison Co." Unless otherwise stated, documents which Edison has provided for the League of Women Voters' ("League") inspection are available to the League at the offices of Isham, Lincoln & Beale, Three First National Plaza, Suite 5200, Chicago, Illinois, 60602.

ANSWERS TO INTERROGATORIES

Interrogatory No. 1

Concerning Contention 1A:

D S02

(a) state specifically what Commonwealth Edison Company ("CECO") has done to evaluate and/or alter its generic QA/QC programs as used at its Byron plant ("Byron") in response to the proceedings regarding CECO's LaSalle Plant;

RESPONSE:

(a) As a result of the proceedings regarding the LaSalle Station, Edison is taking the following steps to evaluate generic QA/QC programs at Byron:

(1) An extensive audit of site contractors QC records is currently being performed to examine areas such as: accountability of records, completeness, alteration of records, correctness of data, uniqueness of individual reports, sequence of signatures, review of calibration by outside agencies, and corrective action taken for out-of-calibration equipment. The Byron documentation audit is estimated to take about three weeks and will involve five to seven auditors. During this time, it is expected that an estimated 8,000 records will be reviewed as part of this audit. It is likely that the Byron audit will be similar to the audits conducted at Braidwood and LaSalle; the Braidwood audit lasted three weeks, involved seven auditors, and a total of 8,466 documents were reviewed.

Normally, an on-site audit lasts three to five days, and it involves two to three auditors. Major audits conducted by site personnel last three to five days and involve approximately five to six auditors. The Byron documentation audit, however, will be more extensive in that it will take more time and involve a greater number of auditors.

(2) Edison has added an additional nine graduate engineers to its site QA organization to increase to its site QA organization. Also, four non-graduate technicians have been added to the office organizations to augment quality assurance surveillance and document review. In addition, six technicians have been provided by an independent testing contractor to perform special inspections of plant elements to verify that installations are in conformance with vendor and architect/engineer design documents.

(3) Edison's reviews of the contractors' QA/QC procedures have resulted in revisions to site procedures for Edison's heating, ventilation, and air conditioning ("HVAC") contractor.

Interrogatory No. 1

(b) state specifically what CECo has done to evaluate and/or alter its QA/QC program at Byron in response to the continuing IE reports citing QA/QC deficiencies at the Byron Plant:

RESPONSE:

(b) Edison evaluates and, if necessary, changes its QA/QC program as a result of deficiencies identified in

NRC I.E. reports. The specific action taken for identified deficiencies is described in detail in Edison's responses to the NRC reports. Each required response includes corrective action taken, action taken to prevent further noncompliance, and the date when full compliance will be achieved. In many of the responses, Edison committed to implementing programmatic changes to the QA/QC programs. These reports and Edison's responses are available for review by the League.

Interrogatory No. 1

(c) describe with particularity what has been done to re-evaluate the quality and conformance level of work performed under the QA/QC procedures which have subsequently been determined to be inadequate;

RESPONSE:

(c) If Edison determines QA/QC procedures are inadequate, it re-evaluates and revises the QA/QC program. Two examples of extensive re-evaluation and revision of QA/QC programs at Byron involve Hatfield Electric Company and Reliable Sheet Metal. The specific circumstances with respect to these matters are as follows:

(1) In January, 1981, a stopwork order was issued to Hatfield Electric Company as a result of identified deficiencies of site procedures. Procedures found to be inadequate relative to performance of QC inspection were revised to cure deficiencies. In addition, a complete reinspection

program was subsequently initiated for conduit and cable tray hangers. It is expected that this reinspection program will be completed by January 1, 1983 for Unit 1, and by March 1, 1983 for Unit 2.

(2) As a result of deficiencies identified in the Reliable Sheet Metal QA program, work on the HVAC system was stopped on September 17, 1982. The deficiencies involved inconsistencies between the contractor's inspection practices and generally accepted inspection guidelines. Procedures have been, or are being, revised to more clearly define inspection requirements. In addition, a backfit program has been established. This program will cover all safety related installations except for the documented hanger welding inspection performed and found acceptable by the on site independent testing inspection contractor. Completion of this inspection activity is projected for May 1, 1983.

Interrogatory No. 1

(d) identify and produce all documents relied upon in the preparation of the answers to parts (a), (b), and (c) of Interrogatory No. 1.

RESPONSE:

(d) 1. Letter from G. Sorenson to R. Bombach and R. Irish, dated September 17, 1982.

2. Letter from G. Sorenson to A. Domagale, R. Bombach, and R. Irish, dated September 22, 1982.

3. Letter from R. Irish to R. Tuetken, dated October 22, 1982.

4. Edison's Quality Assurance Manual Audit Report For Braidwood, dated August 4, 1982.

5. Edison's Quality Assurance Manual Audit Report For The LaSalle Station, dated August 4, 1982.

Interrogatory No. 2

Concerning Contention 8:

(a) (i) state whether CECO has compiled for Byron a list or lists of "important to safety" equipment as that term is defined in the November 20, 1981 Memorandum of Harold R. Dentor, "Standard Definitions for Commonly Used Safety Classification Terms." and, if the response is in the negative, state with particularity why no such list has been compiled;

(ii) if such a list of important to safety equipment has been compiled, produce the list(s) and state specifically for each piece of equipment included therein the criteria used to classify it as important to safety and what environment was assumed during that classification;

(iii) state whether all components of each item of equipment on the list of inportant to safety equipment also have been qualified as important to safety and, if so, the criteria used and what environment was assumed for the qualification of those components; and

(iv) state with specificity how equipment included in important to safety equipment lists differ from safety related equipment;

RESPONSE:

2(a)(i) Edison has not compiled a list of "important to safety" equipment. No such list has been compiled because no requirement exists to compile such a list and because Edison believes the classification of structures, systems, and components described in Section 3.2 of the FSAR is adequate. Table 3.2-1 lists all category 1 equipment which will be installed at Byron. This equipment is generally referred to as "safety related" or "safety grade," not "important to safety," equipment.

(ii) Not applicable.

(iii) Not applicable.

(iv) Not applicable.

Interrogatory No. 2

(b) state whether CECO has undertaken or is undertaking a site-specific probabilistic risk assessment ("PRA") or similar study or analysis for the Byron Plant to confirm the accuracy of any list of important to safety equipment or for any other purpose; if not, indicate whether or not CECO plans at any time to undertake such a study or analysis;

RESPONSE:

(b) Edison understands the "site-specific probabilistic risk assessment (PRA)" cited in the interrogatory to be the equivalent of a level 3 PRA (i.e. See NUREG/CR-2300 which describes and defines the various PRA levels) which would include: detailed event and fault trees specific to Byron; a Byron specific containment evaluation including event tree formulation, containment response, and sensitivity evaluations, and source term evaluations; and a site specific consequence analysis using CRAC, CRAC II, CRACIT, or equivalent codes. No such PRA has been undertaken and none is planned for Byron. In the course of preparing testimony, two independent evaluations are being prepared. One, by Edison, is complete in draft form and consists of hand calculations to rebaseline the Zion Probabilistic Safety Study for Byron. The second, by Westinghouse, is also not yet complete and will consist of computer assisted event and fault tree development and limited containment evaluations.

Interrogatory No. 2

(c) provide a copy of the PRA performed for CECO's Zion facility (which the Byron FSAR states is similar in design to Byron);

RESPONSE:

(c) A copy of the Zion Probabilistic Safety Study is available for examination by the League.

Interrogatory No. 2

(d) state with specificity each instance where a Byron PRA would differ from the Zion PRA and provide a listing of major differences between Byron and Zion which would affect PRA and risk assessment results, specifying in each instance the impact of the difference on the probability of accidents and radioactive releases and on the consequences of such accidents and/or releases;

RESPONSE:

(d) Due to the fact that a full level 3 PRA has not been conducted for Byron, it is not possible to detail the impact of each specific site and design difference between the two plants. The following are examples of obvious differences between the two plants which would probably have an impact on probabilistic results: 1) Byron will have four diesel generators, Zion has five; 2) Byron will have two containment spray trains per unit, Zion has three per unit; 3) The Byron service water is expected to withstand greater seismic induced accelleration; and 4) The Byron site has a significantly lower population density than does the Zion site. The draft evaluation identified in the response to interrogatory 2(b) attempts to identify those differences judged to be significant. It does not quantify the individual effects of those differences. Also, a level 3 PRA for Byron would most likely take into account reductions in the postulated source term from core melt accidents which derive from consideration of studies conducted following the TMI-2 accident, considerations of aerosel behavior, and other recent information. A Byron level 3 PRA might also draw on the future development of more realistic containment evaluation tools to remove unnecessary conservatism in this area.

<u>Interrogatory No. 2</u>

(e) state whether you agree that a Byron-specific PRA would be useful for the safe operation of the operation of the Byron Plant;

RESPONSE:

(e) Any reliable information, be it a PRA or other information, concerning the manner in which a nuclear power plant operates, can, in a sense, always be considered useful. However, at this point in time, there is sufficient information in which to conclude that Byron will operate safely. In addition, it is obvious that the development of a Byron level 3 PRA would involve a very substantial committment of time and resources. Accordingly, Edison does not believe that the diversion of such resources from other safety related activities is warranted. In that sense, Edison believes that a Byron level 3 PRA would not be useful.

Interrogatory No. 2

(f) state whether you agree that a Byron-specific PRA is necessary for the safe operation of the Byron Plant;

RESPONSE:

(f) No. See also response to 2(e) above.

Interrogatory No. 2

(g) state whether you agree that a Byron-specific PRA would be useful for understanding large accidents and their mitigation (including emergency preparedness) at Byron.

RESPONSE:

(g) For the reasons given in response to Interrogatory 2(e) above, Edison does not believe a Byron level 3 PRA would be useful for these purposes.

Interrogatory No. 2

(h) if your answer to (e), (f) or (g) is no, explain in detail the reasons for your answer, and if your answer to (b) is no, explain in detail why no Byron-specific PRA is contemplated;

RESPONSE:

(h) At this time, Edison does not plan to undertake a level 3 PRA for Byron because such a study is not necessary for the safe operation of Byron Station. Edison has concluded that such a study is not necessary for the safe operation of Byron Station on the basis of: 1) the extensive safety reviews of Byron performed by Edison, its contractors, and the NRC; 2) Byron's compliance with regulations, regulatory requirements, and industry codes and standards; 3) Edison's experience with the operation of other nuclear plants; and 4) the extensive experience and safety record of the nuclear industry at large.

Interrogatory No. 2

(i) identify and produce all documents relied upon in the preparation of your answers to Interrogatory No. 2.

RESPONSE:

(i) 1. Zion PRA

2. The draft evaluation prepared by Edison described in response to Interrogatory 2(b).

Interrogatory No. 3

Concerning Contention 19:

(a) state whether any Byron site-specific accident consequence model has been constructed and what computer program, if any, was used in its construction;

RESPONSE:

(a) Edison has not prepared a Byron site-specific accident consequence model. Edison is aware that the NRC Staff has employed such a model in its environmental impact evaluation of Byron. In addition, the Sandia Laboratories has apparently developed such a model. See also response to Interrogatory 15(b) infra.

Interrogatory No. 3

(b) provide a copy of the material used or to be used as input for construction of the Byron site-specific accident consequence and a copy of the field model;

RESPONSE:

(b) The material requested would have to be sought from the NRC Staff or Sandia Laboratories.

Interrogatory No. 3

(c) describe with particularity the dates, locations, scope, and subsequent evaluations of any off-site emergency drills conducted or planned to be conducted in relation to Byron and indicate with specificity any differences between the Byron drills conducted or planned and drills previously carried out at the Zion facility;

RESPONSE:

(c) A Byron off-site emergency drill currently is scheduled by Edison for May 11, 1983. Since the Byron drill will not be conducted for at least six months, it is not possible, at this time, to identify specific differences between the Byron and Zion exercises.

Interrogatory No. 3

(d) identify and produce all documents relied upon or referred to in your answers to parts (a), (b), and (c) of this Interrogatory; and

RESPONSE:

(d) 1. Sandia Laboratory report, NUREG/CR-2239 "Technical Guidance for Siting Criteria Development" November, 1982.

2. Letter from J. Hind to C. Reed, dated May 12, 1982.

Interrogatory No. 3

(e) provide a citation or citations to the document or documents where the State of Illinois has designated 10 miles as the radius of the Low Population Zone and 50 miles for the radius of the Emergency Planning Zone, and also where the State has discussed the consideration involved in and/or the reasons for so designating those radii.

RESPONSE:

(e) The generic Illinois Plan for Radiological Accidents provides for a 10 mile plume exposure pathway EPZ and a 50 mile ingestion pathway EPZ consistent with the federal guidance set forth in NUREG-0396 "Planning Bases for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants" and 10 CFR §50.54(s)(1). The site-specific Byron plan currently is being developed.

Edison believes that the site-specific plan will establish a plume exposure EPZ of approximately 10 miles and an ingestion exposure EPZ of approximately 50 miles.

Interrogatory No. 4

Concerning Contention 22:

(a) state what specific measures are currently being taken or are expected to be taken by CECO to prevent or inhibit the process of steam generator tube degradation, which causes include but are not limited to flow-induced vibration in the preheater section, and state for each such measure whether (and, if so, how) differs from measures previously adopted at other nuclear plants;

RESPONSE:

(a) The various phenomena that can cause steam generator tube degradation may be catagorized as follows:

1. Intergranular corrosion.
2. Denting
3. Thinning or wastage
4. Pitting
5. Wear

The first four of these degradation phenomena are caused by corrosion which results from the ingress of impurities into the steam generator. Each of these has been addressed in the design or construction of Byron Station or will be addressed in the implementation of the Byron Station secondary water chemistry control program. The fifth item will be addressed as a separate issue.

Typically, intergranular corrosion (IGC) occurs in regions of the steam generator where highly caustic copper bearing alloys in the condenser and feed water heater system. This phenomenon occurs at the intersection of the Inconel 600 tubes and the carbon steel support plates. Corrosion products typically will plug the crevice between the support plate and the tube. If water chemistry is not carefully controlled, chloride ions enter the porous corrosion product deposit and are concentrated to the point where corrosion of the carbon steel support plate occurs. Since the corrosion product has a lower density than the carbon steel that has been corroded away, the crevice between the tube and support plate is closed. Continuation of the corrosion results in compressive forces being applied to the tube which then dent and leads to potential tube leakage.

At Byron Station, this problem has been addressed in two ways. First, copper alloys have been eliminated from the secondary system. The removal of copper removes the catalyst for the corrosion in the tube support plate crevice. Second, the Byron chemistry control program is designed specifically to minimize the ingress of contaminants into the steam generator. The result of this action will be to eliminate corrosion in the crevices. This has been found to be effective at several other plants.

Thinning or wastage of Inconal 600 Tubing has been observed in plants as the result of long-term sodium phosphate treatment. This was typically associated with a high phosphate sludge pile in older steam generators where environments can develop. In older steam generators, an 18" deep crevice existed between the Inconel 600 tube wall and the tube sheet. This crevice was extended in some steam generators by the accumulation of corrosion products (sludge) and sodium phosphate on top of the tube sheet. Under some conditions, chemicals could enter this crevice. Boiling could concentrate these chemicals and form a highly caustic solution that could then attack the Inconel 600 tubing. Pre-1975 IGC has been eliminated by changing from a phosphate to an all volatile water treatment and by periodic "sludge lancing."

Even with these two changes, IGC was observed again in 1980 at some plants. This recent occurance is due to the long term concentration of caustic material in the tubesheet crevice.

At Byron, this concern has been eliminated by expanding the tube so that it is in intimate contact with the full depth of the tube sheet. This process is called "full depth tube rolling" and eliminates the tube sheet crevice which in turn eliminates the existance of an area in which chemicals can concentrate and can ultimately cause corrosion. In addition, sludge will be removed periodically by "sludge lancing."

Denting is a phenomenon which has occured at many plants as a result of the ingress of contaminants such as chloride and is accelerated by the presence of copper deposits transported to the steam generator from a synergistic effect between the sludge and phosphate could result in wastage of the Inconal 600 tube wall. When the all volatile chemical treatment (A.V.T.) has been used, however, this form of tube degradation has not occured. At Byron Station, Edison will be using the all volatile chemical treatment, and therefore, thinning or wastage of Inconel 600 is not a concern.

The fourth type of tube degradation is pitting. This is a relatively recently observed phenomenon at seawater cooled plants that have had significant ingress of high chloride cooling water into the steam generators. This phenomenon is caused by the concentration of chloride ions under a deposit on the tube walls and is associated with copper deposited on the tube walls. An electro chemical cell is set up and the Inconel 600 Tubing is attacked locally which forms a pit. This type of tube degradation has only been observed at two plants. Both plants are seawater cooled and have other problems, such as denting, associated with chloride intrusion. In addition, these plants have significant amounts of copper deposited in the steam generators. At Byron Station this degradation phenomenon should not be a problem because copper alloys have been eliminated from the secondary system and because the secondary water chemistry control program is designed to control ingress of contaminants. Byron station has an additional advantage in that it is not seawater cooled.

The fifth degradation phenomenon is wear of the steam generator tubes as a result of flow induced vibration and the rubbing of the tube against the tube support plates or as a result of a foreign object rubbing against the tube. This results in the thinning of the tube wall. Administrative controls will be used at Byron Station to exclude foreign objects from the steam generators.

Flow induced vibration appears to be a mechanical phenomenon related to the design of the steam generators. To date it has been observed in the Westinghouse Model D2, D3 and D4 designs. An extensive testing program is underway and a solution has been proposed for the model D2 and D3 type steam generators. Byron Station Unit 1 has a Model D4, counterflow design, preheater section. It is likely, due to design similarities, that the phenomenon will also be observed in Model D5 steam generators. No Model D5's are yet in operation. To date only one plant using Model D4 steam generators is in operation. That unit currently is operating in Yugoslavia, and Westinghouse is conducting a test program to evaluate the extent of vibration in the preheater. The results of this evaluation will be used by Westinghouse to determine the extent of wear and, if necessary, to design a modification for the Byron steam generators.

Interrogatory No. 4

(b) for each of the accident scenarios which have been postulated as applicable to Byron, describe with particularity what radioactive material

> would be released by a steam generator tube failure, the form in which it would be released, and in what possible pathways it would be released;

RESPONSE:

(b) The only postulated design basis accident scenario involving a steam generator tube failure that is applicable to Byron is the "Steam Generator Tube Rupture" event discussed in Section 15.6.3 of the Byron FSAR. The radioactive material that would be released by a steam generator tube rupture, the form in which it would be released, and the possible pathways by which it would be released, are all described in detail in Sections 15.6.3.1, 15.6.3.2, and 15.6.3.3 of the FSAR.

<u>Interrogatory No. 4</u>

> (c) provide copies of all operating procedures concerning steam generators, water quality and chemistry control and any other operating procedures which are significant to the control of the operation of the steam generators within the design limitations, including but not limited to pressure, temperature, fatigue and corrosive limits, and if any of the above procedures are not yet available but are expected to be produced prior to operation of the Byron Plant, provide the titles of these procedures;

RESPONSE:

(c) The following is a listing of the Byron Operating and Chemistry Procedures or documents which concern the steam generators, their water quality, and chemistry control. For each procedure, the status of its development is provided. Procedures listed as "identified" have not been drafted. The titles of the procedures are as follows:

1. Operating Procedures

Procedure Number	Procedure Name	Approved	In Draft	Identified
BOA SEC-8	S/G Hi Conductivity/Operating Limitations			X
BOG CD-3	Placing S/G in Wet Lay-up	X		
BOP PS-3	S/G Blowdown Sample			X
BOP PS-5	S/G Monitoring			X
BOP SD-1	Start-up of S/G Blowdown			X
BOP SD-2	Shutdown of S/G Blowdown			X
BOP AF-3	Placing the S/G in Wet Lay-up	X		
BOP CF-8	Main Feed System Chemical Feed			X
BOS 7.2.1-1	S/G Press/Temp Limitations with Primary or Secondary Coolant less than 70' F			X
BOS 4.5.0-1a	S/G Inoperable			X
BOS 4.7-1a	RCS Chemistry Surveillance			X
BOS 7.2.1-1a	S/G Press/Temp Limit Exceeded			X
BVS 4.5.0-1	ASME Surv. Requirements for S/G's			X
BVS 4.5.1-1	S/G Inspection Shutdown			X
BVS 4.5.2-1	S/G Tube Inspection			X
BVS 4.5.3-1	S/G Inspection Frequencies			X
BVS 4.10-11	S/G Eddy Current Examination			X

The following are titles of operating procedures which will be written following the submittal of the Westinghouse recommendations for the D-Model Steam Generators and other operating procedures which will be developed for the condensate polisher system.

