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

Docket Nos. 50-373
and 50-374

JUN 21 1979

Mr. Byron Lee, Jr.
Vice President
Commonwealth Edison Company
P. O. Box 767
Chicago, Illinois 60690

Dear Mr. Lee:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - LA SALLE COUNTY STATION, UNITS 1 & 2

As a result of our site visit on fire protection on April 2 to 5, 1979, we find that we need additional information to complete our review of your fire protection systems. Enclosure 1 contains our request for additional information which supplements our February 26, 1979 request.

In addition, we are enclosing Enclosure 2 which requests additional information in the areas of instrument and control systems, materials engineering and radiological assessment. Our concerns in each of these areas were either discussed with or made known to your personnel.

Please inform us after receipt of this letter of the date you can supply the requested information so that we may factor that date into our review schedule.

Please contact us if you desire any discussion or clarification of the information requested.

Sincerely,

Olan D. Parr, Chief
Light Water Reactors Branch No. 3
Division of Project Management

Enclosures: As Stated

cc w/enclosures: See next page

911041

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ENCLOSURE 1

010.0 AUXILIARY SYSTEMS BRANCH

Note: Questions which reference a specific fire zone in Unit 1 apply equally to the counterpart fire zone in Unit 2.

010.59 (RSP) In your comparison to Appendix A to BTP 9.5-1, you stated that LSCS is in compliance with position E.3.(b). However, at our site visit we were informed that only deluge control valves are electrically supervised, and other valves in the fire protection system are sealed with wire seals. Wire seals are not acceptable. State your compliance with section E.3.(b) by providing electrical supervision with alarm and annunciation in the control room for all valves in the fire protection system,

010.60 (RSP) In your comparison to Appendix A to BTP 9.5-1, you state that LSCS is not in compliance with position E.3.(d) regarding standpipe and hose located inside containment. State your compliance with section E.3.(d) by providing standpipe and hose stations outside the containment access with sufficient hose so that all areas of containment can be reached.

010.61 Provide diagrams which indicate the routing of all flammable and combustible liquid or gas lines in all plant areas containing or exposing safe shutdown systems, including those areas of the turbine building adjacent to, or integrated into the auxiliary building.

010.62 (RSP) On page 9.5-4 of the FSAR, you state that "suspended ceilings are of negligible combustibility." State your compliance with Section D.1.(f) of BTP 9.5-1, Appendix A, by providing sufficient data to demonstrate that such ceilings are of noncombustible construction as defined in Section B.4 of Regulatory Guide 1.120, or replace the ceilings with materials which will comply.

010.63 Fire Area 1 Refueling Floor

On Page H.3-1 of the FSAR, you indicate that the walls and roof of this zone are metal on unprotected steel supports, and that two open stairwells, as well as other unprotected floor openings, exist between this zone and lower elevations of the reactor building. Analyze the effect on safe plant shutdown or on radiation release if a fire were to cause the collapse of the roof and/or walls of this zone.

010.64 Fire Area 2 Reactor Building

(RSP) (1) Figures 9.5-1, Sheet 5, and 9.5-2, Sheet 4 indicate for Fire Zone 2B1 that a single failure in the fire suppression system could cause the loss of the deluge system for the standby gas treatment system charcoal filter and the hose stations at columns 11A and 15C, leaving only 50 ft. of hose at column 8.9BC to combat a fire in the filter area. In addition, Figure 9.5-1a, Sheet 1, indicates the presence of redundant trains of safe shutdown cables in this area near the charcoal filters. The deluge system should be converted to an automatic system, and that the length of hose at the column 8.9BC hose station be increased to 100 ft. to assure that fire protection water will be available to combat the fires in this area.

(RSP) (2) The door from Fire Zone 2G to the equipment access air lock which leads to the outside is a large, non-rated, motor-driven swing door. As a minimum, a 3 hr. rated fire door should be provided for this opening and an automatic sprinkler system be provided in this area of the reactor building around the railroad tracks to protect against a fire in materials which may be present in this area. The fire protection should

also extend to areas in this fire zone which contain redundant divisions of safe shutdown equipment and/or circuits. See our position stated in Questions 010.54(4) and 010.56(7) concerning fire protection for redundant trains in close proximity to each other.

(RSP) (3) Figure 9.5-1, Sheet 22, indicates that a 3 hr. rated fire door separates the off-gas building from the reactor building, Fire Zone 2G. However, on our site visit there were no labels on this door. A 3 hr. rated fire door should be provided to close this opening.

(RSP) (4) Figure 9.5-1, Sheet 29 indicates for Fire Zone 2H1, that the isolation dampers from this zone to the steam tunnel are 3 hr. fire rated dampers. However, on our site visit we were informed that these dampers are not fire-rated. Three hr. rated fire door/dampers should be installed in these vent openings to provide a complete 3 hr. fire barrier between the reactor building and the steam tunnel.

010.65 <u>Fire Area 4 Auxiliary Building</u>

(1) You indicate in the FSAR on Page H.3-57 that the roof over Fire Zone 4A, auxiliary building upper ventilation equipment floor, is mostly metal decking on 1 hr. protected structural steel. Analyze the effect on safe plant shutdown if a fire were to cause the collapse of the roof or the 1 hr. rated walls of this zone. Consider that such a structural collapse may affect the structural integrity of other floors or buildings, adjacent to the fire zone.

(2) As in Question 010.65(1) above, analyze the effect on safe plant shutdown if a fire were to cause structural collapse of the ceiling or walls of Fire Zone 4B, lower ventilation equipment floor.

(3) Your fire hazard analysis in the FSAR, Page H.3-64, states that the floor of Fire Zone 4C2, auxiliary building main floor, is a 2 hr. rated construction. Figure 9.5-1 Sheet 13 indicates it is only 1 hr. rated. For this area and any others with similar construction, analyze the effect on safe plant shutdown of a collapse of this floor slab. Consider, at least for this area, that the floor supports for the control room may be tied in with floor supports of this area.

(RSP) (4) During our site visit we observed a flammable liquid storage cabinet in the film room of Fire Zone 4C4, computer room area. The flammable liquid storage cabinet should be removed from this area as stated in Section D.2a of Appendix A to BTP 9.5-1, and the storage of flammable liquids should be limited to areas located in non-safety related buildings.

(5) During out site visit we were informed that the Halon suppression system
(RSP) will not be installed in Fire Zone 4E1, auxiliary equipment room. As a minimum an automatic suppression system should be installed in this area. Also the positions stated in Questions 010.54(4) and 010.56(7) should be implemented.

(6) On page 9.5-12 of the FSAR you state that the manual deluge systems for the auxiliary electric equipment room, Fire Zone 4E2, supply air filters are operated from the same room. Indicate how the deluge systems will be actuated assuming a fire in the filters which fills the auxiliary electric equipment room with dense smoke.

(RSP) (7) During our site visit we observed that exhaust fans for the battery rooms are being installed to exhaust air from the floor of the rooms. The exhaust air systems should be modified to remove air at the ceiling level in each of the battery rooms to prevent pocketing of hydrogen gas at the ceiling.

(RSP) (8) The battery rooms should be provided with automatic suppression systems unless they are separated from other areas of the plant by 3 hr. fire rated barriers.

(RSP) (9) Fire Zone 4F1, Page H.3-86. During our site visit we observed an enclosed bus duct which passed vertically through Fire Zone 4F1, Division 1 essential switchgear room at the south wall. We were advised that the bus duct penetrations of the floor and ceiling slabs will be sealed around the outside of the duct <u>only</u> - that there would not be a seal inside the cable duct. This is not acceptable. Indicate your compliance with Section D.1(j) of Appendix A to BTP 9.5-1 which requires that all penetrations of rated fire barriers be sealed to provide a fire resistance equivalent to that of the fire barrier. This includes seals in fully enclosed electrical raceways where the raceway itself is not enclosed in a 3 hr. rated fire barrier.