2. Chemistry Program Descriptions

Procedure Number	Procedure Name	Approved	In Draft	Identified
BPD 100-3	Flushing		X	
BPD 100-4	Secondary Chemistry Monitoring		X	
BPD 100-5	Hot Functionals		X	
BPD 100-7	Circulating Water Chemistry			X
BPD 100-8	Failed Fuel			X
BPD 200-1	Quality Control		X	
BPD 200-7	Data Management		X	
BPD 300-3	NRC Requirements			X

3. Chemistry System Descriptions

Procedure Number	Procedure Name	Approved	In Draft	Identified
BCD 200-1	Condensate		X	
BCD 200-2	Feedwater		X	
BCD 200-3	Heater Drain		X	
BCD 200-4	Main Steam		X	
BCD 200-5	S/G Blowdown		X	
BCD 200-6	Condensate Polishing			X
BCD 300-2	Auxiliary Feedwater		X	
BCD 300-3	Auxiliary Steam		X	
BCD 300-5	Chemical Feed		X	
BCD 300-7	Circulating Water		X	
BCD 300-10	Make-up Demineralizers		X	
BCD 300-14	Process Sampling			X

4. Chemistry Procedures

Procedure Number	Procedure Name	Approved	In Draft	Identified
BCP 300-9	S/G Tube Leak Detection		X	
BCP 300-10	Secondary System Air Inleakage			X
BCP 300-11	Condenser Tube Leak Detection			X
BCP 400-T22	Chemical Addition Log	X		
BCP 400-T35	Chemical Addition to the Secondary Side			X
BCP 400-T37	Secondary Side Chemistry Data			X
BCP 700-1	Limitations and Actions			X

Interrogatory No. 4

(d) describe in detail the design features and material specified for the steam generators at Byron Units 1 and 2, including but not limited to the differences, if any, in components for use at Unit 1 and Unit 2 and the reasons for these differences, and provide a list of other U.S. nuclear units furnished by Westinghouse which utilize the same steam generator designs as are found at Byron; if exact duplicates do not exist, identify which plants utilize the individual design and material features employed at Byron;

RESPONSE:

(d) The Byron steam generators are described in Section 5.4.2 of the Byron/Braidwood FSAR. Steam generator materials are discussed in Section 5.4.2.1. The design bases are described in Section 5.4.2.3, and the design description of the Byron steam generators is presented in Section 5.4.2.4 of the FSAR.

Byron Unit 1 employs Westinghouse Model D4 steam generators. Westinghouse Model D5 steam generators are employed in Byron Unit 2. The Model D4 and D5 steam generators are very similar in design. A Model D5 steam generator is essentially a Model D4 generator which incorporates the following additional design features:

1. The steam generator tubes employed in the Model D5 are given a heat treatment during manufacture to relieve surface stresses and enhance their corrosion resistance.

2. The material of the tube support and baffle plates was changed from carbon steel to stainless steel to provide additional corrosion resistance. In conjunction with this change, the thickness of the tube support plate was increased and additional stay rods, which interconnect the baffle plates, were added in the preheater section because of the differences in allowable stresses and geometry associated with the change.

3. The tube support plate tube holes utilize a multi-lobed configuration to permit the boiling water to sweep the tube surface and therefore reduce the potential for corrosion.

4. The flow distribution baffle holes are slightly enlarged over those for the D4, which provides some additional flow between the baffles below the preheater section. The flow distribution baffle is slightly higher in order to maintain appropriate flow distribution.

5. The number of swirl vanes at the top of the steam generator was increased from 12 to 16 to add additional moisture separation capacity.

U.S. nuclear power plants employing Westinghouse Model D4 and D5 steam generators include Byron Units 1 and 2, Braidwood Units 1 and 2, Shearon Harris Units 1 and 2, Commanche Peak Units 1 and 2, Catawba Unit 2, and Marble Hill Units 1 and 2.

Interrogatory No. 4

(e) provide copies of procedures and/or specifications pertaining to the in-service inspection of the steam generators, including but not limited to procedures and/or specifications related to the maintenance of occupational radiation exposure ALARA;

RESPONSE:

(e) Byron Radiation Procedure (BRP) 1620-2 "Radiological Controls for Steam Generator Work" will be written to describe the specific radiation protection measures to be taken for in-service inspection of the steam generators. Examples of some of the specific radiation protection measures that will be described in this procedure are: ventilation requirements, shielding, protective clothing, respiratory protection, radiation surveys, and air sampling schedule. In addition to BRP 1620-2, the Byron Station ALARA Program will be utilized to assess any ALARA considerations for in-service inspections of the steam generators. The Byron Station ALARA Program consists of Byron Administrative Procedures (BAP) 700-1 "ALARA Program, 700-2 "ALARA Review", 700-A1 "Appendix - ALARA Review Requirements", 700-A2 "Appendix - Guide for Dose Reduction Effort", 700-T1 "ALARA Review", 700-T2 "ALARA Review Log, 700-T3 "ALARA Action Review Follow up", and Commonwealth Edison Co. Radiation Evaluation Program, Dosimetry Program, and Cost/Benefit Program will be used prior to the start of an in-service inspection on the steam generators to review, document, and implement reasons necessary to maintain radiation exposures ALARA for in-service inspections of the steam generators.

The letter from Tom Tramm to the NRC dated April 27, 1982, describes the plan for inservice inspection of the steam generators.

Interrogatory No. 4

(f) provide copies of any reports available to CECO concerning results of generic studies of steam generator problems conducted by or for CECO, EPRI, the NRC, National Laboratories, other utility groups, consultants, any other entity, group or individual, and if such reports contain recommendations for changes or provisions that could be implemented at the Byron Plant, provide a description of CECO's evaluation of such recommendations and whether or not they have been or are being implemented at Byron and indicate with specificity the reasons for CECO's response to that evaluation;

RESPONSE:

(f) The following is a list of reports available*/ that concern studies of steam generator problems:

C-E/EPRI Program RP-623-1, PWR Model U Tubed Steam Generator Corrosion Studies, Fourth Progress Report, July, 1979

EPRI S146-1, Topical Report - Estimation of Diffusion Coefficient for Electrolytes in Hot Water-January, 1980 (Westinghouse)

Laboratory Studies of the Effectiveness of Boric Adid in Dent Inhibition, WCAP 9676 - February, 1980 (Westinghouse)

Background Report on Qualification of Hydrogen Monitoring as a Measure of Steam Generator Corrosion, August, 1976 -December, 1978, (S117-1, October, 1980) (Westinghouse)

Third Quarterly Progress Report for S146, Period of July 1 -September 30, 1979, Diffusion and Hideout in Crevices, Westinghouse, Feb., 1980

EPRI Third Quarterly Progress Report for period covering April, 1979 through June 30, 1979 on "Neutralization of Crevice Acids RP623-2 dated March, 1980. (Combustion Engineering Inc.)

EPRI Progress Report for Period Covering July to August, 1979 on "Indian Point Unit #3 - Part 1, Hydrogen Monitoring Program S116-1 Tash 400. (Westinghouse, 8/79)

Improvements of tightness of condensers for Seawater Cooled Plants - New Techniques.

---

*/ The availability of these documents is subject to agreement by the League which adequately protects any proprietary information which is contained in the documents.

Thermal Hydraulic Characteristics of a Combustion Engineering Series 67 Steam Generator - EPRI Volume 1 - April, 1980

Examination of Steam Generator Tube R45C52 from the Ginna Nuclear Power Plant - Final Report - May, 1980

Steam Generator Crevice Gap Measurement by Induced - Vibration Analysis Final Report - May, 1980. (EPRI)

Effect of Out-of Plane Denting Loads on the Structural Integrity of Steam Generator Internals - EPRI Project S-169-1 Final Report August, 1980

Thermal Hydraulic Analysis of Once-Through Steam Generators. EPRI Project S-131-1, Final Report, June, 1980

Third Quarterly Progress Report on "Optimization of Metallurgical Variables to Improve Stress Corrosion Cracking Resistance of Incone 600" March, 1980 - RP1708. July, 1980 (Westinghouse)

Westinghouse Background Report "A Summary of Westinghouse Experience in Treating Steam Generator with Boric Acid (WCAP - 9682) dated December, 1979

EPRI Third Progress Report for SGOG Project S148-1, Non Proprietary Corrosion Inhibitors for Solvents to Clean Steam generators - September 15 - December 15, 1979

EPRI PWR Steam Side Chemistry Program 6th Quarterly Report, December 1, 1979 to February 28, 1979 RP-699-1 (Westinghouse)

EPRI Third Quarterly Report, "Alternative Steam Generators Materials and Designs", RP 623-4. (Combustion Engineering Inc.)

EPRI Second Quarterly Progress Report on "Omptimization of Metallurgical Variables to Improve the Stress Corrosion Cracking Resistance of Inconel 600 - December, 1979 RP 1708 (April, 1980) (Westinghouse)

EPRI Third Progress Report - (April 1 to June 30, 1979) Evaluation of Condensate Polishing to Steam Generator Corrosion, March, 1980 RP 623-3. (Combustion Engineering Inc.)

EPRI Report on Project S127-1, Chemical Cleaning of Nuclear Steam Generators covering the period of April, 1978 - Jan., 1979. (Babcock & Wilcox)

EPRI Project S149-1, Cleaning Steam Generators off-line (Soaking) with Chelants. (Westinghouse)

EPRI Project RP404-1, PWR Secondary Water Chemistry study covering period of January through June, 1979. (NWT Corp)

EPRI S146-1, 1st and 2nd Quarterly Progress reports - Diffusion and Hideout in Crevices. (Westinghouse)

EPRI R.P. 699-1. Progress Report and the 5th Quarterly Report of "PWR Steam Side Chemistry Follow Program" (Westinghouse)

EPRI Task Agreement S112-1 "Laboratory Studies related to steam Generator Tube Denting," Fourth & Fifth Progress Report (Westinghouse)

Sixth Quarterly Report - EPRI Project 2L21-1 "Tube Support Plate Crevice Blockage and Thermal/Hydraulic Test Program" (Combustion Engineering, Inc)

EPRI RP1171-1 Task 6 Report The Stress Corrosion Cracking of Inconel 600 in Acidic Sulphate Solutions at 289 C. (Central Electricity Research Laboratories, 12/79)

EPRI S139-1-2nd, 3rd & 4th Quarterly Reports, Instrumentation of Steam Separators.

Sundesert Nuclear Plant Availability Improvement Design Program

NUREG/CR-0718 "Steam Generatory Tube Integrity Program, Phase I report, September, 1979

Steam Generator Owner's Group Program 7/14/78

TMI-2 OTSG EDDY Current Inspection Results

EPRI Review and Evaluation of Existing Chemistry Models Feb. 5, 1982

Induced - Vibration Analysis Probe for Measurement of Steam generatory Tube to Support Plate Clearance Final Report -EPRI February, 1982

PWR Steam Generator Cost - Benefit Methodology - Final Report, March, 1982 (EPRI)

Evaluation of Pulse - Echo Ultrasound for Steam Generator Tube-to-Support Plate Gap Measurement - EPRI Final Report, March, 1982

Secondary Water Chemistry at Rhingals Unit 2 - EPRI Project S170-1 Topical Report February, 1982

Effect of changing the Hydrazine Injection Point at the Caroline PWR & St. H.B. Ronbinson Plant - Final Report February, 1982 (CPRl)

EPRI Steam Generatory Chemistry & Materials Program Assessment and Plan, December 1, 1979

Optimization of Metallurgical Variables to Improve Stress Corrosion Resistance of Inconel 600 EPRi Project 621-1 Final Report - March, 1980

Corrosion Related Failures in Power Plant Condensers TPS-79-730, EPRI Final Report August, 1980

Laboratory Studies Related to Steam Generator Tube Denting -6th & 7th Quarterly Report - From July 24, 1979 to Jan. 24, 1980 (Westinghouse)

Evaluation of Alternate Allows for Tubing Steam Generators -2nd Annual Report - EPRI RP1450 August 1, 1980. (NCO Research & Development Center, Inc)

PWR Steam Side Chemistry Program 7th - 9th Quarterly Report -March 1 to November 30, 1979. (EPRI)

Fatique of Inconel 600 Under Typical Steam Generatory Conditions 1st Quarterly Report - EPRI Project S110-1

Westinghouse Performance Evaluation of the (Kent-Cambridge) Hydrogen Analyzers WCAP-9694, S117, June, 1980 and October, 1980

Metallurgical Characterization Studies of Nuclear Steam Generatory Tube Sections Removed from Point Beach 1 in October, 1979, EPRI Contract S.138-1 February 25, 1981 (Westinghouse)

Prevention of Condenser Failures - State of Art - Jan. 21-22, 1981 EPRI Seminar. (IVAR MULTER, SWEDISH STATE POWER BOARD)

PGE Report to NRC on Assessment of U-Bench Removed from Trojan Steam Generatory

Application of an Eddy Current Technique to Steam Generator U = Bend Characterization, Final Report April, 1982

Steam Plant Surface Condenser Leakage Study Update, Final Report May, 1982

HITCH Computer Code; Chemistry and PH Estimates of Concentrating Aqueous Solutions

Improving Oxygen and Hydrazine Monitoring for Control Oxygen in the Secondary System of Nuclear Power Plants. WCAP-9957

Optimization of Metallurgical Variables to improve the Stress Corrosion Cracking Resistance of Inconel Alloy 600, Tenth Quarterly Report, PP1708 April 6, 1982

Workshop Proceedings - U - Bend Tube Cracking in Steam Generator Proceedings June, 1981

Workshop Proceedings - Corrosion of Inconel 600 Steam Generator Tubing in Tubesheet Crevice Proceedings, May, 1981

The Design and Construction of Model Steam Generator for Corrosion Testing of Alternative Materials Project 623-4-Topical Report, August, 1981

The Role of Exygen as related to Steam Generator Tube Denting WCAP-9943, Project S109-1

EPRI 1st Progress Report-Evaluation of Condensate Polishing Relative to Steam Generator Corrosion. (Combustion Engineering)

EPRI Second Progress Report-Evaluation of Condensate Polishing Relative to Steam Generator Corrosion. (Combustion Engineering)

Instrumentation of Steam Generators Combines, 2nd, 3rd and 4th Quarterly Reports. EPRI S139-1, Combustion Engineering Inc. November, 1979

Rationale for Chemical Control of Feed and Boiler-Task 2, Denting Corrosion of PWR Heat Exchangers: The Generation of Acid Solutions in the Tube/Tube Plate Crevise by the Action of Dissolved Oxygen. (Central Electricity Research Labs, July, 1979)

Rationale for Chemical Control of Feed and Boiler Water-Task 4, the pH of PWR Steam Generator Water under various Fault Conditions. (Central Electricity Research Labs, July, 1979)

Laboratory Studies related to Steam Generator tube denting. EPRl-Contract S-112-1, Third quarterly report. (Westinghouse)

Second Progress Report for Steam Generator Owners Group EPRl Project S148-1, Non-proprietary inhibitors for Solvents to clean Steam Generators. (Petroline Corp)

1st Progress Report for Steam Generator Owners Group Project EPRl S148-1, Non-proprietary Corrosion Inhibitors for solvents to clean Steam Generators. (Petroline Corp)

First and Second Progress Reports for Steam Generator Owners Group. EPRl Project S-128-1, Steam Generator Chemical Cleaning: Development on Qualification of Process for Current and new units. (Combustion Engineering Inc.)

Steam Generator Sludge Pile Model Boiler Testing. EPRl Project S119-1 Final Report, July, 1981

Optical Scanner System for Internal Inspection of Steam Generator Tubes. EPRl Project S103-2, Final Report, July, 1981

Fatigue of Inconel 600 under typical Steam Generator Coditions. (Westinghouse EPR1 Project S110, 2nd Quarterly Report October-December 1980

Estimation of Diffusion Coefficients for Electrolytes in Hot Water. EPR1 Project S146-1 Topical Report-August 1981

Steam Generator Chemical Cleaning: Demonstration Test #2 in a Pot Boiler. EPR1 Project S128-1, Topical Report, August 1981

Chemical Cleaning Demonstration Test #1 in a Mark-up Steam Generator. EPR1 Project S127, Topical Report, April 1981

Demonstration Test on PWR Steam Generator Tube-Tubesheet Crevice Flushing Procedures. EPR1 Project S183-1, Final Report, May 1981

Optimization of Metallurgical Variables to Improve the Stress Corrosion Cracking Resistance of Inconel Alloy 600 4th, 5th, and 6th, Quarterly Progress Reports. June-Dec. 1980, RP1708. (Westinghouse)

Steam Generator Chemical Cleaning Demonstration Test #1 in a Pot Boiler. EPR1 Project S-128, Topical Report, April 1981

Flow-Induced Vibration Analysis of Three Mile Island Unit #2 Once Through Steam Generator Tubes, Volume 1 EPR1 Project S140-1, Final Report, June 1981

Review of the Hydrazine/Oxygen Reaction Kinetics - EPR1 Project S-117-1, May 1979 and April 1981 (Westinghouse)

Hydrogen Monitoring Development Hydrazine Decomposition Contribution to Total Hydrogen Production. EPR1 Program S117-1, Task 200 Feb. 1981 (Westinghouse)

Cleaning Steam Generators Off-line (Soaking) with Chelants. Project S140-1, June 19, 1981. (Westinghouse)

Evaluation of Condensate Polishers for the Control of PWR Steam Generator Crevice Corrosion. August, 1979 through July, 1988. (NWT Corp)

Primary to Secondary Hydrogen Permiation - Project S117, Feb., 1981 (Westinghouse)

Mechanisms of linaer growth of Oxide Scales on Chromium Alloy Steels - 1st Progress Report RP-1171-2 October 2, 1981. (Central Electricity Research Labs)

Steam Generator Mock-Up Facilities, EPR1 Research Projects S126, 1172, Final Report April, 1981

Profilimetry for Steam Generator Tube Dent Characterization EPRl Project S-108-1 Final Report November, 1981

Corrosion - Product transport in PWR Secondary Systems - EPRl Project 404-1 704-1, December, 1981

Optimization of metallurgical variables to improve the Stress Corrosion Resistance of Inconel 600, Eighth Quarterly Report, June, 1981. EPRl RP 1708-1 (Westinghouse)

Evaluation of Alternate Alloys for PWR Steam Generators, Third & Fourth Semi-Annual Report August, 1981, July, 1981. EPRl RP 1450-1 (INCO Development)

Cleaning Steam Generator Off Line Soaking with Chelants. Tasks 300 & 400 April, 1981 (Westinghouse)

Evaluation of Condensate Polishers for the Control of PWR Steam Generator Crevice corrosion. August through December, 1980 (NWT Corporation)

Visual Inspection equipment for the Secondary Side of Steam Generators. EPRl Project S155-1, Final Report, May, 1981

Mathematical Modeling of Tube Bundle Flow During Steam Generator Wil Layup.
Project S164-1, Final Report, April, 1981

Nuclear Steam Generator Tubes form the R.G.E. Ginna Station - A detailed Metallographic and Microanalytical Evaluation.
Project S138-2, Topical Report, January, 1981 (Westinghouse)

Metal Cation Inhibitors for Controlling Denting Corrosion in Steam Generators. EPRl Project S147-1, May 6, 1981 (Center for Surface & Coatings Research)

Diffusion and Hideout in Crevices, 4th and 5th Quarterly Report, October to March, 1980 (Westinghouse)
EPRl Project S146-1

Indian Point Unit #3 - Hydrogen Monitoring Program II June/September, 1980

Low Power Boric Acid Conditioning at Indian Point #3

Indian Point #3 - Steam Generator Chemical Return Study - September 30 - October 3, 1980

Microscopic Exam of Point Beach #2 Steam Generator Tube A(18-37) removed in April, 1980

Optimization of Mitallurgical Variables to improve stress corrosion resistance of Inconel 600

Tests of Isothermal Soaking Procedures for limiting tube denting in Steam Generators April, 1981

7th Quarterly Report - Optimization of Metallurgical Variables to improve Stress Corrosion cracking resistance of Inconel Alloy 600. (January-March, 1981) RP 1708.

4th Quarterly Report - R. P. 699-1

Neutralization of Crevice Acids - EPR1 Contract RP-623-2 Second Progress Report

Neutralization of Crevice Acids-EPR1 Contract RP-623-2 1st Progress Report

Determination and Verification of required Water Chemistry & Limits Task Agreement S111-1 Third and Fourth Quarterly Progress Reports

Evaluation of Steam Generator U-Bend Tubes from Trojan Nuclear Power Plant, final report. EPR1 Project S-138-4 Sept., 1982. (Westinghouse)

Condenser Inleakage Monitoring System Development, Final Report EPR1 project S182-1 (NWT Corp., Sept. 1982)

Optimism of Metallurgical Variables to Improve Stress Corrosion Resistance of Inconel 600, 12th Quarterly Report, April-June, 1982 (Westinghouse)

Alloy Steel Corrosion Kinetics and Oxide Morphologies In Acid Chloride Environments. EPR1 Contact RP 1171-2 (Central Electricity Research Laboratories)

Description of Westinghouse Model Boiler #2; Test facility and modeling of MB2 with Athos-2

Boiling Heat Transfer in Narrow Eccentric Annulus. EPR1 project S133-1 final report (Northwestern University)

PWR Power Plant Pump Reliability Data Interim Report, Sept. 1982 (Science Applications, Inc)

Topical Report, EPR1 Project S116-1, "Indian Point Unit 3, Hydrogen Monitoring Program, Part III" (Westinghouse, Jan., 1982)

Topical Report, Project RP623-2 "Test of On-Line Additions of Calcium Hydroxide for limiting tube denting in Nuclear Steam Generators, Dec., 1981 (Combustion Engineering Inc.)

Effect of Sodium Hydroxide on the Corrosion of Steam Generator Materials under high temperature Heat Transfer conditions. (Combustion Engineering Inc)

The Effect of Sodium silicate on the Corrosion of Steam Generator Materials Under High Temperature Heat Transfer Conditions (Combustion Engineering Inc.)

The Effect of Sulfuric Acid on the Corrosion of Steam Generator Materials Under High Temperature Heat Transfer Conditions (Combustion Engineering Inc.)

The Effect of Resin Ingress on the Corrosion of Steam Generator Materials Under High Temperature Heat Transfer Conditions (Combustion Engineering, Inc.)