(RSP) (10) Fire Zone 4F3, Page H.3-89. During our site visit we were informed that the Halon suppression system will not be installed to protect the cables above the suspended ceiling in Fire Zone 4F3 auxiliary building ground floor. Indicate your compliance with BTP 9.5-1 Appendix A Section D.1(f) and provide adequate fire detection and suppression systems for the cable in the concealed space. In addition, the area below the ceilings should be provided with an automatic fire detection system.

010.66 Fire Area 5 Turbine Building

(1) Fire Zone 5A4, cable area elevation 749 feet, contains ESF Division 2 cables from both units. Describe the procedure used to shut down both units with available onsite power if a fire were to cause loss of these Division 2 cables from both units. Consider that in case of loss of offsite power, there is only one Division 1 diesel generator to handle the necessary loads of both units. Show that this diesel has adequate capacity for shutting down both units.

(RSP) (2) Because of the relatively high concentration of cables in Fire Zones 5A4 and 5B13, balance-of-plant cable area including some Division 2 cables, it is our position that an automatic water suppression system be installed in these areas in accordance with Section D.3.(c) of BTP 9.5-1 Appendix A.

(RSP) (3) The wall which separates Fire Zone 5B9, motor driven reactor feed pump room , from Fire Zone 8A1, HPCS diesel ventilation equipment room, is not a rated fire barrier. This wall should be upgraded to provide a 3 hr. rated fire barrier between these two zones to prevent a fire in Zone 5B9 from spreading into Zone 8A1.

(RSP) (4) The 125-V battery and cables for ESF Division 1 of Unit 2 are located in Fire Zone 5C11, turbine building ground floor. The 2 hr. fire rated walls which separate the battery room from the remainder of this zone, as well as the floor and ceiling of the battery room, should be upgraded to provide a 3 hr. fire barrier to separate the battery room from Fire Zone 5C11, and that all ESF cable associated with these batteries be separated from the turbine building by 3 hr. fire rated barriers.

010.67 Fire Area 7 Diesel Generator Building

(RSP) (1) Smoke or other products of combustion entering the diesel generator room air intakes could render the diesel generators inoperable. Such a scenario which could render all diesel generators for one unit inoperable is possible given fires in any of the outside transformers adjacent to the diesel generator building, a fire in any of the diesel fuel oil tank rooms, day tank rooms, or diesel generator rooms, all of which vent to the roof of the diesel generator building. This is also true for a fire in the bus duct cables immediately outside the air intakes for the diesel generators. This situation is not acceptable. Modify your design such that a fire will not render all the diesel generators inoperable or describe how you will safely shut down either one, or both units, if such a fire occurs concurrent with the loss of offsite power.

(RSP) (2) During our site visit we observed that the doors to and between each of the diesel generator rooms are not provided with curbs to contain a fuel oil spill. All entrances to the diesel generator rooms, including the doors between the rooms, should be provided with curbs to contain a fuel oil spill.

(3) During our site visit we observed that the discharge nozzles for the CO_2 systems in the diesel generator rooms were limited to a single line of nozzles directly over the diesel generators. Indicate the design parameters for this total flooding system and provide the results of tests which verify that the design concentration can be attained using this single line of nozzles throughout the rooms for the required soak time.

(4) Describe the provisions utilized to prevent a rupture at any point in the fuel oil lines from draining the entire contents of one fuel oil tank.

(RSP) (5) Your design of the diesel generator building, with the fuel oil storage tanks located below the diesel generators, is not in compliance with Section F.10 of BTP 9.5-1 Appendix A. Provide a liquid tight, 3 hr. fire rated barrier, including all penetration seals, between the fuel oil storage tank rooms and the diesel generator rooms, and all other areas. In addition, the fuel oil storage tank area and any other area which is susceptible to a large diesel fuel oil spill (if conditions of (4) above are possible) should be provided with a backup automatic or manual fixed suppression system in addition to the primary automatic suppression system.