TMI-1 OTSG Failure Analysis Report, July, 1982 GPU Nuclear Corporation

Topical Report, Cleaning Steam Generators off-line (soaking) with Chelants Task 200 Westinghouse December, 1981

Third Quarterly Progress Report for EPRl Project S110, "Fatigue of Inconel 600 under typical steam generator conditions", January-March, 1981 (Westinghouse)

Topical Report "Tubesheet Joint Thermal and Hydraulic Testing" WCAP 10026 (Westinghouse)

Topical Report "Effect of Moisture Separator Drain Routing on OTSG Secondary System Chemistry" EPRI Project 704-1 (NWT Corp)

Final Report, PWR Steam Side Chemistry Follow Program EPRI Project 699-1 (Westinghouse Corp)

Examination of Steam Generator Tube A(18-37) from the Point-Beach Unit 2 Nuclear Power Plant, EPRI Project S138-3 Final Report August, 1982 (Westinghouse)

Examination of Three Steam Generator Tubes from the Ginna Nuclear Power Plant. EPRI Project S138-2. Final Report. August, 1982 (Westinghouse)

CALIPSOS Code Report, Vol. 1, 2 EPRI NP-1391 Interim Report, April, 1982

Dynamic Thermal-Hydraulic Behavior of PWR U-Tube Steam Generators Simulation Experiments & Analysis EPRI Special Report, May, 1981

Evaluation of Oconee Steam Generator Debris, EPRI NP-2082 final report, October, 1981 B&W

Improvement to the COBRA-TF (EPRI) Computer Code for Steam Generator Analysis Final Report, EPRI RP1121-1 (Batelle, Pacific Northwest Laboratories)

Loss of Feedwater Transients in PWR U-tube steam generators: simulation experiments & analysis EPRI-1367 (Special Report)

Magnetic Flux Leakage for Measurement Crevice Gap Clearance and tube support plate inspection, EPRI S126-1 Final Report Colorado State University

Topical Report: Model of Vaporous Carry-over, April, 1981 (NWT Corp.)

Topical Report "Model of Vaporous Carry-over" EPRI RP 704-1, April, 1981 NWT Corp.

Topical Report "Modeling of Cooling Water Inleakage Effects in PWR Steam Generators", EPRI RP404-1 April, 1981 NWT Corp.

Topical Report, PWR FLECHT SEASET Steam Generator Separate Effects Task EPRI RP959-1 WCAP 9724 Westinghouse Corp.

Review of Transient Modeling of Steam Generator Units in Nuclear Power Plants, EPRI RP684-1 Interim Report, October, 1980 University of Michigan

Simplified Models for transient analysis of Nuclear Steam Generators, EPRI RP684-1 Interim Report. April, 1981 University of Michigan

Single-tube Thermal & Hydraulic Tube Support Test, Final REport Vols. 1&2 EPRI RPS118-1 Sept. 1981 Westinghouse Corp.

Stress Corrosion Cracking of Alloy 600 Special EPRI Report, November, 1981

Thermal-Hydraulic Analysis of the Combustion Engineering Series 67 Steam Generator, Final Report, January, 1981 EPRI RPS130-1 Atomic Energy of Canada, Ltd.

Thermal-Hydraulic Analysis of the Combustion Engineering System 80 Steam Generator, Final Report, Sept., 1980 EPRI RPS-130-1 Atomic Energy of Canada, Ltd.

Thermal-Hydraulic Characteristics of a Combustion Engineering System 80 Steam Generator, Interim Report, Vol. 1&2, Sept. 1980 EPRI RPS129-1 Combustion Engineering

Thermal-Hydraulic Characteristics of a Westinghouse Model F Steam Generator, Interim Report Vol. 1, March, 1981 EPRI RPS129-1 Combustion Engineering

Thermal-Hydraulic Characteristics of a Westinghouse Model 51 Steam Generator, Interim Report, Vol. 1 March, 1981 EPRI RPS129-1 Combustion Engineering

Transient Modeling of Steam Generator Units in Nuclear Power Plants: Computer Code TRANSG-01, Interim Report, March, 1980 EPRI RP684-1 University of Michigan

The URSULA2 Computer Program, Vols. 1-4 Final Report, January, 1980 EPRI RP1066-1; CHAM of North America Incorporated

Evaluation of Steam Generator Tube 85-127 from Oconee 1B, Final Report, April, 1981 EPRI RPS136-1 Babcock & Wilcox Co.

Development of Sensors of Instrumentation for the TMI-2 OTSG Tube Vibration Measurements Program Topical Report, June, 1981 EPRI RPS140-1 Babcock & Wilcox

Evaluation of Secondary System Oxygen Control in PWR power plants Final Report, June, 1982 EPRI RPS104-2 Burns & Roe, Inc.

Field Experience with Multifrequency-Multiparameter Eddy Current Technology, Final Report, March, 1982 EPRI RPS115-1 Batelle Columbus Labs.

Flow-induced Vibration Analysis of Oconee-2B OTSG Tubes, Final Report, June 1981 EPRI RPS-176-1 Babcock & Wilcox

Guide to the Design of Secondary Systems and their components to Minimize Oxygen-induced Corrosion Final Report, March, 1982 EPRI RPS189-1 Bechtel Group

Radiographic System for Evaluation of Steam Generator Support Plate Integrity Final Report, Sept., 1981 EPRI RPS105-1 Combustion Engineering

Secondary Water Chemistry Control at Genkai NG. 1 Topical Report, May, 1981 EPRI RPS-170-1 NWT Corp.

Single-Tube Thermal & Hydraulic Tube Support Test, Final Report, Vols. 1 & 2 Sept., 1981 EPRI RPS118-1 Westinghouse Corp.

Static Strain Analysis TMI-2 OTSG Tubes Topical Report, Dec., 1981 EPRI RPS176-1 Babcock & Wilcox

Tests of Isothermal Soaking Procedures for Limiting Tube Denting in Nuclear Steam Generators Interim Report, April, 1981 EPRI RP623-2 Combustion Engineering Corp.

The vast majority of these documents do not specifically address Byron Station's steam generators or provide recommendations relative thereto. Specific recommendations or guidance aimed at Byron Station cannot be extracted from such a large compilation of material in the time frame available for response to this interrogatory. Therefore, Edison objects to the interrogatory on the grounds it is unduly burdensome.

Interrogatory No. 4

(g) provide detailed information concerning CECO's evaluation of the potential cracking problem of steam generators as described in NRC Information Notice 82-37, dated September 16, 1982, as it may apply to the Byron steam generators, and if this problem is applicable to Byron, describe in detail the corrective actions, if any, to be taken by CECO, or if no corrective actions are planned, describe in detail the reasons for CECO's position on this problem;

RESPONSE:

(g) The matters discussed in NRC Information Notice 82-37 do not pertain to issues raised in League contention 22. Therefore, Edison objects to the interrogatory on the grounds it seeks information which is neither relevant nor likely to lead to the discovery of relevant information.

Interrogatory No. 4

(h) Identify and produce all documents not already requested above relating to or relied upon in your answer to Interrogatory No. 4.

RESPONSE:

(h) Except for the following, all documents are identified in the specific responses to interrogatory 4.

1. Byron/Braidwood FSAR

2. Westinghouse instruction manuals for the Byron steam generators (Technical Manual

1440-C312, January 1980, proprietary, and Technical Manual 1440-C282, July 1976, proprietary).

Interrogatory No. 5

Concerning Contention 28:

(a) state whether any Byron-specific PRA or similar study, including but not limited to failure modes and effects analyses, systems interaction analyses, and dependency analyses, and either utilizing or not utilizing a list of important to safety equipment, has been performed to identify potential adverse systems interactions at Byron, and (i) if yes, provide a copy of the study and its results, (ii) if no, describe in detail the reasons why no such study has been done, and (iii) if no dependency analysis has been done, state with specificity what assurance there is, if any, that common cause failure will not impact upon more than one redundant safety system or function;

RESPONSE:

(a) The response relative to a Byron-specific PRA is included in Edison's response to Interrogatory No. 2. Other analyses, such as failure modes and effects analyses, have been performed. These analyses and assessments of dependencies are reported in the Byron FSAR. The following tables provide a summary of those analyses and the appropriate FSAR references.

Table 1

System	FSAR Failure Analyses Performed	Identified Dependencies
1. RCFC's	Single active failure analysis (Table 6.2-57)	One of two RCFC trains required for long-term cooling of containment following DBA. RCFC's depend on ESW, ESFAS, and vital buses (AC) (Section 6.2).

2. Containment Spray	Single active failure analysis (Table 6.5-1)	One of two spray trains required for short-term pressure reduction and fission product removal in containment following DBA. Sprays depend on RWST, vital buses (AC), ESFAS, manual action on recirculation mode, and ESW or auxiliary building ventilation to pump cubicle (Section 6.5 and Sections 9.2 and 9.4).
3. ECCS	Single active failure analysis (Table 6.3-5) Single passive failure analysis (Table 6.3-6) Failure modes and effects analysis (Table 6.3-10)	One of two trains required for core cooling during accident. ECCS depends on RWST, vital AC buses, ESFAS, manual action on recirculation switchover, CCW, ESW for SI and charging pump operation, and ESW or auxiliary building ventillation to pump cubicles for recirculation phase operation (Section 6.3 and Sections 9.2 and 9.4). (Also Nitrogen for accumulators).
4. Reactor Protection	Failure modes and effects analysis (by reference) (Section 7.2) Reactor trip correlation showing diversity (Table 7.2-4) Single failure analysis (Section 7.2.2.)	Logic (Table 7.2-1) actuation required for reactor trip. Reactor protection depends on variety of redundant and diverse instruments, 120V AC power from Class IE AC source or DC bus/inverter, and HVAC (Section 9.4).
5. ESFAS	Failure modes and effects analysis (by reference)(Section 7.3.2) Single failure analysis (Section 7.3.2)	Logic (Table 7.3.1) actuation required for ESF actuation. ESFAS depends on support systems noted in Section 7.3.1.1.5.
6. Class IE Electrical System	Division/bus independence and separation (Section 8.3)	Class IE system depends on off-site power system or DG. In turn, this requires DG cooling (ESW). DG fuel oil and other auxiliaries (Section 9.5), HVAC (Section 9.4) and ESFAS for DG actuation.

7.	ESW	Single active or passive failure analysis (Table 9.2.2)	ESW loads shown (in Table 9.2.1). ESW depends on AC power (IE) and UHS makeup systems discussed in Section 9.2.5 and HVAC (Section 9.4).
8.	CCW	Single active or passive failure analysis (Table 9.2.5)	CCW loads discussed in Section 9.2.2. CCW depends on IE AC power and ESW (Section 9.2.5) and HVAC (Section 9.4)
		Leakage analysis (Section 9.2.2)	
9.	ESF HVAC	Failure analysis (Table 9.4-10)	System supports areas and equipment noted in Section 9.4.5. System depends on Class IE AC power, ESW (Section 9.4.5).
10.	Auxiliary Feedwater	Reliability analysis performed.	Auxiliary feedwater cools the primary system. It depends on AC power or diesel-driven pump train, ESW or condensate storage tank, atmospheric relief or turbine bypass (FSAR Question 010.52).

Table 2

	Phenomena	FSAR Reference
1.	Fire	Fire Protection Report
2.	Flooding	Section 3.4
3.	Pipe Whip	Sections 3.6, 3.8, 3.9, B3.6
4.	Water Impingement	Section 3.6
5.	Localized Steam Environment	Section 3.6, 3.11, C3.6, A3.6
6.	Electrical Fault	Sections 8.2 and 8.3
7.	Missiles	Sections 3.5 and 3.9

Interrogatory No. 5

(b) if no such study as described in part (a) above has been done, state (i) whether a Byron-specific PRA or similar study as detailed in part (a) of this Interrogatory would be useful in the safety evaluation and operation of the Byron Plant, (ii) whether such a study would be necessary in the safety evaluation and operation of the Byron Plant, and (iii) if your answer to (i) or (ii) above is no, specify the reasons on which that position is based;

RESPONSE:

Not applicable. See response to 5(a) above.

Interrogatory No. 5

(c) state whether CECO has identified or knows of any attempts to identify potential adverse systems interaction with respect to the Byron Plant, and if yes, describe with particularity the identification process and its results;

RESPONSE:

(c) The analyses discussed in response to 5(a) above are presented, along with results, in the FSAR.

Interrogatory No. 5

(d) state whether CECO has taken any steps or knows of any steps which have been taken by others to respond to the concerns addressed by Dr. S. Hanauer to E. G. Case (NRC), August 18, 1977, quoted in paragraph 3.1.3 of the Affidavit of Richard B. Hubbard and Gregory C. Minor, November 12, 1980, and if yes, describe those actions in detail;

RESPONSE:

(d) The systems interaction concerns addressed in the August 18, 1977 memorandum were addressed in a September 23, 1977 memorandum from E. G. Case to S. H. Hanauer. The NRC made no recommendations for hardware changes but did issue an I&E Circular which called for review of administrative controls over testing.

Interrogatory No. 5

(e) state whether an accident resulting from a combination of human error and equipment failure could occur at Byron, and specify the reasons for your answer;

RESPONSE:

(e) Accidents involving combinations of human error and equipment failure, as with any postulated event combinations, cannot be assigned zero probability. Clearly, therefore, such events "could" occur.

Interrogatory No. 5

(f) state whether any study of the kind identified in (a) above has been performed for any other CECO nuclear plant, and, if so, produce a copy of each such study and state with particularity why such a study has been performed at other CECO plant(s) but not at Byron;

RESPONSE:

(f) The types of analyses noted in the response to 5(a) above have been performed for many other Edison plants and are documented in appropriate FSARs and/or supplementary reports to the NRC. These studies, as well as those for Byron, are available for examination at Edison or in the public document room. Only one plant-specific PRA has been performed by Edison. That study, the Zion Probabilistic Safety Study, is available as noted in response to Interrogatory 2(c). The Zion PRA was required to satisfy concerns raised by the NRC and others regarding the population density surrounding the site. Edison's reason for not performing a Byron Site-Specific PRA is discussed in response to Interrogatory 2(h).

Interrogatory No. 5

(g) identify and produce all documents relating to or relied upon in your answer to Interrogatory No. 5.

RESPONSE:

(g) 1. Byron FSAR

2. Zion PRA

3. Affidavit of Hubbard and Minor, dated November 12, 1980.

4. August 18, 1977 Memorandum for E. G. Case from S. H. Hanauer.

5. September 23, 1977 memorandum for S. H. Hanauer from E. G. Case.

6. I&E Circular 77-13, "Reactor Safety Signals Negated During Test".

7. September 28, 1977 memorandum for E. G. Case from S. H. Hanauer.

8. October 5, 1977 letter from A. Schwencer to R. L. Bolger.

9. FSAR Chapter 7.

10. FSAR Question 031.43.

11. WCAPs 7913; 7488L (Prop) and 7672 (Nonprop); 8584 (Prop) and 8760 (Nonprop); 7706L (Prop) and 7706 (Nonprop).
12. October 3, 1977 letter from J. G. Keppler to Byron Lee (I&E Inspection Report numbers 50-295/77-16 and 50-304/77-20).
13. October 4, 1977 letter from J. G. Keppler to Theron Boyce.
14. October 4, 1977 letter from J.G. Keppler to J. J. O'Connor.
15. November 9, 1977 letter from J. J. O'Connor to J. G. Keppler.
16. September 30, 1977 letter from Dr. Ernst Volgenau to T. G. Ayers.
17. October 31, 1977 letter from Byron Lee to Dr. Ernst Volgenau.
18. December 15, 1977 letter from H. D. Thornburg to T. G. Ayers.

Interrogatory No. 6

Concerning Contention 32:

(a) state with specificity what CECO believes to be adequate environmental qualification methodology for use at Byron;

RESPONSE:

(a) An adequate environmental qualification methodology consists of the following:

(1) identification of the environmental conditions in which equipment required to mitigate postulated accidents and safely shut down the plant remains functional; and

(2) demonstrating that such equipment will remain functional when subjected to these environmental conditions by type tests and/or analysis.

Interrogatory No. 6

(b) state with specificity what CECO has done, is doing, or proposes to do at Byron to satisfy the environmental qualification methodology outlined in subpart (a) above.

RESPONSE:

(b) Edison is doing the following:

(1) Edison has instituted an environmental qualification program to ensure that Class 1E electrical equipment is environmentally qualified in accordance with the methodology described in (a) above. A complete description of this program is set forth in "Equipment Environmental Qualification Report, Byron/ Braidwood Stations" (hereinafter "Edison EQ Report"), which has previously been provided to the League in connection with earlier document production requests.

(2) Edison is also in the process of developing and implementing a qualification program to qualify active safety related mechanical equipment in harsh environments. The documentation of the program is not yet complete.

Interrogatory No. 6

(c) state whether you agree that such methodology should apply to Byron's important to safety equipment and to components thereof as well as to safety-related equipment, and explain your answer in detail.

RESPONSE:

(c) Edison has committed to qualify safety related equipment to be installed and used at Byron per the methodologies identified in response to (a) above. "Safety-related" equipment is defined as equipment required to mitigate postulated accidents and safely shut down the plant. As such, for environmental qualification purposes, equipment which is important to safety has been designated as safety-related.

Interrogatory No. 6

(d) state with specificity whether all Byron "important to safety" equipment has been qualified per the requirements of NUREG-0855 and, if not, state with specificity which equipment has and has not been so qualified;

RESPONSE:

(d) NUREG-0588 places no qualification requirements on "important to safety" equipment. The Edison environmental qualification program addresses "safety-related" equipment. Therefore, no list of equipment as requested in this interrogatory exists.

Interrogatory No. 6

(e) state with specificity whether all Byron safety-related equipment has been qualified per the requirements of NUREG-0588 and, if not, state with specificity which equipment has and has not been so qualified.

RESPONSE:

(e) No; the equipment which has yet to be qualified is described at sections 4.2.1.1 and 4.2.2.1 of the Edison EQ Report.

Interrogatory No. 6

(f) state whether the NRC has completed its review of CECO's equipment qualification program at Byron and, if not, provide the schedule for its completion.

RESPONSE:

(f) The NRC has not documented its completed review of the equipment qualification program at Byron. Any schedule for completion of such review would have to be obtained from the NRC Staff.

Interrogatory No. 6

(g) state with specificity the regulatory criteria used to judge the adequacy of CECO's equipment qualification program at Byron.

RESPONSE:

(g) NUREG-0876, Safety Evaluation Report related to the operation of Byron Station, Units 1 and 2, refers to NUREG-0588 and the Commission Memorandum and Order, CLI-80-2L (May 23, 1980). Any additional information would have to be obtained from the NRC Staff.

Interrogatory No. 6

(h) identify and produce all documents relied upon in or relating to your answers to Interrogatory No. 6.

RESPONSE:

(h) (1) NUREG-0876

(2) NUREG-0588

(3) "Equipment Environmental Qualification Report, Byron/Braidwood Stations."

(4) IEEE 323-1974

Interrogatory No. 7

Concerning Contention 39 and with regard to the Byron FES, pp. 5-57 to 5-59:

(a) state with particularity the basis for the estimated groundwater travel time from the Byron Plant to the nearest spring and then to the Rock River as 24 years and describe with particularity any field tests which have been performed to verify this conclusion;

RESPONSE:*/

(a) The field tests which have been performed to verify groundwater travel time include water pressure testing and well pumping tests of the bedrock units. During the initial subsurface exploration, borings penetrated into the Ordovician age St. Peter sandstone. Within each formation, between the Dunleith and the Harmony Hill member of the Glenwood formation, water pressure tests were conducted. The results of the interpretation

---

*/ The conclusions reported in the Byron FES are based on NRC Staff analyses. However, Edison has conducted certain studies and analyses which may relate to the matters sought in interrogatory 7. The answers to interrogatory 7(a) through (c) are derived from these analyses.

and evaluation of field data are reported in FSAR Section 2.5.1.2.3.21 through 2.5.1.2.3.23. Specific relationships for hydraulic conductivity and corresponding porosity were determined from pumping tests and are reported in Section 2.4.13.2.3 of the FSAR.

Interrogatory No. 7

> (b) state with particularity the basis for the conclusion that the travel time for most of the accident-affected groundwater would be greater than 24 years and describe with particularity any field tests which have been performed to verify this conclusion;

RESPONSE:

(b) Edison believes that the following considerations were not taken into account in estimating travel time for radionuclides in the groundwater in the event of postulated accidents: (1) the effect of foundation grouting; (2) the effect of dispersion within the groundwater regime; and (3) the effect of residual heat dissipation in the event of a postulated accident.

Interrogatory No. 7

> (c) state with particularity the basis for the conclusion that in the event of release of radionuclides into the water pathways, "measurable retardation" by the dolomite aquifier, especially for cesium, would occur during the groundwater travel process, and indicate what specific effects that retardation would have on CECO's exposure dose calculations;

RESPONSE:

(c) No credit was taken for nor consideration given to retardation of radionuclides during the groundwater travel time analysis. Such retardation would be expected to reduce the total exposure dose from release of radionuclides into the water pathways.

Interrogatory No. 7

(d) state with particularity the number and location of municipal wells actually unaffected by recharge from a contaminated Rock River because they screen into aquifers not closely connected to the water table aquifer, and the specific effects of that figure on CECO's exposure dose calculations;

RESPONSE:

(d) The conclusion reported in the FES is based on NRC Staff evaluations. The specific information sought in this interrogatory would have to be obtained from the NRC Staff.

Interrogatory No. 7

(e) state with particularity (i) the reasons that the current amount of grouting beneath the plant site would be ineffective to prevent contamination of groundwater flow, (ii) the reasons additional grouting and well point dewatering would allow isolation of "radioactive contamination near the source" when the present grouting does not, and (iii) the reasons why additional steps are not now being taken to interdict the flow of contaminated groundwater if the current level of grouting will be ineffective for that purpose;

RESPONSE:

(i) The conclusion reported in the FES is based on NRC Staff evaluations. The specific information sought in this interrogatory would have to be obtained from the NRC Staff.

(ii) The conclusion reported in the FES is based on NRC Staff evaluations. The specific information sought in this interrogatory would have to be obtained from the NRC Staff.

(iii) Edison is not taking any additional steps to interdict the flow of contaminated groundwater because it is very improbable that such an event would ever occur and because taking any further steps would unnecessarily interrupt groundwater flow in the area around the Byron Station. In addition, no regulations exist which require Edison to take any further action.

Interrogatory No. 7

(f) in the event of a radioactive release to the underground aquifers, indicate with specificity what measures have been taken or are planned to be taken in the future to prevent the further migration of contaminated material away from the Byron site;

RESPONSE:

(f) At this time Edison has taken no measures and has no specific plans for measures to prevent migration of radioactive material from underground aquifers. Edison's investigation into this matter, however, continues.

Interrogatory No. 7

(g) for each of the accident scenarios postulated as applicable to Zion which would also be applicable to Byron and which were assumed to lead to the release of radioactive materials to the groundwater or to the area beneath the Byron plant, or in the vicinity of the Byron plant, state with specificity by isotopes what varieties of radioactive material would be released, the range of core temperatures which have been assumed for any accident scenarios involving a core melt, and the assumed depth to which the core could sink, and the basis for these assumptions at Byron;

RESPONSE:

(g) Absent a PRA specific to Byron, Edison cannot specifically identify those scenarios which would conceivably lead to release of radioactive materials to the groundwater or identify specific isotopes, the range of core temperatures, or the depth to which the core could sink. However, Edison's investigation of this matter continues.

Interrogatory No. 7

(h) state with particularity any data known to CECO on potentiometric surfaces for the Byron site (and the region surrounding the Byron site) water table aquifer and confined aquifer;

RESPONSE:

(h) A discussion of the water table aquifer at the Byron site and the surrounding region is provided in FSAR Subsection 2.4.13.1.2.2. A listing of the data is provided in FSAR Table 2.4-25. The piezometric (potentiometric) surface of the confined Cambrian-Ordovician Aquifer is discussed in FSAR Subsection 2.4.13.2.1. FSAR Table 2.4-22 shows the yearly changes in the piezometric levels at public groundwater wells completed in the Cambrian-Ordovician aquifer. A contour map of the piezometric surface of the Cambrian-Ordovician aquifer is shown in FSAR Figure 2.4-26.

Interrogatory No. 7

(i) state with particularity all data known to CECO on the permeability and/or transmissivity of the water table aquifer and confined aquifer in the Byron area, including all measurements and how those measurements were made;

RESPONSE:

(i) A discussion of the transmissibity of the confined aquifer and the methodology used to collect the data for this analysis is provided in the Byron FSAR, Subsection 2.4.13.1.3. A discussion of the hydrogeologic properties of the water table aquifer is provided in FSAR Subsections 2.4.13.2.3. and 2.4.13.3. It should be noted that the water table aquifer, the Galena-Platteville dolomites, has very little primary permeability. The groundwater moves primarily along solution enlarged joints that provide secondary permeability. Therefore, the permeability of the water table aquifer is quite variable depending on the presence or absence of secondary permeability. The secondary permeability of the dolomites under the plant has been reduced by the extensive grouting program for the plant foundations.

Interrogatory No. 7

(j) state with particularity all data known to CECO on the measurements of the porosity of the rocks underlying the Byron site, the specific yield of the Byron site aquifers, and how those measurements were made;

RESPONSE:

(j) The total porosity of the rock units underlying the Byron site ranges from 15 to 20% as measured from borehole geophysical logs. These geophysical logs are shown on FSAR Figures 2.5-230 through 2.5-245. The effective porosity of the Galena-Platteville dolomites is estimated to range from 5 to 10% based on regional data for these rock units. The effective porosity is the amount of interconnected pore space through which fluids can pass, expressed as a percent of bulk volume. Effective porosity is less than total porosity since a part of the total porosity will be occupied by static fluid being held to mineral surfaces by surface tension.

The capacity of the Byron confined aquifers are discussed in FSAR Subsections 2.4.13.1.2.3 and 2.4.13.1.3. The specific capacity of the water table aquifer, as determined from pumping tests in two domestic wells performed in the Galena-Platteville dolomites, was .43 gpm/ft and 33.8/gpm/ft. The great variation in the specific capacity of these two wells is due to the amount of secondary permeability encountered by each well. The water table aquifer, the Galena-Platteville dolomites, has very little primary permeability. The groundwater moves primarily along solution enlarged joints that provide secondary permeability. Therefore, the greater number of solution enlarged joints encountered by a well, the higher the specific capacity.

Interrogatory No. 7

(k) state with specificity all data known to CECO on the dispersivity of the Byron water table aquifer and confined aquifer and the methods used to acquire that data; and

RESPONSE:

(k) A discussion of accident effects and the methodology utilized in the analysis thereof are presented in FSAR Subsections 2.4.13.3 and 2.4.12. The water table aquifer, the Galena-Platteville dolomites, is hydraulically separated from the lower confined Cambrian-Ordovician aquifer by the Harmony Hill Shale Member of the Glenwood Formation (see FSAR Subsection 2.4.13.2.3). Therefore, the analysis of accident effects on groundwater was limited to the Galena-Platteville aquifer. To be conservative in the analysis of accident effects, the effect of dispersion was ignored.

Interrogatory No. 7

(l) identify and produce all documents relied upon in or relating to your answers to Interrogatory No. 7.

RESPONSE:

(l) All documents used are indentified in the reponse to specific interrogatories.

Interrogatory No. 8

Concerning Contention 42:

(a) state whether worker radiation exposure levels at Byron were calculated with a current doseconversion factor based on models contained in ICRP-2 (NUREG/CR-0150);

RESPONSE:

(a) No.

Interrogatory No. 8

(b) if the answer to (a) above is no, indicate what method was used;

RESPONSE:

(b) The inplant man-rem doses used are whole body exposure levels due to gamma radiation emanating from confined sources. ICRP-2 and NUREG/CR-0150 relate to doses due to inhaled or ingested radionuclides, not whole body doses and thus are not representative of routine occupational exposures.

Details on the methodology used to calculate gamma radiation fields may be found in the FSAR, Section 12.3.2.1.9.

Interrogatory No. 8

(c) do you agree that low doses of radiation produce more cancers per rem than high doses of radiation, and if your answer is no, explain in detail the reasons for this position;

RESPONSE:

(c) No. The best reliable evidence available to the scientific community strongly suggests that for exposure to low-LET radiation, the linear model probably tends to overestimate the risk of most radiation-induced cancers in man but that the model can be used to define the upper limits of risk. Edison is aware of claims of higher risks from low-dose levels described by Bross, Mancuso, Stewart, Kneale, Najarian, Morgan, Bertell and others. However, to date examination of the work performed by these individuals and their reports does not support these claims. While some of these studies and data may be worthy of further investigation, they are, at this point, not considered convincing enough to argue effectively against the conservatism associated with the linear hypothesis for the human being. (See also response to 16(a) infra.)

The following is a discussion of some of the more significant studies which purport to challenge the linear hypothesis.

Several recent reports (Bross and Natarajan 1972; 1977; 1979; Bross and Driscoll, 1981; Mancuso, Stewart and Kneale, 1977; Najarian and Colton, 1978; 1975; Bertell, 1977) have been interpreted by their authors and by some people to indicate that the currently derived and applied radiation risk estimates, which are based primarily on the UNSCEAR (1977), NAS-BEIR Committee (1980) and ICRP (1977) Reports, underestimate the risk of radiation at all dose levels. These authors especially claim that application of the "linear, no-threshold hypothesis" (that the radiation risk per unit dose as derived by linear interpolation from available epidemiological data at high-dose levels of radiation dose holds all the way down to zero excess incidence and zero dose above natural background levels is not sufficiently conservative in estimating risk at low doses, but rather underestimates it. All reports of expert advisory committees including ICRP (1977), UNSCEAR (1977), NCRP (1980), and NAS-BEIR (1980) disagree with such claims and present the current and detailed scientific evidence which does not support such claims.

Mancuso, Stewart and Kneale (1977) have reported preliminary findings on the work and mortality experience of 24,939 male workers with 3,520 certified deaths (death certificates) and of a number (not specified) of female workers with 412 certified deaths at the Hanford nuclear facility, Richland, Washington, between 1943 and 1971. In their preliminary report of 1977, which was largely limited to analysis of cancer mortality data on the 3,520 male deaths for which death certificates were available, they claimed that analysis demonstrated a radiation-induced excess of cancers, greater than the linear dose-response hypothesis would indicate. However, their analysis has been widely criticized by leading epidemiologists and statisticians primarily because of serious deficiencies in methodology, formulations and conclusions (see NCRP 1980; Hutchison et al, 1979; NAS-BEIR, 1980; Reissland, 1978; GAO, 1981; Anderson, 1978; Mole, 1978; Gilbert and Marks, 1979). Additional analyses of the data have been performed which show little or no radiation induction effect (see references cited in previous sentence).

Some of the more serious methodological flaws in the Mancuso, Stewart and Kneale (1977) pertain to inadequacies of radiation dosimetry, confounding factors which could have caused cancer in workers in the absence of radiation, selection bias, and inconsistencies with the spontaneous incidence of cancer in the exposed populations.*/ Their report does not state the actual individual radiation doses received by Hanford workers who died of cancer; they only provide mean cumulative radiation doses. Their analysis did not take into account the calendar year in which the cancer began in the individual and in the study population and made no correction for the fact that the incidence of the cancers observed in the Hanford workers also increased during the period of the study in the United States population at large. Thus, their conclusion, showing an increase in cancer with increasing dose accumulation over increasing time, fails to take into account that even in the absence of the increasing dose of radiation, there is a similar increase in cancer incidence in the United States as a whole when the incidence of cancer in the general population is plotted against increasing time.

Other analyses of the same data published by Gilbert and Marks (1979) and by Hutchison et al., (1979)

---

*/ (See NCRP, 1980; Hutchison et al., 1979; NAS-BEIR, 1980; Reissland, 1978; GAO, 1981; Anderson, 1978; Mole, 1978; Gilbert and Marks, 1979; Marks and Gilbert, 1978; Darby and Reissland, 1981).

point to the possibility of an association with the work experience of the study population for two types of cancer: cancer of the pancreas and multiple myeloma. In these studies, there is no radiation relationship for lymphatic or cancers of the blood-forming tissues other than multiple myeloma, i.e. no excess of leukemias (which experience such as in the Japanese atomic bomb survivors, suggests should have been most observable where radiation is a factor).

Since the recorded radiation doses in the Hanford workers were very small, perhaps on the order of a few rads, then the very low cancer-doubling dose estimates reported by Mancuso, Stewart and Kneale (1977) in their report would be spurious. Their doubling-dose estimates have been strongly disputed by numerous scientists since their values would be inconsistent with known and established radiobiological evidence. If the estimated small dose in the worker population actually caused a doubling of the spontaneous rate of cancers, then natural background radiation in the United States would produce more than the actual numbers of cancer cases observed in the entire U.S. population. This just cannot occur. Therefore, it appears that if the Mancuso, Stewart and Kneale (1977) cancer-doubling doses are correct, something other than radiation was the cause of the observed cancers in the Hanford workers. In the light of these criticisms, Mancuso, Stewart and Kneale (1977) now appear to have modified their estimates of cancer-doubling doses and are presently quoting doubling doses of 2-150 rem in the worker population; however, the lower range still is inconsistent with existing knowledge and experience in cancer epidemiology and statistics (Stewart et al., 1980; Kneale et al., 1981).

Another controversial study is that of Najarian and Colton (1978) on the Portsmouth Naval Shipyard workers in New Hampshire. Since the Najarian and Colton (1978) study, much criticism of their analyses has been recorded; these flaws include bias in worker selection, worker history, radiation dosimetry, and confounding factors such as exposure to other carcinogens in the workplace (Hamilton, 1979). In their initial report, Najarian and Colton (1978) estimated that, since 1959, the Portsmouth Naval Shipyard in New England has serviced nuclear-powered ships; and that, over the past 20 years, about 20,000 people have been employed there, of whom about 20% were exposed to radiation. In their search of worker death certificates from 1959-1977, they estimated that 1,450 former Portsmouth Naval Shipyard employees had died before age 80. The authors then contacted near relatives of the deceased by telephone to determine whether these ex-employees were radiation-exposed workers. They were successful in obtaining telephone information on 525 cases and they established that 146 were probably exposed to radiation during their working lives. They then concluded that, compared with mortality in U.S. while males for 1973, the observed numbers of cancers and leukemias in this selected worker population were considerably greater than those expected. The actual numbers were quite small and the conclusions based upon those numbers turned out to be erroneous.

However, even Majarian and Colton (1978) listed the important inadequacies in their survey. Their study was an analysis of cancer deaths only and provided no information on the total worker population at risk. There could be a significant bias in the information supplied by relatives, since this was recall information. No information was provided on how long workers worked at the shipyard, how long nuclear workers were exposed to radiation, and the amounts of radiation the workers received. Dosimetry data were not provided. No consideration was given to any confounding factors such as the carcinogenic effects of other toxic agents such as asbestos, smoking or industrial solvents, which could have acted either alone or in an additive fashion or through a multiplicative mechanism with radiation to cause the apparent excess deaths from cancer and leukemia.

There are further serious statistical and/or methodological inadequacies in the Najarian-Colton (1978) survey (Hamilton, 1979). For example, in order to exclude the effects of carcinogens other than radiation, the authors should have been shown that the cancer frequencies in the study population increased with increasing radiation exposure; however, knowledge of the lifetime-accumulated doses of the former employees no longer employed at the shipyard was not available. More importantly, if the radiation work at the Portsmouth Naval Shipyard began only in 1959, it is unlikely that changes in overall cancer frequency induced by radiation would appear before a minimum latent period of at least 10 years after beginning of exposure, or, in the case of leukemia, after 5 years. These are roughly the minimum latent periods for cancer and leukemia induction in other exposed populations studied. With this in mind, it is vital to review the Najarian-Colton (1978) data analysis which was divided into two periods: cancer deaths occuring during the period from 1959-1969, when radiation effects would not be expected to appear and cancer deaths occuring from 1970-1977, when radiation effects might be expected to begin to appear.

In those workers who died between 1959-1969, about 25% had cancer listed on their death certificate as the cause of death. However, only 33 radiation workers died during this period and about 40% of their deaths were recorded as due to cancer. In those workers who died between 1970-1977, about 25% had cancer as the cause of death. Hence, there was no significant difference between the percentage of cancer deaths between the two periods for all workers. Moreover, of the 113 radiation workers who died during the 1970-1977 period, about 40% were due to cancer -- no more than during the earlier previous 10-year period. Thus, there were no differences in the incidence of cancer deaths in all workers and no differences in the incidence in the radiation workers during the two periods. The data, therefore, are not concordant with well-established medical and epidemiological data on the effect of the latency period on expression of risk of radiation-induced cancer. The absence of any apparent latent period effect casts considerable doubt on any conclusions by Najarian and Colton (1978) (and others who have chosen to cite these conclusions as evidence of very low-level effects) about the contribution of radiation to the unexplained high numbers of cancer deaths among the radiation workers (Reissland and Dolphin, 1978). And finally, when dosimetery data were made available to Najarian and Colton, a number of serious inconsistencies in their analysis became apparent; for example, one-third of the leukemia cases reported in their original paper had no history of radiation and another one-third had negligible levels of exposure. With the new dosimetry data, statistical analyses showed no significant differences in the cancer incidence in the different exposure levels (NCRP, 1980). In addition, the list of chemical and physical agents probably present at the Portsmouth Naval Shipyard during the past 25 years includes over 40 potentially harmful chemicals. (Hamilton, 1979). The common occupational carcinogens affecting health and work in the

United States are quite well known. The presence of so many chemical carcinogens in the workplace underscores the difficulty in assessing the effects of low levels of radiation in this and other nuclear worker populations.

The final report of the U.S. Department of Health and Human Services, Public Health Service Centers for Disease Control, National Institute for Occupational Safety and Health (NIOSH)'s Epidemiologic Study of Civilian Employees at the Portsmouth Naval Shipyard, based on a total cohort of 24,545 civilian white males employed at PNS between 1952 and 1977, is now available (Rinsky et al., 1982). The report found no excess of deaths due to malignant neoplasms and due specifically to neoplasms of the blood and blood-forming tissues (leukemias) in civilian workers at the Portsmouth Naval Shipyard. This NIOSH study found no relationship between exposure to radiation and mortality from any cause among the worker population when compared to the United States while male population. Furthermore, no excess in leukemia mortality was observed in the radiation exposed population when compared to the non-radiation exposed employees of Portsmouth Naval Shipyard. This report has been reviewed by a National Academy of Science - National Research Council Scientific Advisory Committee (NAS-NRC, 1982); the committee did not disagree with the NIOSH study findings.

Dr. Bross (Bross and Driscoll, 1981; Bross and Natarajan, 1972; 1977; Bross et al., 1979, see also Bertell,

1977) has claimed that the risk for cancer-induction following diagnostic X-ray exposure in pregnancy or following adult diagnostic X-rays, which are low-level radiation exposures, is greater than that observed at high doses and high dose rates. Furthermore, he believes he has identified susceptible subgroups in the general population which are especially sensitive to radiation damage. His belief derives from his analysis of the Tri-State Leukemia Survey (Graham et al., 1966; Gibson et al., 1972) wherein he studied an association between what he has termed certain "indicators of susceptibility" (e.g., viral infections, bacterial infections and allergy) shown by the leukemic child from birth until diagnosis of leukemia. Bross concluded "the apparently harmful effects of antenatal irradiation are greatly increased in certain susceptible subgroups of children possessing the indicators associated with a slightly higher intrinsic risk of leukemia". However, re-analysis of Bross' observations (Smith et al., 1973) shows that children with leukemia are simply more prone to viral and bacterial infections and allergies before the clinical onset of the leukemic disease, i.e., these indicators characterize the disease itself and do not relate to the child's inherent susceptibility or sensitivity to induction leukemia. The incidence of these "indicator" diseases as part of the pre-leukemia phase of leukemia in children is well known in pediatric medicine and in clinical hematology. Analysis of Bross' data shows that the incidence of these "indicator" diseases before the clinical onset of leukemia is the is the same in children who had received no irradiation in utero as in those who had. The hypothesis of Bross, that there is a susceptible portion of the population at higher risk of leukemia, has also been challenged on the grounds that Bross' methods yield no way to identify susceptible individuals ahead of time and, therefore, permit no way to test his thesis (Smith et al., 1973).

More recently, Bross has claimed that the relatively small radiation exposures (in the millirad range) from diagnostic X-rays in adults significantly increases the risk of leukemia (Bross et al., 1979). In coming to this conclusion, it appears that Bross erroneously assumes that, in the absence of diagnostic X-rays, the incidence of heart disease and leukemia in the general population is zero, and, of course, this is not the case. Were this not his erroneous assumption, the fact that his "dose-response" curves of adults exposed to diagnostic X-rays are flat below 10 rad exposure would suggest a threshold existed in the dose-response relationship. Indeed, a more conventional relative risk analysis recently done (Boice and Land, 1979) found little or no increase in risk of leukemia from a small number of diagnostic X-rays in the Bross study populations. Bross also erroneously assumes that relative risks are fixed and that the "percentage of the population affected" varies with dose, i.e., he assumes that the basic response variable is the proportion of the irradiated population affected by radiation rather than the dose. Conventional relative-risk analyses assume that everyone is affected and that the relative risks vary with dose. The reason for Bross' unconventional methodological approach is unclear. The position taken here by Bross appears to be at odds with his earlier paper (Bross and Natarajan, 1972) in which he postulated the existence of a sensitive subgroup of fixed size whose relative risk of leukemia increased rapidly with increasing X-ray dose. Finally, in Bross' analysis, it should be noted that the leukemia risk (or "percent affected") increases dramatically only in males, and then only after large numbers of diagnostic X-rays, but that females appear to be unaffected. No radiation dosimetry was performed in the Tri-State Survey. However, the cause-effect relationship is obscured by the fact that very large numbers of diagnostic X-rays --approximately 40 or more within 10 years -- implies that a disease state is present and is perhaps deriving from heart disease or a preleukemic sensitivity to infections.

Further interpretations of the Tri-State leukemia study data are introduced by Bross (Bross and Natarajan, 1972; 1977; Bross et al., 1979), interpretations which have subsequently been severely criticized in the open scientific literature (Smith et al., 1973; Land, 1977; 1979; Oppenheim, 1977; Boice and Land, 1979; Rothman, 1977; MacMahon, 1972; Hamilton, 1979) as have the conclusions Bross has drawn.

Dr. Bross has recently claimed (Bross and Driscoll, 1981) that the Portsmouth Naval Shipyard workers sustained very large numbers of lung cancer deaths as a result of exposure to low level radiation. In Bross's paper on this study, he takes data from the Naharian-Colton survey and draws spurious conclusions using unconventional statistical methods. The Bross and Driscoll (1981) analysis once again makes unsubstantiated claims on the subject of susceptible persons or subpopulations developing cancer, far beyond the acceptable dose range. In their attempt to re-analyse the data from the Portsmouth Naval Shipyard Study, Bross and Driscoll (1981) claim that the official publication of Rinsky et al (1981) was purposely misleading, and they further claim that the intentional misleading underestimated the lung cancer risk by a factor of 20 to 200. By re-grouping selected data for lung cancer (which, incidentally, do not appear in the Rinsky et al (1981) paper) Bross concludes that, above the 1-rem-range and with greater than 15-year follow-up, there is a two-fold increase of lung cancer. This would mean an excess of 189 deaths per 106 persons exposed per year per rem compared with the ICRP (1977) and NAS-BEIR (1980) estimates of about 1 lung cancer death per 106 persons exposed per year per rem. Since no detailed denominators, nor basis for expected cases nor host factors are given or corrected for in the Bross analysis, his conclusions cannot be evaluated nor substantiated. Smoking was not examined in any detail as an important confounding factor in Bross' analysis.