010.68 Verify that the safety-related cable trays in Fire Zones 10A1, 10B1 and 10C3, Off-Gas Building, do not contain circuits required for safe plant shutdown.

010.69 (RSP) **Class IE and associated circuit cables at the La Salle County Station are marked at each end and in junction boxes. Cables, located in cable trays, are not marked and the markings, located in junction boxes, lacked sufficient durability to be readily recognized as color coded markings. Visual verification that the cable installation is in conformance with separation criteria was not possible.**

In accordance with the specific identification criteria of IEEE Standard 384-1974 and Regulatory Guide 1.75 (Revision 1), we require that both Class 1E and associated cables be marked in a manner of sufficient durability and at a suffcient number of points to facilitate visual verification that the installation is in conformance with separation criteria. Cables must be marked either before or during installations. Since this has not been done at the La Salle station, we require a detailed review of cable identification and routing methods. In this regard, we require a positive physical check of the installed cable be performed using high-frequency tracing (or other method) and sample testing techniques. We will request that I&E review your positive physical check of the installed cable.

010.70 In accordance with Section 8.3.1.3.2 of the FSAR, exposed conduits are supposed to be marked by color codes at the beginning and the end of the run, on both sides of a wall through which the conduit passes, and at both sides of junction boxes.

Class 1E conduits at the La Salle County Station are not marked by color code in accordance with the FSAR. Correct this deficiency. We will request that I&E inspect this deficiency.

010.71 (RSP) In regard to separation between Class IE and non-Class IE cable trays and cables, the separation is less than 3 ft. between trays separated vertically, the separation is non-existent between cables rising from both Class IE and non-Class IE trays in cable spreading areas (cables are bundled). It is our concern that failures or faults in non-Class IE cables will degrade Class IE circuits below an acceptable level.

For the non-Class IE cables rising from trays with no separation from Class IE cables, we consider that they are associated circuits and should meet the guidelines for associated circuits of IEEE standard 384-1974 and Regulatory Guide 1.75.

For vertical separation of less than 3 ft. between Class IE and non-Class IE trays, the results of an analysis should be provided in accordance with section 5.1.1.2 of IEEE standard 384-1974, that demonstrates that failure or faults in non-Class IE circuits will not degrade Class IE circuits below an acceptable level or these non-Class IE circuits must be considered associated circuits and meet the applicable requirements of IEEE standard 384-1974 and Regulatory Guide 1.75 for associated circuits.

010.72 To assure that redundant safety related cable systems are separated from each other so that both are not subject to damage from a single fire hazard, the following information for each Class IE system required to bring the plant to safe cold shutdown should be provided:

(1) Provide a Table listing electrical equipment required or essential for safe shutdown.

(2) Define each equipments location by fire area.

(3) Define each equipments redundant counterpart with a description of its locations with respect to its redundant counterpart.

(4) Identify the essential cabling (instrumentation, control, and power) for each equipment.

(5) Describe the routing of each essential cable identified in item (4) (by fire area) from source to terminations.

(6) Identify each location where essential cables are located in the same fire area with their redundant counterpart.

(7) For each location identified in item (6), describe the effects on safe shutdown if both redundant cables are lost from an exposure fire.

010.73 Section H.3.4.3 (page H.3-63) of Appendix H to the FSAR indicates that for the design basis fire in the control room: (1) the auxiliary equipment room contains the remote shutdown panels, (2) all circuits in the remote shutdown panels are electrically isolated from the main control room, and (3) remote shutdown circuits are unaffected by loss of the control room circuits.