It has been claimed by Dr. K.Z. Morgan (Morgan, 1975) that low-level exposure may, in fact, be more hazardous per unit of absorbed dose than that at high doses and and dose rates. However, in that assessment Morgan made no clear differentiation between effects of high- and low-LET radiation; hence, Dr. Morgan has never demonstrated that the claim holds for low-LET radiation. Furthermore, he placed emphasis on the potential effects of high-LET radiations at high doses from internally-deposited radioisotopes; this situation does not obtain for the low-level dose range of low-LET radiation exposure.

Certain human thyroid tumor data derived from the young Israeli children irradiated for ringworm of the scalp (Modan et al., 1977) appear to show that the risk coefficients at low doses may be equal to or even greater than those at high doses and dose rates. However, there are substantial uncertainties in the dosimetry. Interpretation of the low-dose thyroid cancer effect in the Modan series (Modan et al., 1977) must consider the possibility that (a) imprecise irradiation techniques or restless children could have resulted in direct thyroid exposure; (b) pituitary irradiation may have influenced thyroid cancer rise; (c) there may have been interactions between radiation and other factors such as ethnic, nutritional deficiencies or goiter to alter the risk. These results must be balanced against the possible influence of pituitary irradiation in these cases, the lack of thyroid tumors in other similar series, and the lack of such an effect in children in Utah who apparently received thyroid doses from fallout radioiodine much larger than those reported in the Modan series, but did not demonstrate an apparent increase in thyroid tumor incidence.

Interrogatory No. 8

(d) state specifically the realistic person-rem dose per year for each Byron reactor and why you consider that dose to be realistic, the number of major reactor overhauls, including but not limited to the replacement of steam generators, expected to be performed during the lifetime of each reactor, and the resulting person-rem from each of those overhauls;

RESPONSE:

(d) Byron FSAR Section 12.4.4 (Attachment B) "Estimated Annual Occupational Exposures" discusses and estimated man-rem totals for Byron Station for operations and for refueling outage work. The levels reported are consistent with Zion experiences. No extrapolation has been made to include postulated major tasks such as steam generator replacement.

Man-rem totals for replacement of steam generators at Byron Station can be estimated by utilizing data obtained from NUREG/CR-1595, which documents the Virginia Electric & Power Co. Surry Nuclear Station Steam Generator replacement estimated man-rem totals. (See specifically, Table 9). Surry Station is a 775 MWe per unit, 3-loop (3 steam generators) Westinghouse, PWR, and the man-rem totals were: unit 2 - 2140 man-rem; unit 1 - 1750 man-rem. Byron

Station is a 1120 man-rem unit, 4-loop (4 steam generators), Westinghouse, PWR, and the estimated man-rem totals associated with steam generator replacement would be 2850 man-rem for the first unit and 2280 man-rem for the second unit. The reason for the 570 man-rem reduction for the second unit is based on experience which would be gained as a result of the work performed on the first unit. These estimates assume that any Byron Station steam generator exposure rates and replacement procedures would be approximately the same as those experienced and utilized at the Surry Station.

Interrogatory No. 8

(e) state specifically the provisions made for the staffing of a Byron health physics department and for the training of that staff;

RESPONSE:

(e) The planned staffing of the Health Physics staff in conjunction with operation of both units is as follows:

Staff Title	Number of Personnel
Radiation Chemistry Supervisor (Radiation Protection Manager)	1 *
Station Health Physicist	1
Health Physicist	3
Health Physics Engineering Assistants/Engineering Technicians	4
Health Physics Foreman	6
Radiation Chemistry Technicians	28 **

* The Radiation Chemistry Supervisor supervises both Health Physics and Chemistry Programs.

** Radiation Chemistry Technicians perform both Chemistry and Health Physics Functions, thus all 28 technicians would not be assigned to only Health Physics. Edison estimates that 70% of the total 28 technicians' time would be spent on Health Physics related responsibilities.

The training for the Health Physics staff, which includes the Radiation Chemistry Supervisor, Station Health Physicist, Health Physicist, Health Physics Engineering Assistant/Engineering Technicians and Health Physics Foreman, will be as described in the Byron FSAR Section 13.2.1.6 and implemented in Byron Administrative Procedures (BAP) 560-1 560-TL. The following is a summary of the general training program:

Objectives - The Objectives of the management training program are to:

a. Ensure that each person is sufficiently trained in his/her job in order to fulfill any ANSI (or NRC, etc.), training requirements specified for the individual's position.

b. Ensure that each person is sufficiently trained in his/her job in order to be able to adequately implement his/her responsibilities as specified in the individual's job position description.

The Training Program -

When an individual enters a particular job position for the first time at Byron Station, the Rad-Chem Supervisor will evaluate the person's experience and training record. As necessary, the Rad-Chem Supervisor will outline a plan in order to reasonably assure that the individual is capable of performing his/her duties in a manner that will meet the objectives. This plan may include the following items:

a. Attendance of offsite training courses or seminars.

b. Attendance of selected training sessions offered for Rad-Chem Technicians.

c. Attendance of other Company or Station training courses.

d. Temporary assignment to locations in the Company where appropriate on-the-job experience can be obtained.

The training for the Radiation Chemistry Technicians will be as described in Byron FSAR Section 13.2.1.7 and the draft copy of Byron Procedure "Radiation Chemistry Technician Training Program." The Byron Chemistry Guidelines 1930-1 through 1930-10, 1930-T1 through 1930-T10 and Byron Radiation Guidelines 1930-1 through 1930-10, 1930-T1 through 1930-T10 will be used to implement the training program.

Interrogatory No. 8

(f) provide copies of any studies performed by, or known to, CECO concerning expected values of in-plant radiation exposure and of design and procedure changes, addition of equipment and/or tools to reduce such exposure;

RESPONSE:

(f) See: FSAR § 12.3, Response to Q331.3, SER §§ 12.1, 12.3.

Interrogatory No. 8

(g) as regards steam generators, provide detailed information on material selection, hardware configuration, maintenance tooling, and access platforms and cranes that have been specified so as to reduce or minimize the in-plant radiation exposure;

RESPONSE:

(g) 1) Permanent galleries and access ladders have been provided at both the lower and upper man-way locations. The configuration and location of the galleries and ladders is shown in S&L drawings M-913, sheets 3 and 4, elevations 390'-0 and 391'-8-1/2; M-913 sheet 13 elevation 448'-9;

2) A manway handling system is being installed to ease removal and installation of the manways at the bottom of each steam generator;

3) Remote control inservice inspection equipment will be used; and

4) Air in the area of the steam generator manways will be filtered to remove radioactive contaminants before releasing it to the containment.

Interrogatory No. 8

(h) describe with particularity all Byron plant features which have been modified or added so as to provide a reduction of in-plant radiation exposure;

RESPONSE:

(h) See: FSAR § 12.3, Response to Q331.3, SER §§ 12.1, 12.3.

Interrogatory No. 8

(i) provide copies of all CECO procedures written for the implementation of ALARA provisions at Byron; and

RESPONSE:

(i) Copies of these procedures are available for inspection at Edison.

Interrogatory No. 8

(j) identify and produce all documents relied upon in or relating to your answers to Interrogatory No. 8 not otherwise requested above.

RESPONSE:

(j) All documents used are identified in the reponse to specific interrogatories.

Interrogatory No. 9

Concerning Contention 61:

(a) State in detail how the current environmental qualification methodology which CECO is using for Byron differs from the methodology in use prior to the events at TMI-2.

RESPONSE:

(a) The environmental qualification methodology used to qualify equipment at Byron was not altered or modified as a result of the events at TMI-2. However, since NUREG-0588 was published following TMI-2, Edison's environmental qualification methodology for Byron equipment was modified in accordance with the guidelines provided in that document.

Interrogatory No. 9

(b) With regard to the discussion in the Byron FSAR concerning NUREG 0737 and Byron equipment which is similar or identical to the equipment which failed at TMI-2, state with particularity which items of equipment and components of equipment in that discussion have been classified as important to safety and which have been classified as safety-related only.

RESPONSE:

(b) The term "important to safety" is not used in the classification of byron systems or components for environmental qulification purposes. In response to the requirements of NUREG-0737, the following safety related hardware will be included in the Byron Design: 1) reactor head vent; 2) post-accident sampling system; 3) direct position indication system for pressurizer safety valves; 4) noble gas effluent monitor; 5) containment high range radiation monitor; 6) containment pressure monitor; 7) containment water level monitor; 8) containment hydrogen monitor; 9) reactor vessel level indication system; and 10) additional accident monitoring instrumentation as required by Category I of Reg. Guide 1.97 Rev. 2.

Some equipment, for example the core exist thermocouples and other accident monitoring instrumentation, the power supply to the PORV's, and the power supply to the PORV block valves, was included in the Byron design prior to the issuance of NUREG-0737, and has been upgraded to safety related quality as a result of the issuance of that document.

Interrogatory No. 9

(c) State whether a full Class 9 analysis of Byron has been conducted to establish the worst case environment for use in qualification of equipment important to safety, and (i) if your answer is yes, provide all data on the study, and (ii) if your answer is no, explain the reasons why such an analysis was not conducted.

RESPONSE:

(c) No such class 9 analysis has been performed for Byron Station. The "Class 9" events do not consitute a design basis for Byron Station. The design bases for Byron Station have been established in accordance with NRC regulations and regulatory requirments. See also Edison's response to Interrogatory 2.

Interrogatory No. 9

(d) State whether a full Class 9 analysis of Byron has been conducted to establish the worst case environment for use in qualification to safety related equipment, and (i) if your answer is yes, provide all data on the study, and (ii) if your answer is no, explain the reasons why such an analysis was not conducted.

RESPONSE:

(d) See response to 9(c) above.

Interrogatory No. 9

(e) State with particularity what safety margins are used by CECO in establishing the range of accident environments that equipment important to safety must be qualified to withstand.

RESPONSE:

(e) The Byron equipment qualification program does not use the classification "important to safety". See also response to 9(f) below.

Interrogatory No. 9

(f) State with particularity what safety margins are used by CECO in establishing the range of accident environments that safety related equipment must be qualified to withstand.

RESPONSE:

(f) Although the safety margins have not been quantified, significant safety margins result from the following: the accident environments used in the Byron equipment qualification program are calculated using conservative methodology in accordance with the guidelines in Standard Review Plan Section 3.6 and 6.2 and NUREG-0588. The accident conditions relating to pressure, temperature and humidity are calculated using initial conditions which are conservatively chosen to predict the most severe conditions. The methodology and computer programs used to perform these calculations have been established to be conservative by comparison with NRC benchmark standard programs. The radiation level calculations similarly employ conservative assumptions as to the strength, geometry, and position of the radiation sources. The use of these conservative methodologies results in margin between the actual expected conditions and the conditions specified for environmental qualification. Additional margin results from the fact that equipment or component test conditions and duration are typically more severe than specified and because the components are acceptable only if they do not fail.

Interrogatory No. 9

(g) Identify and produce all documents relied upon in or relating to your answer to Interrogatory No. 9.

RESPONSE:

(g) (1) NUREG-0588

(2) Byron FSAR

(3) NUREG-0737

(4) U.S. NRC Standard Review Plan (NUREG-75/087), September 1975.

Interrogatory No. 10

Concerning Contention 62:

(a) state whether or not you agree that multiple independent or common-cause failures of systems and equipment are possible at Byron;

(1) if your answer is no, explain the reasons for your answer in detail;

(2) if your answer is yes, state with particularity (i) which Byron-specific multiple failure sequences you believe could lead to a class 9 accident, (ii) what measures CECO is employing or contemplating employing to prevent or mitigate the occurrence and the effects of such class 9 accidents, and (iii) if no Byron-specific multiple failure sequences/class 9 scenarios have been developed, explain in detail why they have not been; and

RESPONSE:

(a) Clearly, multiple independent or common-cause failures of systems and equipment cannot be assigned a zero probability at Byron or any other facility. Therefore, such events are "possible".

(1) Not applicable.

(2)(i) Absent a detailed Byron-specific PRA it is not possible to answer with particularity which specific sequences could lead to a Class 9 accident. Typically, such PRA's examine tens of thousands of sequences. The draft evaluation identified in response to interrogatory 2(b) identifies certain sequences which are typical of those dominant sequences that might derive from a Level 3 site-specific Byron PRA.

(2)(ii) The safety measures described in the Byron FSAR are designed to prevent Class 9 events.

(2)(iii) The development of Byron-specific sequences of and by themselves is not necessary for the safe operation of the plant.

Interrogatory No. 10

(b) identify and produce all documents relied upon in or relating to your answers to Interrogatory No. 10.

RESPONSE:

(b) All documents used have been identified in the response to the specific interrogatory.

Interrogatory No. 11

Concerning Contention 63:

(a) state specifically which systems, equipment, and equipment components at Byron which were classified as non-safety related prior to the events at TMI have been, as a result of those events, reclassified important to safety, safety related, or have been assigned to an intermediate category between safety related and non-safety related, and if no such reclassification has occurred, explain in detail why not;

RESPONSE:

(a) See response to 9(b) supra.

Interrogatory No. 11

(b) state whether any Byron-specific non-design basis studies, including but not limited to a PRA, have been done or are planned in order to evaluate or reclassify any equipment classified as non-safety related prior to TMI-2, and if no such studies have been done or are planned, explain in detail why not;

RESPONSE:

(b) No Byron-specific non-design basis studies or PRAs have been performed or are planned for the purpose of evaluating or reclassifying any equipment classified as non-safety related, because there is no regulatory requirement that such studies be conducted.

Interrogatory No. 11

(c) state with specificity whether CECO has evaluated improvements in risks which might result from the addition of safety features, including but not limited to filtered/vented containment, to reduce the releases during a Class 9 accident at Byron, and (i) if your answer is yes, provide all data regarding that evaluation, and (ii) if your answer is no, explain in detail why not;

RESPONSE:

(c) Edison has not performed such an evaluation. Such evaluations are not required for individual licensing proceedings. They are the subject of generic evaluations by the NRC and the nuclear power industry which are currently in progress.

Interrogatory No. 11

(d) state with specificity whether CECO has evaluated the improvement in risks that may result from the addition of a core catcher beneath the pressure vessel to delay release of core melt material to the environment, and (i) if your answer is yes, provide all data regarding that evaluation, and (ii) if your answer is no, explain in detail why not;

RESPONSE:

(d) Edison has not performed such an evaluation for the same reasons as stated in the response to 11(c) above.

Interrogatory No. 11

(e) identify and produce all documents relied upon in or relating to your answer to Interrogatory No. 11.

RESPONSE:

(e) 1. NUREG-0737

2. Byron FSAR

Interrogatory No. 12

Concerning Contention 77

(a) State specifically each piece of "important to safety" equipment and the components of such equipment which have been environmentally qualified by subjecting them first to the aging effects of radiation, temperature, and vibration, and then subjecting them to seismic testing requirements, and state with particularity the design, procedures, content, and results of any such testing.

RESPONSE:

(a) The term "important to safety" has not been used to classify equipment installed at Byron which will be subject to Edison's environmental qualification program. Only safety-related equipment is required to mitigate accidents and bring the plant to a safe shut-down condition and as a result only safety-related equipment requires qualification.

Interrogatory No. 12

(b) If no such qualification procedures have been employed, explain in detail why not.

RESPONSE:

(b) Not applicable. See response to 12(a) above.

Interrogatory No. 12

(c) State whether all Byron "important to safety" equipment has been analyzed and qualified for the full plant life (estimated at 30-40 years), and if not, state in detail which equipment has not been and the length of time for which it has been qualified.

RESPONSE:

(c) Not applicable. See response to 12(a) above.

Interrogatory No. 12

(d) State whether all Byron safety-related equipment has been analyzed and qualified for the full plant life (estimated at 30-40 years), and if not, state in detail which safety-related equipment has not been and the length of time for which it has been qualified.

RESPONSE:

(d) Certain components which must be qualified have not been qualified for a full 40 years. The qualified life of the class 1E equipment is contained in the Edison EQ Report. A list of equipment with limited life is being prepared in conjunction with Edison's maintenance and replacement program which will be utilized at Byron. The mechanical equipment qualification program, which is being developed, will also identify limited life equipment.

Interrogatory No. 12

(e) State whether all Byron "important to safety" equipment has a qualified life established through an acceptable qualification program, and (i) if yes, identify and provide all documents relevant thereto, and (ii) if no, explain why in detail.

RESPONSE:

(e) Not applicable. See response to 12(a) above.

Interrogatory No. 12

(f) State whether all Byron safety-related equipment has a qualified life established through an acceptable qualification program, and (i) if yes, identify and provide all documents relevant thereto, and (ii) if no, explain why in detail.

RESPONSE:

(f) No. Currently, prior to receiving authorization to operate, Class 1E equipment in harsh environments must be qualified. This qualification program is fully described in the Edison EQ Report. At this time several components are still in the process of qualification. Class 1E equipment in harsh environments which is not qualified prior to fuel load will be identified and a justification for interim operation without full qualification will be provided.

Interrogatory No. 12

(g) Identify and produce all documents relied upon in or relating to your answers to Interrogatory 12.

RESPONSE:

(1) "Equipment Environmental Qualification Report, Byron/Braidwood Stations"

(2) Byron FSAR

(3) NUREG-0588

Interrogatory No. 13

Concerning Contention 108:

(a) state whether you agree that the effects of accident-related radiation releases at Byron could reach as far as 100 miles;

(1) if your answer is no, state the maximum distance you contend the effects of such radiation releases could reach and state in detail the reasons for your answers, and include all data on any Byron-specific studies which have been done or which support those reasons; or

(2) if your answer is yes, (i) indicate what provisions have been made for emergency plans for areas beyond the 50 mile EPZ, and (ii) if no such plans have been made, state with particularity why not;

RESPONSE:

(a) In general, it is extremely improbable that the effects of accident-related radiation releases of any significance would extend as far as 100 miles away from the site. However, if one postulates a highly unlikely accident scenario combined with extreme meterological conditions it is conceivable that effects of accident related radiation releases could reach as far as 100 miles.

(a)(1) Not applicable.

(a)(2)(i) Although the local emergency plan is in the process of development, Edison does not believe that it will contain any specific planning provisions for areas beyond an ingestion pathway EPZ of approximately 50 miles.

(a)(2)(ii) Specific emergency planning for areas which extend beyond the ingestion EPZ are not required by applicable federal guidance. In addition, because of the unlikelihood of significant radiation effects extending beyond the ingestion EPZ, Edison does not believe that specific planning provisions for these areas are required to protect the public safety.

Interrogatory No. 13

(b) state whether any Byron-specific accident consequence study (including any computer study) has been done to determine the adequacy of the 10 and 50-mile EPZ's and, if such a study has been done, identify and produce the data used, the program used, the assumptions used, and the results of the study;

RESPONSE:

(b) To Edison's knowledge, a Byron-specific accident consequence study to determine the adequacy of the EPZ's has not been conducted.

Intrrogatory No. 13

(c) if no such study has been done, state with particularity why not;

RESPONSE:

(c) The Illinois Emergency Service and Disaster Agency ("ESDA") selects the size of the EPZ based on federal guidance. The federal guidance is based upon consideration of accident consequences at a number of nuclear power plants. Edison does not believe that there are any special circumstances associated with Byron which would warrant a site-specific analysis to determine the adequacy of the 10 and 50 mile EPZ's.

Interrogatory No. 13

(d) state whether CECO has considered the effectiveness of using an actual consequence analysis resulting from a Class 9 accident to establish a realistic EPZ or extended EPZ for Byron, and (i) if your answer is yes, provide all data regarding that evaluation, and (ii) if your answer is no, explain in detail why not;

RESPONSE:

(d) No. Illinois ESPA, not Edison, determines the size of the EPZ. See also response to 13(c) above.

Interrogatory No. 13

(e) state whether the impact of a radiological accident at Byron has been evaluated by neighboring states, and, if so, indicate whether that evaluation included each state's emergency preparedness and planning;

RESPONSE:

(e) Since a portion of the State of Wisconsin is within the 50 mile radius surrounding Byron, Wisconsin, through its Division of Emergency Government, is involved in the development of Byron emergency plans.

Interrogatory No. 13

(f) explain in detail what provisions have been made at Byron for the possibility that, during an accident, personnel would be excluded from the EOF or other facilities due to ground dose exposure in the vicinity;

RESPONSE:

(f) The Byron annex to the Commonwealth Edison Generating Station Emergency Plan does not contain any specific provisions regarding the potential inaccessability of the Dixon EOF. If necessary, Edison would obtain of its other EOF's as a back-up facility. Edison has EOF's at Morrison, Zion, and Mazon, Illinois. In addition, its Corporate Command Center can function as an EOF. Of these EOF's, the most likely back-up facility to Dixon would be Morrison, due to its relative proximity to Byron.

Interrogatory No. 13

> (g) describe in detail what steps have been taken to insure that field monitoring teams at Byron will be capable of providing the necessary data to update dose calculations during an emergency;

RESPONSE:

(g) Field monitoring teams use a common set of procedures (Environmental Director and Environmental Group Procedures) which are tested through drills and exercises. These procedures will be made available for the League's inspection at Edison's offices. If necessary, the procedures are modified to improve team performance based on the results of these drills.

Interrogatory No. 13

> (h) state in detail what accuracy is expected for the value of radiation releases (in curies of each isotope released) which are to be used in

dose calculation or offsite doses during an accident at Byron;

RESPONSE:

(h) The effluent radiological monitoring and sampling system provides measurement, indication, and control of radioactivity in those streams which discharge to the environs outside the plant boundaries. The effluent monitoring systems provide operating personnel with a recorded measurement of the radioactivity levels present in each of the plant's air exhaust and liquid discharge systems.

In particular, the auxiliary building vent stack effluent stack monitors each consist of five detectors (air particulate, gas (low and high), iodine, and background substraction). Additional features associated with these monitors include automatic isokinetic sampling, automatic grab sampling, and tritium sampling. All detectors consist of beta scintillators, except for the iodine channel which consists of a NaI detector. Actual accuracy of the monitors as it relates to the determination of potential releases during an accident cannot be determined until equipment installation is complete since many factors such a placement of the isokinetic probes, final installed geometry of the detectors, flow rates through the stack and sampling system, etc., will affect the accuracy of the monitoring systems. The detector sensitivities will be established for all detectors through their calibration to commercial standards that have been standardized using a measurement system traceable to the National Bureau of Standards.

The Byron effluent monitoring system complies with the NRC post-Three Mile Island requirements. The monitor readouts do not provide isotopic readout information, except for I-131. The Iodine channel consists of a NaI detector and single channel analyzer that monitors an adjustable window around the major I-131 photopeak. The other effluent monitor channels measure gross beta courts. Calibration curves allow determination of detector sensitivity for most particulate or gaseous isotopes of interest. The reliance upon readouts from the installed effluent monitors provides the operator with immediate, conservative, and reasonably accurate information that will provide the basic input to computerized dose assessment procedures. Collection of grab samples following an accident and the transfer of these samples to a counting laboratory equipped with gamma spectroscopy will provide an isotopic breakdown of isotopes released to the environment. However, this information would not be available in the first few hours of an incident. This information would therefore be used for updates to initial offsite dose calculations and for historical purposes.

Interrogatory No. 13

(i) state in detail the accuracy with which iodine release (in curies of Iodine) is expected to be known during an accident

RESPONSE:

(i) The Iodine detector within the effluent radiological monitoring and sampling system consists of a charcoal cartridge assembly and lead plug on the front end of a shield and a NaI(Tl) detector assembly and lead plug on the opposite end. The sample enters the shield, passes through the replaceable charcoal cartridge, and then exits the shield. The charcoal cartridge absorbs iodine and is viewed with the NaI(Tl) integral line gamma scintillation detector. A single channel analyzer monitors an adjustable window around the major I-131 photopeak. Drift free operation is assured for the single channel analyzer via Am-241 doping of the NaI(Tl) crystal and a temperature compensation sensor. The approximate detector sensitivity is 1.01 E05 cpm/uCi for I-131. Actual accuracy of the detector system cannot be established until after equipment installation is complete.

Edison has developed an Offsite Dose Calculation System (ODCS), a computer-based method of estimating the environmental impact of unplanned airborne releases of radioactivity from nuclear stations. In developing the ODCS, Edison has adopted for use the atmospheric transport and plume gamma dose models recommended by the NRC in its Regulatory Guide Series (e.g., RG 1.23, 1.109, and 1.111) and in the publication "Meteorology and Atomic Energy" (TID-24190, July, 1968). The utilized models should be adequate for the purposes intended: to help the control operator and the ODCS operator reach a decision concerning the necessity to recommend protective actions in the vicinity of the plant during the initial phase of an accident, i.e., before field personnel are fully capable of measuring the radiation intensity from the plume, and to make a reasonably conservative estimate of radiation dose to the public. Once field personnel are dispatched and the plume's behavior is being tracked from the ground and/or air, then the role of a predictive meteorlogical model is reduced.

The report entitled, ORNL 5528, "The Uncertainty Associated with Selected Environmental Transport Models," reviews the uncertainty in atomspheric dispersion models out to a distance of 50 miles. The attached tables are extracted from the document, ORNL 5528. The tables summarize the uncertainty associated with concentration predictions made by the Gaussian plume atmospheric dispersion model. Commonwealth Edison research findings support the accuracy estimates for locations near the plant.

Table 5 An estimate of the uncertainty associated with concentration predictions made by the Gaussian plume atmospheric dispersion model[a]

Conditions	Range of the ratio $\frac{\text{Predicted}}{\text{Observed}}$
Highly instrumented flat-field site; ground-level centerline concentration within 10 km of continuous point source	0.8-1.2
Specific hour and receptor point; flat terrain, steady meteorological conditions; within 10 km of release point	0.1-10
Ensemble average for a specific point, flat terrain, within 10 km of release point (such as monthly, seasonal, or annual average)	0.5-2
Monthly and seasonal averages, flat terrain 10-100 km downwind	0.25-4
Complex terrain or meteorology (e.g., sea breeze regimes)	b

[a]T. V. Crawford (Chairperson), Atmospheric Transport of Radionuclides, pp. 5-32 in Proceedings of a Workshop on the evaluation of Models Used for the Environmental Assessment of Radionuclide Releases, ed. by F.O. Hoffman, D. L. Shaeffer, C. W. Miller, and C. T. Garten, Jr., USDOE Report CONF-770901, NTIS, April 1978.

[b]The group which assembled these estimates did not feel there was enough information available to make even a "scientific judgment" estimate under these conditions.

Table 6 Some validation results for ensemble averages predicted by the Gaussian plume model

Conditions	Range of the ratio $\frac{\text{Predicted}}{\text{Observed}}$
Annual average SO_2 concentrations for Roane Co., Tennessee; both point and area source emissions included	0.5-$\lesssim$ 2
Continuous gamma-ray measurements 0.04-6.8 km downwind of a boiling water reactor	0.33-1.78
Gamma-ray doses downwind of Humboldt Bay Nuclear Power Plant	0.5-$\lesssim$ 2
Monthly gamma-ray doses for four stations downwind of a nuclear power plant at an inland site	0.30-4.78 individual stations 1.55 mean of all data

Table 7 Validation results for Gaussian plume model predictions out to 140 km

Conditions	Range of the ratio Predicted/Observed
^{85}Kr measurements 30-140 km downwind of the Savannah River Plant	
Weekly and annual averages	0.25-4
Seasonal averages, Spring	2-4, 69% of samples 2-10, 100% of samples
Summer	0.5-4, 46% of samples 0.5-10,85% of samples
Fall	0.5-4, 31% of samples 0.5-10, 85% of samples
Winter	2-4, 69% of samples 2-10, 92% of samples
Annual Average	1-4, 77% of samples 1-10, 92% of samples
10-hour averages, six variations of the model	0.5-2, 42-65% of samples 0.1-10, 79-95% of samples

Table 8 Some validation results for Gaussian plume model predictions in speed, inversion conditions both complex terrain and also under low wind

Conditions	Range of the ratio Predicted / Observed		
Review of a number of experiments conducted in complex terrain for plume centerline concentrations	0.01-300, individual measurements close to the source		
	0.50-2, 2-15 km downwind of source		
Review of a number of experiments conducted under low wind speed, inversion conditions			
	stability category		
	E	F	G
smooth desertlike terrain[a]	2.3-10	1.3-12	3.6-20
wooded flat terrain[a]	20-25	20-40	20-30
wooded hilly terrain[a]	50-350		300-500

[a]Ratios estimated from curves provided by Van der Hoven.[41]

Interrogatory No. 13

(j) identify and produce all documents relied upon in or relating to your answers to Interrogatory No. 13.

RESPONSE:

(j) Except for the following all documents used are identified in the response to the specific interrogatory:

NUREG - 0396

IPRA

Byron Specific Annex To GSEP

Environmental Director Procedures

Environs Group Procedures

Off Site Dose Calucation Manual

Interrogatory No. 14

Concerning Contention 109:

(a) with reference to the Class 9 accident scenarios and release categories which have been postulated for Zion in its PRA which would also be applicable to Byron, what quantities of actinide isotopes have been assumed to be released during core melt accidents, specifically including, but not limited to, the released quantities of plutonium, neptunium, and americium;

RESPONSE:

(a) Absent a detailed PRA for Byron, it is not possible to establish the scenarios leading to specific release quantities of specific isotopes. Were a PRA performed for Byron, and were that PRA to use the same release categories as employed in the Zion Study, it is likely that the release quantities for a given release category would be similar with slight adjustments for the variance in power level between the plants. The Zion Study, including release category source terms and uncertainty histograms, is available as noted in response to Interrogatory 2(c).

Interrogatory No. 14

(b) Identify with particularity and provide a detailed geologic map of the rock outcroppings located in or near the Rock River in the vicinity of the Byron site.

RESPONSE:

(b) Surficial geologic maps of the site vicinity are available and depict the overburden materials and isolated bedrock exposures in or near the Rock River (SCS, 1980; ISGS, 1978). Extensive geological investigations by Edison and the Illinois State Geological Survey have been made throughout the site vicinity, and rock outcrops have been studied and described during the course of this work. Discussions of bedrock conditions in the site vicinity are provided in the text of the FSAR and attachments. Significant bedrock exposures are identified on several maps in the FSAR (Figures 2.5-16, 2.5-20, and 2.5-31). While not compiled on one map, sufficient documentation of rock outcrops in or near the Rock River exists on the maps noted above, and the significance of bedrock geology in the site vicinity is adequately provided in the FSAR text.

Interrogatory No. 14

(c) state with particularity all data concerning any model which has been used to measure radionuclide migration into the groundwater, and in particular include information on the assumptions used regarding chemical reactions with and/or retardation of radionuclides by material of the rock underlying the Byron site;

RESPONSE:

(c) As described and analyzed in Sections 2.4.12 and 2.4.13.3 of the Byron FSAR any accidental release of radioactive effluent to the groundwater would be released from the Auxiliary Building to the nearest down gradient offsite well. This well is located approximately 1,960 feet from the Auxiliary Building.

The analysis considers the dilution of the radioactive effluent in the groundwater and its time of travel according to Darcy Law for flow through the groundwater. To be conservative, the analysis does not consider the effects of dispersion, adsorption and ion exchange. In addition, no chemical reaction or retardation of radionuclides in the rock underlying the Byron site were considered in the analysis.

Interrogatory No. 14

(d) identify and produce all documents relied upon in or relating to your answers to Interrogatory 14.

RESPONSE:

(d) Soil Conservation Service, 1980, Soil survey of Ogle County Illinois

Illinois State Geological Survey, 1978

Soil Geomorphology of North Eastern Illinois

Special Publication Guide Book

Joint Field Conference of Soil Science

Society of American and Geological Society of America - Open File Report.

Interrogatory No. 15

Concerning Contention 111:

(a) state specifically all data concerning provisions made for calculating radiation dosage at Byron for the widely varying radiosensitivity to cancer induction by ionizing radiation which is found in a heterogenous population;

RESPONSE:

(a) Data on sensitive subpopulations provide at once one of the strongest direct pieces of evidence for the existence and importance of repair (and hence of dose-rate dependence) in radiation carcinogenesis in man and the identification of (fortunately quite small) groups which apparently are abnormally sensitive to radiogenic cancer.

It is well known that differences in sensitivity to radiogenic cancer occur as a function of age and hence sensitive subpopulations do exist on this basis. Claims have been made additionally by Bross, on the basis of epidemiological data, that such groups may exist on the basis of other conditions or diseases present (i.e., allergy prone, virus infection of mother while individual was in utero, etc.), but Bross' claims have been shown to be unsupported by the data.

A notable development during the past decade is the increasing recognition that there are human genotypes that confer both increased susceptibility or resistance to DNA damage and increased cancer risk after exposure to carcinogenic agents, including ionizing radiation. The role of constitutional susceptibility to cancer induction is not well enough understood, however, for it to be used as a factor to modify risk estimates (NAS-BEIR, 1980). Inasmuch as the risk estimates developed for the BEIR Report are averages for large populations that presumably include many genotypes, it is unlikely that the present risk estimates would be notably altered if data representing very small subsets of abnormally radiosensitive persons could be recognized and excluded from the calculations of the NAS-BEIR (1980) Committee. If population subsets can be identified as being at substantially greater risk of radiation carcinegenesis, and at the present this has not been the case, their risk will require separate estimation. At the present time, the incidence and sensitivity differentials of these diseases, particularly for ionizing radiation, appear to be so low that even their increased sensitivity to radiation would not be likely to influence detectably the dose-effect response of a large random population containing a normal number of such individuals. Accordingly, account has been taken in present radiation protection guides to protect the susceptible subpopulations and protect the various susceptibilities of the tissues and organs of the body to cancer-induction.

Interrogatory No. 15

> (b) state specifically what plans or provisions CECO has made for monitoring by air the micro-meteorological patters of ground passage and radioactive fallout following Byron plant accidents involving releases of radiation of the air pathway;

RESPONSE:

(b) Edison has developed a set of computer programs for calculating the offsite doses resulting from releases. However, in the event of an accident, field assessment techniques will be used as the principal method for measuring the levels of radioactivity. The computer programs will be used principally to make approximate estimates of the locations and magnitude of radionuclide concentrations.

Interrogatory No. 15

> (c) state specifically the plans which CECO has developed for training the public, and in particular public officials such as police and firemen, for procedures to be followed during a radiological emergency at Byron in order to reduce radiation exposure to the public;

RESPONSE:

(c) It is expected that training of local governmental officials will be conducted by the Illinois Emergency Services and Disaster Agency, the Department of Nuclear Safety, and Edison. The scope of Edison's responsibility with respect to this training has yet to be determined.

In past exercises at Edison's other stations, ESDA and DNS have provided various training courses for local emergency response personnel. Edison has participated in the courses when requested by providing representatives to answer questions concerning its emergency plan and provide general background information regarding the plant in question. Edison is in the process of developing an informational pamphlet which will be distributed to members of the public regarding steps and precautions which should be taken in the event of an emergency.

Interrogatory No. 15

(d) state specifically the reasons for calculating internal dose and dose commitments at Byron to periods typically of 50 years, where the current life expectancy is approximately 70 years;

RESPONSE:

(d) Edison has adopted the dose committment factors published by the NRC. These factors assume that the hypothetical individual is twenty years of age and his life expectancy is approximately seventy years. Thus, the dose calculation covers a period of fifty years.

Interrogatory No. 15

(e) state whether you agree that the acceptable radiation level for the Byron plant when operating in conformance with ALARA should be one millirem per year, and give detailed reasons for your answer;

RESPONSE:

(e) Byron will be operated to meet applicable federal and state limits. It is expected that under most circumstances, the calculated dose to an individual living offsite will be approximately one millirem.

Interrogatory No. 15

(f) state whether you agree that Byron should have a minimum of 50 off-area monitoring stations equiped with air samplers, fallout trays, gummed paper collectors, and rain water collectors to evaluate the alpha as well as the beta and gamma activity, and (i) if your answer is no, give detailed reasons for your answer; (ii) if your answer is yes, state with specificity what plans CECO has to establish such monitoring stations and the number of such stations planned;

RESPONSE:

(f) Edison disagrees. Sufficient monitoring can be accomplished with far less than fifty stations. For the Byron site, Edison believes that approximately eight monitoring sites are needed. Locations of monitors for measuring airborne iodine and particulate radioactivity and gamma radiation from the noble gases will be placed near population centers and at a few other locations so as to give a uniform distribution around the site.

Interrogatory No. 15

(g) state whether you agree that NTA thick emulsion film monitoring is insufficient for a personnel neutron monitoring program at Byron, and (i) if your answer is no, explain your answer in detail; (ii) if your answer is yes, explain in detail what other monitoring techniques CECO is planning to use, including but not limited to electro-chemical etching of polycarbonate foils and CR-39 foils;

RESPONSE:

(g) Edison intends to use CR-39 personnel neutron dosimeters. The neutron energy threshold of this detector is much lower than the NTA film emulsions and exhibits good stability over time.

In addition, neutron monitoring is performed with an REM-meter. The response characteristic of this instrument for the neutron energy spectrum at commercial nuclear power plants is well documented. Individuals required to enter neutron radiation fields are timekept based on the REM-meter measured dose rates and the time spent in the area. That is, a neutron dose is calculated. This calculated dose is used to augment the dose information obtained from the neutron dosimeter. The use of a calculated neutron dose equivalent to supplement the neutron dosimeter is consistent with NRC Regulatory Guide 8.14 "Personnel Neutron Dosimeters."

Interrogatory No. 15

(h) (i) explain with particularity the methods CECO is planning to use at Byron for: (1) identifying shortlived iodine and noble gases; (2) identifying the chemical form of radioiodine; (3) distinguishing between airborne gases and particulates; and (4) measuring quantitatively the carbon-14;

RESPONSE:

(h) (i) (1 & 2) The following is a listing of the Byron Chemistry procedures which will identify iodine, particulates and gas radionuclides using the EG&G ORTEC

Ge(Li) detectors model 18011-10185-S with a EG&G ORTEC MCH model #7040 connected to a Digital Equipment Corp. 11/44 computer. Each procedure is identified as to whether it has been drafted or identified. The procedures are as follows:

Procedure Number	Procedure Name	Draft	Identified
BCP 200-3	Approval of Automated Analytical Instrumentation System Chemical Procedures	X	
	Appendix A		
	AAIS-CCP-0001 General Radionuclide Analysis of a Liquid Sample		
	AAIS-CCP-0002 General Radionuclide Analysis of a Gas Sample		
	AAIS-CCP-0003 Particulate Radionuclide Analysis		
	AAIS-CCP-0004 Iodine Radionuclide Analysis		
	AAIS-CCP-0023 Soluble Gas Radionuclide Analysis		
	AAIS-CCP-0024 Gas Waste Radionuclide Analysis		
	AAIS-CCP-0034 Chimney Effluent Radionuclide Analysis		
	AAIS-CCP-3001 GE Detector Efficiency Calibration		
	AAIS-CCP-2001 MCA Performance Test		

The following is a listing of the Byron Chemistry procedures to make gross Beta and Alpha measurements using Carberra proportional counters model #2201 and #2201s. Each procedure is identified as to whether it has been drafted or identified.

Procedure Number	Procedure Name	Draft	Identified
BCP 200-1	Manual Operation of the Canberra Proportional Counter	X	
BCP 500-16	Manual Calibration of the Canberra Proportional Counter		X
BCP 200-3	Approval of Automated Analytical Instrumentation System Central Chemical Procedures	X	

Appendix A

AAIS-CCP-0051	Gross Alpha Activity Analysis of a Liquid Sample
AAIS-CCP-0052	Gross Beta Activity Analysis of a Liquid Sample
AAIS-CCP-0053	Gross Beta Acitivity Analysis of an Air Particulate Filter Sample
AAIS-CCP-2202	Proportional Counter Performance Test
AAIS-CCP-3002	Proportional Counter Efficiency

The following is a listing of the Byron Chemistry procedures which pertain to gross tritium analysis using the Packard Liquid Scintiallation Counter Model Tricarb 2660. Each procedure is identified as to whether it has been drafted or identified.

Procedure Number	Procedure Name	Draft	Identified
BCP 500-19	Calibration of Packard Liquid Scintiallation Counter, Tri-Carb 2660	X	
BCP 220-1	Operation of the Packard Liquid Scintation Counter, Tri-Carb 2660	X	

(3) Normally, airborne gases will be sampled by using a glass air sampler and a radionuclide analysis would be performed identifying the noble gases collected in the

sampler. In addition normal particulate samples will be collected by utilizing a particulate filter paper and a radionuclide analysis or gross beta analysis would be performed on the particulate filter paper. Thus, airborne gases and particulates would be distinguished by utilizing two different sample collection devices and analyzing each for radionuclide identification.

(4) Carbon 14 analysis can be performed on the Packard Liquid scintillation counter. However, there are no requirements to perform Carbon-14 analysis at Byron Station.

Interrogatory No. 15

(ii) if no monitoring systems, as described in subpart (i) above, are planned, state in detail in reasons that no such monitoring will be conducted;

RESPONSE:

(h) (ii) Not applicable.

Interrogatory No. 15

(i) state whether you agree that it is unsatisfactory to measure only absolute values of alpha, beta, and gamma dose levels at Byron, and (i) if your answer is yes, specify in detail what CECO is doing to measure the emissions of individual radionuclides at Byron; (ii) if your answer is no, give detailed reasons for your answer;

RESPONSE:

(i) Byron Station will measure absolute values of Alpha, Beta and gamma dose levels and will routinely sample, analyze and quantify emissions of radionuclides in the effluent pathways from the Station. In addition, concentrations of specific radionuclides will be measured as discussed in the response to Interrogatory 15(h) above.

Interrogatory No. 15

(j) identify and produce all documents relied upon in or relating to your asnwers to Interrogatory 15.

RESPONSE:

(j) All documents used are identified in the response to the interrogatory.

Interrogatory No. 16

Concerning Contention 12:

(a) state whether you agree that spreading a given level of person rems across progressively large numbers of people results in an increasing number of malignancies, and (i) if your answer is no, give detailed reasons for your answer, (ii) if your answer is yes, explain in detail the reasons for the expected utilization of large numbers of transient workers at the Byron Plant;

RESPONSE:

(a) Edison does not agree with this statement. This is another way of saying that the risk per rem at low doses of low-LET radiation is greater than for high doses, so that the linear, no-threshold dose-response relationship underestimates the risk at low doses, and is therefore not conservative. There is no convincing scienti ic evidence for this.

Consideration of repair and recovery of radiation injury in the cells and tissues of the body, and of dose-rate effectiveness factors (NCRP, 1980) leads to the conclusion that the linear hypothesis generally overestimates the risk. The risk per unit dose of low-LET radiation for cell killing and the induction of chromosome aberrations, mutations, teratogenic effects, tumor formation, and shortening of life has been observed in experimental systems to depend consistently upon both the magnitude of the dose and its temporal distribution. In general, the dose-response curves for low-LET radiation for late (carcinogenesis) and genetic effects increase in slope with increasing dose and dose rate. Thus, linear interpolation between the naturally-occurring spontaneous incidence and the incidence observed following exposure at intermediate-to-high doses and dose rates generally overestimates the risk of low-LET radiation at low doses and low-dose rates. This observation has also been incorporated in reports by the ICRP (1977), NCRP (1980) and UNSCEAR (1977).

The existence of dose-rate effectiveness factors has long been recognized from clinical experience and from studies of both genetic and somatic effects in experimental animals. From the studies on somatic effects in animals (NCRP, 1980), the effectiveness per unit dose of low-LET radiation for cancer induction is lower at low doses and low-dose rates that at high doses and high-dose rates. The effectiveness per unit dose of high- vs. low-dose and dose rate exposure ranges from a factor of about 2 to about 20. In other words, linear interpolation from high doses (150 to 350 rads) may overestimate the effects of either low doses (0-20 rads or less) or of any dose delivered at dose rates of the order of 5 rad per year or less by a factor of 2 to 10. This factor is referred to as the Dose Rate Effectiveness Factor.

Although extensive data from human beings permit reasonable risk assessments to be made for exposures to intermediate to high doses of low-LET radiation, these data are not adequate to demonstrate conclusively that a dose rate effect either does or does not exist. The experimental evidence from many different biological effects, including carcinogenesis, and for many species of animals in support of a dose rate effect is so extensive, however, that it would be extraordinary if such dependence did not apply to the same endpoints in the human being as well. Because of the complexity and wide spectrum of the tumorigenic responses to radiation in the experimental animal, however, the NCRP is reluctant at this time to go beyond providing a range of factors within which a single factor for the total yield of tumors in man after exposure of the whole body probably would lie. The DREF range is 2 to 10, when the actual absorbed dose is 20 rads or less, or the dose rate is 5 rads per year or less.

The scientific evidence strongly supports a family of dose-response models for radiation carcinogenesis in animals and in man. The favored dose-response models for carcinogenesis from low linear energy transfer radiation are illustrated in recent NAS-BEIR Report (1980): a general model, with a linear term to represent one-hit kinetics, a quadratic term for two-hit kinetics, and an exponential term that brings the curve down to represent the cell-killing effect at high doses so often seen in experimental work with animal models. When the National Academy of Sciences BEIR Committee (NAS-BEIR, 1980) used these models for low linear energy transfer radiation, plus a linear term for high linear energy transfer radiation, to estimate the leukemogenic effect of low linear energy transfer radiation from the Japanese atomic-bomb survivor experience, they found that: 1) any one model fits about as well as the next; 2) in the low-dose region estimates for the effect of low linear energy transfer radiation based on the linear model are only about twice those based on the linear-quadratic model.

Although the weight of the experimantal evidence generally favors the linear-quadratic dose-response model for low linear energy transfer radiation (NCRP, 1980), extrapolation from mouse to man is hazardous and, for breast cancer, at least, the human data provide fairly strong support for the linear model. Moreover, where the level of uncertainty is high, and human life and health are at stake, a conservative choice of model is indicated. The linear model has the advantage that the scientific uncertainty about dose-response models concerns chiefly the region lying below the linear regression line. The simplicity and ease of application of the linear model are important advantages. Further, since the use of the linear model does not require observations over a wide range of dose, it obviates the necessity for depending so heavily on the experience of only one epidemiological survey, such as the Japanese atomic-bomb survivors. The linear model is a more flexible tool, permitting use to be made of all available epidemiological data representing different exposure situations and populations. Application of the linear model for radiological protection of workers in the workplace or the general population is prudent. Under this model, the risk of radiation-induced cancer remains the same derived from the population collective dose equivalent --- in other words, the risk is the same whether 100,000 people receive a dose of 1 rem, 10,000 people receive a dose of 10 rems, or 1,000 people receive a dose of 100 rems. (See also response to Interrogatory 8(b).)

Interrogatory No. 16

(b) describe in detail what design changes have been made on the Byron steam generators to reduce the frequency with which maintenance is required and to eliminate the need for their replacement or to allow replacement without occupational exposure;

RESPONSE:

(b) see Response to interrogatory 4(d) above.

Interrogatory No. 16

(c) describe in detail any proposed educational program on radiation protection and the effects of radiation exposure, including genetic, teratogenic, and somatic effects, which will be offered to or required of all Byron employees;

RESPONSE:

(c) All Byron Station employees who enter a radiation area unescorted will initially receive the Nuclear General Employee Training Program (N-GET) which addresses radiation protection, the effects of radiation exposure, and protective clothing requirements. The outline and lesson plans for the radiation protection and protective clothing sections of N-GET are available for the league's inspection. Retraining of employees will be conducted annually. This program will be implemented at Byron Station beginning in early 1983. At present, this course is being reviewed at the corporate level to ensure it is fine tuned and tailored to all generating stations. Test questions are being developed to evaluate a student's comprehension of the material presented.

Interrogatory No. 16

(d) describe in detail any prospective program for fecal analyses, differential blood counting, wound decontamination, and lense opacity examination of Byron plant workers;

RESPONSE:

(d) Byron Station has approved procedure BRP 1340-1 "Personnel monitoring for Internal Radioactive Contamination" which will be implemented to evaluate internal radioactive contamination and addresses fecal analysis requirements. Wound decontamination will be performed as described in Byron Radiation Procedure (BRP) 1470-1 "Personnel Decontamination" and documented on BRP 400-T1 "Personnel Contamination Report". Differential blood counting and lens opacity examination of Byron plant workers are not routinely performed. If required, specific recommendations for such examinations would be requested from Radiation Management Corporation, Edison's professional health physics consultant.

Interrogatory No. 16

(e) describe in detail any plans which have been made for dry runs prior to any "hot" operations and/or emergency procedures to be followed by Byron plant personnel in the event of an emergency;

RESPONSE:

(e) Byron plant management is committed to and responsible for maintaining personnel exposures to radiation as low as reasonably achievable (ALARA). It is expected that this goal will be achieved through the implementation of the ALARA Program. The Byron Station commitment to ALARA is stated in Byron Administrative Procedure (BAP) 700-1. This procedure describes the individual and departmental responsibilities for maintaining personnel exposures as low as reasonably achievable. BAP 700-2 describes the methods used to review job assignments for exposure control. ALARA Program (BAP 700-1) and ALARA Review Procedure (BAP-700-2) is available for the league's inspection. As stated in BAP 700-2 Section F.2.h, special training sessions are to be considered if exposure time can be reduced through increased efficiency of work performed. BAP 700-1 section F.6.C requires the Maintenance Department to evaluate jobs which will be routinely (quarterly, semi-annually, or annually) performed and provide training and build "mock-ups" to be used to enhance worker performance.

The following is a list of the Emergency Response Implementing Procedures which are being developed and will be used to direct emergency actions:

Station Group Director's Duties

BZP 100-1	Supervision of Emergencies, Exercise and Drills
BZP 100-2	References to other Applicable Station Procedures
BZP 100-T1	Station Director - Checklist
BZP 100-T2	Operations Director - Checklist
BZP 100-T3	Technical Director - Checklist
BZP 100-T4	Maintenance Director - Checklist
BZP 100-T5	Stores Director - Checklist
BZP 100-T6	Administrative Director - Checklist
BZP 100-T7	Security Director - Checklist
BZP 100-T8	Rad/Chem Director - Checklist
BZP 100-T9	Emergency Incident Data Sheet
BZP 100-T10	Record of GSEP Activities

Emergency Conditions

BZP 200-A1	Byron Emergency Action Levels

Emergency Measures

BZP 310-1	Initial Notifications and GSEP Responses (Primary Responsibility - Station Director)
BZP 310-2	Nuclear Accident Report Form (Primary Responsibility - Station Director)
BZP 310-3	On-Going Emergency Communications (Primary Responsibility - Station Director)
BZP 310-4	Assembly and Evacuation of Personnel (Primary Responsibility - Station Director)
BZP 320-1	Fire Fighting (Primary Responsibility - Operations Director)
BZP 320-3	Area High Radiation (Primary Responsibility - Operations Director)
BZP 320-4	High Airborne Activity (Primary Responsibility - Operations Director)
BZP 320-5	High Smearable Radioactive Surface Contamination (Primary Responsibility - Operations Director)
BZP 320-6	Emergency Treatment of Injured Personnel (Primary Responsibility - Operations Director)
BZP 380-1	Emergency Dose Limits and Radiological Controls for Rescue and Recovery Operations (Primary Responsibility - Rad/Chem Director)
BZP 380-2	Rad/Chem Response to Personnel injuries and Serious Contamination or Exposures (Primary Responsibility - Rad/Chem Director)
BZP 380-7	Estimation of Offsite Dose from an Unplanned Release of Radioactive Effluents (Primary Responsibility - Rad/Chem Director
BZP 380-8	Use of Potassium Iodide (KI) as a Thyroid Blocking Agent (Primary Responsibility - Rad/Chem Director)

BZP 380-9	Initiation of Environmental Monitoring Activities by the Rad/Chem Director (Primary Responsibility - Rad/Chem Director)
BZP 380-10	Post Accident Sampling of Reactor Coolant, Radwaste and Containment Air-General (Primary Responsibility - Rad/Chem Director)
BZP 380-11	Post Accident Sampling of Undiluted Reactor Coolant (Primary Responsibility - Rad/Chem Director)
BZP 380-12	Post Accident Sampling of Diluted Reactor Coolant (Primary Responsibility - Rad/Chem Director)
BZP 380-13	Post Accident Sampling of Undiluted Liquid Radwaste (Primary Responsibility - Rad/Chem Director)
BZP 380-14	Post Accident Sampling of Diluted Radwaste (Primary Responsibility - Rad/Chem Director)
BZP 380-15	Stripped-Gas Sampling of Post Accident Reactor Coolant (Primary Responsibility - Rad/Chem Director)
BZP 380-16	Post Accident Diluted Reactor Coolant/Radwaste Sample Disposal (Primary Responsibility - Rad/Chem Director)
BZP 380-17	Post Accident Sample Transfer (Primary Responsibility - Rad/Chem Director)
BZP 380-18	Post Accident Sampling of Containment Atmosphere (Primary Responsibility - Rad/Chem Director)
BZP 300-A1	State of Illinois Nuclear Accident Reporting System Form
BZP 300-A2	Recommended Protective Actions for Gaseous Release
BZP 300-A3	Byron Station Onsite Assembly Areas
BZP 300-A4	Byron Station Evacuation Routes and Offsite Relocation Centers
BZP 300-A5	Guidance for Augmentation of the Onsite Emergency Organization within 60 minutes
BZP 380-A1	Iodine-Dose Equivalent to Thyroid Reference Reg. Guide 1.109

BZP 380-A2	Wind Direction Data
BZP 380-A3	Locations of Fixed Environmental Radiological Monitoring Stations Air Samplers
BZP 380-A4	Byron Station Environmental Sampling Sites
BZP 380-A5	Conservative Offsite Dose Estimates Gaseous Release
BZP 380-A6	Conservative Offsite Dose Estimates Liquid Releases
BZP 380-A7	Post Accident Sample Transport Routes
BZP 380-T1	Patient Radiation and Medical Status Record Sheet
BZP 380-T2	Dosimetry Issue Log Emergency Personnel Entry into Plant Site or Controlled Area

Facilities and Equipment

BZP 400-1	The Role and Staffing of the Technical Support Center (TSC) (Primary Responsibility - Station Director)
BZP 400-2	Role and Staffing of the Operational Support Center (OSC)
BZP 400-3	Communication System Operation (Primary Responsibility - Operations Director)

Maintaining Emergency Preparedness

BZP 500-1	Operations Checks of Communications System (Primary Responsibility - OPerations Director)
BZP 500-2	Inventory of First Aid Supplies (Primary Respcnsibility - Rad/Chem Director)
BZP 500-3	Inventory of Personnel Decontamination Supplies (Primary Responsibility - Rad/Chem Director)
BZP 500-4	Inventories of Emergency Supplies and Equipment (Primary Responsibility - Rad/Chem Director)
BZP 500-A1	No. 36 Unit First Aid Kit Inventory List
BZP 500-T1	Check List - Communications Checks

BZP 500-T2	Byron Station Stretcher and Blanket Inventory
BZP 500-T3	Checklist - Personnel Decontamination Supplies
BZP 500-T4	Byron Station First Aid Kit Inventory
BZP 500-T5	Byron Station First Aid Cabinet Inventory
BZP 500-%6	Checklist - Technical Support Center
BZP 500-T7	Checklist - Operational Support Center
BZP 500-T8	Checklist - Environmental Monitoring Supplies
BZP 500-T9	Checklist - Support Hospital Supplies
BZP 500-T10	Checklist - Inventory of the Emergency Operations Facility
BZP 500-T11	Checklist - Inventory of Control Room Area Emergency Supplies
BZP 500-T12	Checklist - Inventory of Ambulance Equipment and Supplies

Telephone Directories

BZP 600-A1	Prioritized Call Listing for Staff Augmentation Purposes
BZP 600-A2	Directors Phone List
BZP 600-A3	Station Directory
BZP 600-A4	GSEP Telephone Directory

As stated previously, the GSEP and GSEP implementing procedures are used in conjunction with applicable departmental procedures. The following is a description of the Operations Department Emergency Response Procedures:

<u>Emergency Operating Procedures (BEP)</u>

BEPs are a four procedure set that initiate operator action based upon either a reactor trip or safety injection. They diagnose and mitigate the immediate consequences of a LOCA, SGTR, (Steam Generator Tube Rupture) and LOSC (Loss of Secondary Coolant).

Event Specific Subprocedures (BEP ES)

If certain conditions are met or exceeded in the BEPs, the BEP ES will direct actions to accomplish the given objective or supply new actions based on observed conditions.

Emergency Contingency Actions (BCA)

BCAs are procedures provided due to a direct NRC requirement without regard to their combined failure probability. These include, ATWS, loss of all AC, SGTR contingencies, and others.

Abnormal Operating Procedures (BOA)

BOAs provide guidance to the operator when important parameters or systems are in jeopardy, but RPS or SI have not actuated.

Critical Safety Function Status Trees (BST)

The BSTs are a set of six decision trees that evaluate the six critical safety functions to determine if the function is intact or being challenged. If challenged it will reference the restoration guideline for restoring the function.

Functional Restoration Procedures (BFR)

The BFRs direct operators' action to recover/ restore the degraded safety function dependent on which CSF is challenged and the extent of degradation.

Plant Security actions during emergency conditions are addressed in the Security Contingency Action Procedure Manual.

The interface between the GSEP and the Security Contingency Action Plan is basically one of parallel operation. The plans are compatible. The GSEP emergency response measures, once initiated, are executed in parallel with measures taken in accordance with the Security Contingency Action Procedures.

The Nuclear Station Security Plan, Appendix C, Contingency Events, identifies situations which could be initiating conditions for GSEP response measures. The Station Security Plan provides guidance for decisions and actions to be taken for each security contingency event. As guidance, the Security Plan allows for differing responses depending upon the assessment of the actual situation within each contingency event classification.

The assessment of any security contingency event and the decision to initiate, or not to initiate the GSEP will be the responsibility of the Station Director or the Shift Engineer acting as the Station Director. All identified security contingency events have the potential of being assessed as initiating conditions for an emergency declaration under the GSEP.

Additional procedures that address emergency situations and direct station personnel are as follows:

BAP 300-9	Oil Spill to the Flume or on the Construction Run-off Pond - Corrective Action
BAP 1100-10	Implementing Procedures for Fire (Fire Marshall)
BAP 1100-11	Implementing Procedure for Fire (Fire Chief)
BAP 1100-12	Implementing Procedure for Fire (Fire Officer)
BAP 1100-13	Implementing Procedure for Fire (Fire Brigade)
BAP 1100-14	Implementing Procedure for Fire (Fire Company #1/Maintenance Personnel)
BRP 1740-1	Radiation Protection Practices During Accident Conditions

Interrogatory No. 16

(f) describe in detail any provisions which have been made for only assigning plant workers beyond childbearing age to "hot" operations;

RESPONSE:

(f) Byron Station does not have any provisions for assigning only plant workers beyond childbearing age to "hot" operations. Assignment of workers to jobs in radiation areas will comply with applicable state and federal radiation exposure limits as well as corporate limits specified in the "Radiation Protection Standards" BRP 1000-1 and BRP 1000-A1. Radiation exposure control to "declared" pregnant females will be as described in Reg. Guide 8.13 "Instruction Concerning Prenatal Exposure" and is discussed in the Nuclear General Employee Training program as follows:

4. Reg. Guide 8.13 Instruction Concerning Prenatal Exposure

    a. It is the responsibility of the employer to keep radiation exposures as low as is reasonably achievable and to take all practical steps to reduce the radiation exposure of fertile women employees. However,

        1. Entering restricted areas within the nuclear plant may result in individuals receiving radiation doses up to the allowable Radiation Worker limits (5000 millirems per year).

        2. An embryo or fetus should not be exposed to more than 500 millirems during its 9-month period of development (or 167 millirems per trimester).

    b. Available Alternatives for the pregnant female worker:

        1. Decide not to accept or continue assignments in areas where radiation levels are high enough for a baby to receive 500 millirems or more before birth.

        2. Reduce the worker's exposure, where possible, through use of time, distance, and shielding.

        3. Ask her employer to reassign her to areas involving less exposure to radiation. If this is not possible, she might consider leaving her job.

4. Delay having children until she no longer works in areas where the radiation dose to the unborn child may exceed 500 millirems.

5. Choose to continue working in higher radiation areas with the full awareness that she is doing so at some small increased risk for her unborn child.

c. Additional Points:

1. The embryo is most sensitive during the first three months of development, so the choice of alternatives should be made early and quickly.

2. The actual increased risk of damage to the unborn child is very small.

3. Exposure to the low levels of radiation within the nuclear plant will not affect a woman's child-bearing ability.

Interrogatory No. 16

(g) explain in detail all provisions which have been made for recordkeeping and the computerization of records of worker radiation exposure at Byron, including but not limited to recordkeeping with regard to: alpha, beta, gamma, fast neutron, thermal neutron, epithermal neutron, urine and feces analyses; medical records; potential and actual radiation incidents; skin and clothing contamination; any diagnosis of malignancy; birth defects; and the confidentiality and availability to workers of such records; and

RESPONSE:

(g) As specified in Appendix A of the Byron/Braidwood FSAR, the "Occupational radiation exposure record system is based on Regulatory Guide 8.7, Revision-May 1973."

Regulatory Guide 8.7 states in part that "Radiation exposure records must be maintained in accordance with the requirements of 10 CFR 20.401." Accordingly, Byron Station's health physics recordkeeping program is developed in order to comply with the requirements of 10 CFR 20.401. As required, records must be maintained for the following general subject areas:

a. Radiation exposures of individuals on Form NRC-5; and

b. Results of surveys, monitoring, and disposals. Computer Dosimetry Program.

Each Edison nuclear station has the responsibility of monitoring and recording radiation doses received at the respective nuclear station. The corporate Technical Services, Nuclear group has the responsibility for coordinating the dosimetry program among participating nuclear stations, the computer system staff, contractors who provide dosimetry services, and the data processing department. Each radiation worker is required to complete an NRC Form-4. The NRC Form-4 records the pertinent facts concerning each individual (name, address, social security number, age, sex, previous occupational exposure, etc.). The NRC Form-4 serves as registration form for data entry into the Commonwealth Edison Computer Dosimetry System. Each radiation worker is instructed to read the Privacy Act Statement provided on the reverse side of the NRC Form-4. The Privacy Act Statement includes an explanation of the routine authorized uses of the information provided as well as for the exposure data that will be generated for each individual. The nuclear station records the external radiation exposures received by using a combination of "pencil" and badge type dosimeters. Badges are normally issued on a bi-weekly basis. At the end of each bi-weekly period, the used badges are collected and sent for processing. The processor then supplies Technical Services, Nuclear with a printed listing and a punched card deck containing radiation data collected from the processed badges. Pencil dosimeters enable the station to record an estimate of daily exposures received by personnel at the station. Each day, pencil exposure data are transmitted to the central computer via a teleprocessing unit, and a report is generated which gives the estimated current exposure status for all personnel separated by work group. The estimated current exposure status is based on the latest badge results plus all daily pencil/ timekeeping data recorded since the issuance of the last processed badge results. At the end of each bi-weekly exposure period, processed badges are matched with their corresponding pencil dosimeters and a bi-weekly radiation exposure report is generated. This bi-weekly exposure report contains for each individual: the person's name, social security number, badge number, bi-weekly period ending date, and badge results. An NRC Form-5 is generated for each individual during the current year in accordance with 10 CFR 20. Data on the NRC Form-5 consists of selected NRC Form-4 data, bi-weekly data, direct and indirect biomasses data as well as related commentary data. Included with this response is a complete description of the "Microfiche Output of Bi-Weekly Computerized Dosimetry Reports (NRC Form-5)."

The Radiation Evaluation Program (REP) was initiated by Commonwealth Edison in April, 1977. REP is a computer based occupational exposure accounting system used to document, by work group, the dose expenditure resulting from work performed on various plant systems and components. In addition to each work group's exposure and the plant component worked on, the Program documents the total work effort in person-hours and includes a brief description of the work performed.

Station approved procedure BRP 1480-1 addresses the requirements for performance and recording of contamination surveys. Control of surface contamination is necessary to limit dose rates and resuspension in air of loose radioactive material that may enter the body through inhalation, ingestion, or skin absorption. BRP 1480-1 addresses the performance of the following types of surveys:

a. Area Smear Surveys in the Plant
b. Equipment, Tools, and Radioactive Shipment Surveys
c. Personnel Surveys
d. Routine Clean Area Surveys
e. Sealed Source Leak Tests
f. Environmental and Special Surveys.

The following data sheets are used to record surveys performed above, as appropriate:

a. BRP 1400-T1, Personnel Contamination Report
b. BRP 1400-T2, Indirect Contamination Survey Record Form
c. BRP 1400-T3, Radiation-Contamination Survey
d. BRP 1500-T1, Radioactive Material Receipt Checklist
e. BRP 1500-T2, New Fuel Radiation Survey
f. BRP 1500-T3, Radioactive Shipment Arrival Survey Form
g. BRP 1500-T4, Radioactive Shipment Departure Survey Form
h. BRP 1600-T4, Source Leak Test Record
i. Byron Station Plant Survey Sheets.

Survey results are reviewed by health physics management and are then placed in the Station Files for retention and future use as required.

Station approved procedure BRP 1610-1 addresses the control and inventory of radioactive sources at the station. Section F.5 of this procedure states that radioactive sources will be processed for disposal when they are broken, damaged, or otherwise declared unusable. BRP 1610-1 requires that data sheet BRP 1600-T3, Disposal of Radioactive Sources, be completed for each disposal. Records of disposal are maintained until the NRC authorizes their disposition.

Interrogatory No. 17

(a) Separately with respect to each of the League's Revised Contentions Nos. 1A, 8, 19, 22, 28, [illegible], 34, 39, 41, 42, 47, 53, 54, 61, 62, 63, 71, [illegible], 106, 108, 109, 111, and 112, state in specific detail:

(i) Do you agree that each such Revised Contention is related or applicable to, in whole or in part, a consideration of continued construction and/or permission to operate each or both of the Byron Units? If your answer to this question with respect to any

Revised Contention is yes, please explain your answer in detail. If your answer to this question is no with respect to any Revised Contention, please explain your answer in detail, including all factual and other reasons why you believe each such Revised Contention is unrelated or inapplicable to Byron Units;

RESPONSE:

(a) (i) In as much as the contentions in question have been admitted by the Licensing Board, the contetions must, at this time, be deemed related and/or applicable to Edison's application for a license to operate the Byron facility. As information is gathered during the course of discovery, it may become apparent that the contentions consitute an attempt to challenge NRC regulations, raise matters which are not relevant to the decision whether to grant an operating license for Byron, or are otherwise legally objectionable. Any such objections will be presented in due course.

To the extent the interrogatory seeks to assess the applicability of the contentions to considerations of continued construction of the Byron units, it seeks information which is neither relevant to nor likely to lead to relevant information concerning issues within the scope of this proceeding. Therefore, Edison objects to this portion of the interrogatory.

Interrogatory No. 17(a)(ii)

(ii) With respect to each "no" answer in (i) above state in specific detail whether it is your position that the problem or issue raised by each such Revised Contention is totally inapplicable and unrelated to the Byron Units, in the sense that no consideration of any kind need be had concerning each such Revised Contention's relation or applicability to the Byron Units;

RESPONSE:

(a) (ii) Not applicable. See response to 17(a)(i) above.

Interrogatory No. 17(a)(iii)

(iii) If any part of your answer to (i) or (ii) above relating to any Revised Contention is based in whole or in part upon the position that the subject matter of a Revised Contention is inapplicable (or unrelated) because (1) the subject matter has been considered at the construction phase hearing of the Byron Units; (2) the subject matter is barred from consideration at the operating hearings herein by an NRC regulation, rule, criterion, policy or convention; or (3) a Revised Contention has not specifically set forth a sufficient nexus (within the meaning of the River Bend Decision, ALAB-444, 6 N.R.C. 760 [1977]) regarding the Byron Units, then with respect to each such anwer regarding each such Revised Contention, please also state in specific detail, giving reasons for your position:

(a) Regarding (iii)(1) above, why it is your position that no facts or events have occured subsequent to the issuance of the construction permits herein which present a sufficient ground for re-examining the subject matter of the Revised Contention at the operating stage herein;

(b) Regarding (iii)(2) above, what NRC regulation, rule, criterion, policy or convention you believe bars consideration of the subject matter of the Revised Contention, and why you contend that there is no reason for waiving the applicability of any such regulation, rule, policy, criterion or convention to this proceeding; and

(c) Regarding (iii)(3) above, what fact, opinion, or other analysis of which you are aware (specifically and in detail explaining such fact, opinion, or other analysis) which can form the basis for a sufficient nexus to the Byron Units; in connection with your answer to this sub-part, if you state you are unaware of any facts, opinions, or analyses which can from such nexus, please also state in detail whether (and, if so, why) you believe it is impossible, as a matter of scientific or environmental application, for any nexus to be supplied whatsoever.

RESPONSE:

(a)(iii) Not applicable. See response to 17(a)(i) above.

Interrogatory No. 18

(a) To the extent not done in connection with each Interrogatory above, identify with particularity (including dates, addressor, addressee and subject matter) each document and communication which you either:

(i) have consulted or in any way reviewed in connection with any of your answers to these interrogatories; and/or

RESPONSE:

(a) (i) All documents considered in response to these interrogatories are identified in the response to specific interrogatories.

(ii) believe should be considered or reviewed in connection with any such answer,

in both cases specifying also in detail which document and communication relates, and in what manner it relates, to each of your Interrogatory answers.

RESPONSE:

(a) (ii) Documents which should be considered or reviewed in connection with the above interrogatories are named in the responses to the specific interrogatories.

Interrogatory No. 19

(a) Identify all persons who prepared or assisted in the preparation of any of the answers or parts of the answers to any of the above Interrogatories, specifying for each person which answer(s) he or she prepared or assisted in preparing.

RESPONSE:

(a) The following is a list of people who prepared answers to interrogatories. For those interrogatory subsections that request the identification of documents relied upon in answering that interrogatory, all people who prepared any subsection identified the documents, if any, that they relied upon.

Interrogatory	Preparer(s) of Answer
1(a),(b),(c)&(d)	W.J. Shewski: Commonwealth Edison M.A. Stanish: Commonwealth Edison K.J. Hansing: Commonwealth Edison P.T. Myrda: Commonwealth Edison
2(a)	K.A. Ainger: Commonwealth Edison
2(b)-(h)	G.T. Klopp: Commonwealth Edison
3(a)&(b)	J.C. Golden: Commonwealth Edison G.T. Klopp
3(c)	J.C. Golden

Interrogatory	Preparer(s) of Answer
3(d)	J.C. Golden G.T. Klopp
4(a)	J.C. Blomgren: Commonwealth Edison
4(b)	K.A. Ainger
4(c)	S.P. Barret: Commonwealth Edison T.P. Joyce: Commonwealth Edison D.G. Goldsmith: Commonwealth Edison
4(d)	Edward M. Burns: Westinghouse Electric Corporation
4(e)	J.R. Van Laere: Commonwealth Edison R.C. Ward: Commonwealth Edison L.A. Sues: Commonwealth Edison
4(f)	J.D. Deress: Commonwealth Edison
4(g)	G.T. Klopp
5(a),(b),&(c)	G.T. Klopp
5(d)	G.T. Klopp T.R. Tramm: Commonwealth Edison
5(e)&(f)	G.T. Klopp
6(a)&(b)	J. Regan: Sargent & Lundy
6(c)-(g)	J. Regan K. Green: Sargent & Lundy K.A. Ainger
7(a)-(e)	J. Holish: Sargent & Lundy
7(f)&(g)	G.T. Klopp
7(h)-(k)	A.K. Yonk: Sargent & Lundy
8(a)&(b)	G.P. Lahti: Sargent & Lundy
8(c)	J.I. Fabrikant: Professor of Radiology, University of California Berkeley
8(d)	G.P. Lahti J.R. Van Laere

Interrogatory	Preparer(s) of Answer
8(e)	K. Weaver: Commonwealth Edison J.R. Van Laere
8(f)	G.P. Lahti
8(g)	G.P. Lahti R.C. Ward L.A. Sues
8(h)	G.P. Lahti
9(a)	J. Regan
9(b)	G.T. Klopp K.A. Ainger K. Green
9(c)&(d)	G.T. Klopp
9(e)&(f)	J. Regan K. Green
10(a)&(b)	G.T. Klopp
11(a)-(e)	G.T. Klopp
12(a)-(f)	K. Green J. Regan
13(a)-(g)	J.C. Golden
13(h)&(i)	J.R. Van Laere
14(a)	G.T. Klopp
14(b)	C.S. Kuntz: Dames & Moore
14(c)	G.V. Komanduri: Sargent & Lundy
15(a)	J.I. Fabrikant
15(b)	J.C. Golden W.B. Brenner: Commonwealth Edison
15(d),(e)&(f)	J.C. Golden
15(g),(h)&(i)	J.R. Van Laere
16(a)	J.I. Fabrikant

Interrogatory	Preparer(s) of Answer
16(b)	E.M. Burns
16(c)	J.R. Van Laere E. Carnol: Commonwealth Edison B. Cooper: Commonwealth Edison T.K. Higgins: Commonwealth Edison
16(d)	J.R. Van Laere
16(e)	K. Weaver D. Kozin: Westinghouse Electric Corporation
16(f)	J.R. Van Laere
16(g)	K. Weaver

Interrogatory No. 19

(b) For each of the League's Revised Contentions listed in Interrogatory 17(a), state the following:

(i) the identity of each person expected to be called as a witness at the hearing or otherwise to submit testimony or Affidavit(s) concerning that Contention;

(ii) the substance of the witness's testimony of Affidavit(s); and

(iii) the witness's professional or other qualifications to testify or give Affidavit(s) on the subject matter on which the witness will testify or give Affidavit(s).

RESPONSE:

(b) The following is a preliminary list of witnesses which Edison has tentatively identified. Of course, as the discovery process continues, Edison's list of witnesses may change. The statement of the witnesses professional or other qualifications to testify are being developed and will be provided to the League upon their completion.

Contention 1a - If necessary, Edison intends to call Walter Shewski, Edison's Manager of Quality Assurance, and Michael Stanish, Edison's Byron Quality Assurance Supervisor, as witnesses. These witnesses will present testimony addressing Edison's corporate Quality Assurance Program and the manner that it has been and will be implemented at Byron, to demonstrate that the program complies with the requirements of 10 CFR Part 50, Appendix B.

Contentions 8 and 62 - Edison has retained a consultant to draft testimony on the issues raised in these contentions. The consultant's name is Saul Levine of the NUS Corporation. Dr. Levine is an expert in the area of probabilistic risk assessments and core melt for Class 9 accidents. His testimony will support the adequacy of the risk assessment for Byron set forth in the NRC Staff's Final Environmental Statement. Specifically, he will demonstrate that WASH-1400 is a suitable source and baseline document for the development of the Staff's risk assessment for Byron. Moreover, he will provide his opinion with respect to the quality of the Staff's risk assessment of core melt or Class 9 accidents for Byron.

Contentions 19 and 108 - Edison intends to call Dr. John Golden, Edison's Supervisor of Health Physics and Emergency Planning, to address these contentions.

Dr. Golden will describe the emergency plans which the Company has established for the Byron Station. Dr. Golden will also describe emergency plans which the Company has established for its other nuclear stations located in Illinois and describe the results of drills conducted to demonstrate the adequacy of these plans. Dr. Golden will provide his opinion with respect to the adequacy of the Byron emergency plan.

Edison may also call E. Erie Jones, Illinois Emergency Services and Disaster Agency, Director, or his designee to describe the State of Illinois emergency plans, and the site-specific State and local plans developed for the Byron Station.

Contention 22 - This contention addresses the issue of steam generator tube integrity. The witnesses Edison presently intends to call are Daniel Malinowski, Edward White, Laurence Conway, John Wootten, John Blomgren, and Rudolpho Paillaman. Mr. Malinowski's testimony will discuss the various corrosion and wear phenomena and, in general terms, what actions are being taken to provide or mitigate those phenomena. This witness, an expert on eddy current testing, will provide testimony on the nature and reliability of such testing. Mr. White is an expert on the subject of steam generator tube plugging criteria. He will provide testimony on the subject. Mr. Conway is a mechanical engineer involved in steam generator design.

He will provide testimony on that subject with particular emphasis on the D-4 and D-5 designs from the standpoint of improvements to eliminate corrosion and wear phenomena. Mr. Wootten is an expert on the subject of AVT water chemistry and he will provide testimony in that area. All of the foregoing are from Westinghouse Electric Corporation. Mr. Blomgren, a Commonwealth Edison employee, is an expert on the operation of steam generators and water chemistry matters. He will provide testimony on Edison's water chemistry program for Byron. In addition, he will address actions being taken in the area of eddy current testing and procedures to limit the effects of corrosion and wear phenomena. Mr. Paillaman, of EBSCO Company, will provide testimony on the pre-service inspection conducted by EBSCO with respect to the steam generator tubes in the steam generators for the Byron Station. In addition, Edison may call Mr. Douglas Fletcher of Westinghouse Electric Corporation. Mr. Fletcher would present an over-view of the manner in which Westinghouse has addressed concerns relating to steam generators.

Contentions 28 and 63 - These contentions raise concerns regarding the possibility of adverse systems interactions at Byron. Edison has tentatively decided to use four witnesses. They are Joe LaVallee, a private consultant, William Kortier, of Westinghouse Electric Corporation, Ken Greene and Kent Nowatny, both of Sargent and Lundy. Mr. LaVallee is an expert on the Byron design. Mr. Kortier is an expert with respect to the design of the Westinghouse NSSS system being provided at Byron. Messrs. Greene and Nowatny are also experts with respect to the design of Byron. Their testimony will support the NRC Staff's conclusion that the operating license for Byron can issue despite the existence of an unresolved safety question concerning systems interaction. In addition, their testimony will demonstrate how systems interaction have been taken into account during the course of the design of Byron. Mr. Kortier will direct himself to the Westinghouse NSSS system; Messrs. LaVallee, Green and Nowatny will address balance of plant issues. Edison may also use Jim Westermeier as a witness for the purpose of providing a Commonwealth Edison over-view on this issue. Mr. Westermeier is the project engineer for the Byron/Braidwood projects, and is employed by Commonwealth Edison Company.

Contentions 32, 61 and 77 - These contentions raise environmental qualification concerns. Edison has presently identified four individuals it will likely call as witnesses. They are Ken Greene, John Regan, Doug Paquette and Eric Tsai. Mr. Greene, an engineer with Sargent and Lundy, will provide testimony on the selection of environments to which equipment installed at Byron will be qualified. Mr. Regan, also an engineer with Sargent and Lundy, was principally responsible for developing the "Byron/Braidwood Stations Equipment Environmental Qualification Report". He will describe the equipment qualification methodology used to qualify equipment at Byron. Mr. Paquette, an employee of Commonwealth Edison Company, is responsible for developing a program for assuring that equipment which is not qualified for the full 40 year life of the plant is replaced when required. He will describe the program and how it will be implemented at Byron. Finally, Mr. Tsai, an independent consultant, is responsible for assuring that the equipment provided by Westinghouse Electric Corporation is qualified to the Byron specific qualification requirements. These witnesses will demonstrate that the environmental qualification program for Byron complies with NRC Regulatory requirements and is adequate to protect the public health and safety.

Contention 34 - This contention concerns the effects of overpressure transients on reactor systems including the pressure vessel. Edison may call Ed Burns, Greta Harkness, and a yet to be named metallurgical expert from Westinghouse. Ms. Harkness and Mr. Burns may provide testimony which demonstrates that Byron has an adequate overpressure protection system and that the Byron pressure vessel is very likely to be impervious to overpressure transients because of the high quality materials used in its fabrication.

Contention 39 - This contention pertains to liquid pathway accidents. Edison is currently investigating the possibility of using Saul Levine and Robert Henry as witnesses. Mr. Levine would address matters concerning the probability of accidents which could conceivably lead to release of substantial amounts of radioactivity into the groundwater. Mr. Henry, of Fauske Associates, would address matters concerning the adequacy of the Byron design to withstand such releases. These matters are currently under investigation.

Contention 41 - This contention relates to the possibility of ice build-up and its effects on the ultimate heat sink at Byron. Edison presently intends to call Ken Greene and Richard Netzel, both employees of Sargent and Lundy. Mr. Greene will address the design of the ultimate heat sink at Byron. Mr. Netzel will address the availability of adequate cooling in the event of extreme cold weather conditions.

Contention 42 and 112 - These contentions pertain to occupational exposure matters. Edison presently intends to call Gerald Lahti, Robert Pavlick and James Van Laere as witnesses. Mr. Pavlick, a Commonwealth Edison employee, will describe the Edison corporate ALARA program. Mr. Van Laere, also with Commonwealth Edison Company, will describe the manner in which the corporate ALARA program is implemented at Byron, and the specific measures which will be taken to maintain worker exposures ALARA. Mr. Lahti, of Sargent and Lundy, will describe the design features incorporated at Byron to minimize occupational exposures. If necessary, Edison intends to call Dr. Jacob I. Fabrikant to address health effects of exposures to low levels of radiation matters.

Contentions 47 and 71 - These contentions pertain to seismology and seismic design matters. The Company presently intends to call Anand K. Singh, Alan K. Yonk, Laurence Holish, and Charles Kuntz. Mr. Kuntz is a geologist, employed by Danes and Moore. Mr. Yonk, is also a geologist, who is employed by Sargent and Lundy and Mr. Holish is a geotechical engineer, also employed by Sargent and Lundy. These witnesses will discuss the geology and seismological aspects of the Byron site. Dr. Singh, an expert with respect to seismic design matters, will describe the seismic design basis for the Byron plant.

Contentions 53 and 54 - Contention 53 concerns the issue of whether the pressurizer heaters at Byron should be safety-grade. Contention 54 concerns the safety classification of the PORV's. Edison has not yet identified witnesses it will call with respect to these contentions. The names of witnesses will be provided at a later date.

Contention 109 - This contention pertains to Rock River hydrology. To the extent issues raised by this contention are not addressed in conjunction with the testimony regarding liquid pathway accidents, Edison has identified Lawrence Holish and Saul Levine as potential witnesses. Mr. Levine will address matters related to the probability of occurrence of accidents which could effect the release of radioactivity into the hydrosphere. Mr. Holish will address the hydro-olgical aspects of any such radioactive releases.

Contention 111 - This contention relates to a monitoring of radioactive discharge. The Company presently intends to call Dr. Golden as its expert with respect to this issue. Dr. Golden will describe the monitoring provisions which the Company intends to implement at Byron. To the extent necessary, the Company may also call Dr. Jacob Fabrikant to discuss matters pertaining to health effects of radiation.

Interrogatory No. 20

(a) Identify all persons (and their two closest assistants) whose advice was sought in the preparation of any of the answers or parts of the answers to any of the above Interrogatories, specifying for each person the answer(s) or portions of answers on which their advice was sought.

RESPONSE:

(a) All people who participated in the preparation of the interrogatories are listed in the response to Interrogatory 19(a) above. Any facts or opinions presented in the answers to interrogatories were developed by those individuals. Thus, identification of the two closest assistants of each of these individuals is neither relevant nor will lead to the discovery of relevant information. Accordingly, Edison objects to this aspect of the interrogatory.

Interrogatory No. 20

(b) For each of the League's Revised Contentions listed in Interrogatory 17(a) above, state the following:

(i) the identity of each person (and their two closest assistants) whose advice is expected to be sought regarding the submission of hearing testimony or Affidavits(s) concerning that Contention;

(ii) the substance of both the testimony and Affidavit(s) on which the advice will be sought and the substance of that advice; and

(iii) each person's professional or other qualifications to render advice on the subject matter of the testimony and/or Affidavit(s) on which his advice will be given.

RESPONSE:

(b) The witnesses Edison currently expects to call are listed in the response to Interrogatory 19(b) above. Any facts or opinions presented in their testimony will be developed by those individuals. Thus, identification of the two closest assistants of each of these individuals is neither relevant nor will lead to the discovery of relevant information. Accordingly, Edison objects to this aspect of the interrogatory.

Commonwealth Edison Company

By: [signature]
One of its Attorneys

Michael I. Miller
Alan P. Bielawski
M. Gwen Herrin
Isham, Lincoln & Beale
Three First National Plaza
Suite 5200
Chicago, IL 60603
(312) 558-7500

DATED: November 17, 1982

CERTIFICATE OF SERVICE

The undersigned, one of the attorneys for Commonwealth Edison Company, certifies that on this date he filed two copies (plus the original) of the attached pleading with the Secretary of the Nuclear Regulatory Commission and served a copy of the same on each of the persons at the addresses shown on the attached service list in the manner indicated.

Date: November 17, 1982

Alan P. Bielawski

SERVICE LIST

COMMONWEALTH EDISON COMPANY -- Byron Station
Docket Nos. 50-454 and 50-455

**Mr. Ivan W. Smith
Administrative Judge and Chairman
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

**Dr. Richard F. Cole
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

***Myron M. Cherry, Esq.
Cherry & Flynn
Three First National Plaza
Suite 3700
Chicago, Illinois 60602

*Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

*Chief Hearing Counsel
Office of the Executive
Legal Director
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

*Dr. A Dixon Callihan
Union Carbide Corporation
P.O. Box Y
Oak Ridge, Tennessee 37830

**Mr. Steven C. Goldberg
Ms. Mitzi A. Young
Office of the Executive Legal
Director
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

*Atomic Safety and Licensing
Appeal Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

*Secretary
Attn: Chief, Docketing and
Service Section
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

*Ms. Betty Johnson
1907 Stratford Lane
Rockford, Illinois 61107

**Ms. Diane Chavez
SAFE
326 North Avon Street
Rockford, Illinois 61103

*Dr. Bruce von Zellen
Department of Biological Sciences
Northern Illinois University
DeKalb, Illinois 60115

*Joseph Gallo, Esq.
Isham, Lincoln & Beale
Suite 840
1120 Connecticut Ave., N.W.
Washington, D.C. 20036

*Douglass W. Cassel, Jr.
Jane Whicher
BPI
Suite 1300
109 N. Dearborn
Chicago, IL 60602

* Via U.S. Mail

** Via Express Mail

*** Via Messenger