To assure that the electrical isolation between the control room and the remote shutdown systems is sufficient to preclude a design basis fire in the control room, in the auxiliary equipment room, or at remote shutdown control locations from reducing the safe cold shutdown capability below an acceptable level, provide the following information:

(a) Identify each circuit located on the hot shutdown panel required for shutdown with a description of how it is isolated from the control room circuitry.

(b) For each circuit identified in item (a), provide detailed electrical schematic drawings which clearly describe the electrical isolation between the hot shutdown panel and control room.

(c) Identify each circuit required for safe cold shutdown located in the control room but not on the safe shutdown panels located in the auxiliary equipment room.

(d) For each circuit identified in item (c), provide the results of an analysis that demonstrates that failure (open, short, or hot short) of these circuits due to a design basis fire in the control room will not affect their remote safe shutdown capability.

ENCLOSURE 2

030.0 INSTRUMENTATION AND CONTROL SYSTEMS BRANCH

031.263 (G.2.1.2) (G.3.3.3) (Q031.248) We do not understand your response to Part 2 of Question 031.248. Therefore, please clarify the following:

(1) Part 2 of Question 031.248 deals with a discrepancy between a stated 20 percent and 25 percent reactor pump speed trip. The response refers to FSAR Section G.2.1.2.6.2e which discusses a 25 percent flow valve position interlock.

(2) The response to Questions 031.248 Part 2 also references FSAR Section G.3.3.3.10.2.2 which states that the recirculation pumps can be transferred to high speed from either the low speed motor generator or a dead start if the speed is less than 20 percent. Figure G.A-3, which is also referenced in the response, deals with a 18 percent valve position interlock.

(3) It is our understanding that the actual design includes a valve position interlock which is set at 25 percent of stroke and a speed interlock which is set at 20 percent. Are these understandings correct?

031.264 (RSP) (F807E152TD) (Q031.35) Your response to Question 031.35 states that no nuclear steam supply shutoff system isolation valves are provided with the manual override feature. The response then states that the reactor coolant sample lines and valves in the sample lines for post accident containment atmosphere monitoring are provided with manual override of the isolation signal. These manual overrides were not found during our review of the nuclear steam supply shutoff system final design drawings.

311054

Existing manual overrides must be shown in the final design drawings. Further, you should describe how the manual override complies with IEEE 279-1971, Sections 4.11 through 4.14. Therefore, please amend the FSAR accordingly.

031.260 (RSP) (F807E157TD) (F807E166TD) (F7.3-13)

Discrepancies have been found in various final design drawings during our review. Examples of the problem areas are:

(1) The reference table of General Electric (GE) Figure 807E152TD (Nuclear Steam Supply Shutoff System, Sheet 1) shows contacts 1-2 of coil K30 as being a spare. On sheet 6 of the same figure, this contact pair is shown as being used in indicating light circuitry.

(2) FSAR Figure 7.3-13 (Nuclear Steam Supply Shutoff System, sheet 1) shows the drywell high pressure signal as isolating various valves (E12-F023, E12-F009, E12-F008, E12-F049, E12-F040A, E12-F040B, E12-F053A and E12-F053B) in the RHR System. However, the logic in GE Figure 807E152TD, sheet 6, shows valves E12-F009, E12-F023, E12-F008, E12-F053A and E12-F05313 as not being isolated by this signal.

(3) FSAR Figure 807E152TD, sheet 6 (FSAR Figure 807E152TD, sheet 8), the isolation logic shows the drywell high pressure and reactor low water level 2 isolation signals operate contacts of K1A, B, C, D and C71A - K4A, B, C, D respectively. However, on the same page the isolation logic for high drywell pressure shows that the high drywell pressure signal operates contacts of C71A-K4A, B, C, D while sheets 5 and 5A of the GE drawing (FSAR Sheets 6 and 7) show contacts of K1A, B, C, D as being controlled by reactor low water level at level 2. Also, GE Figure 807E166TD, sheets 10 and 13, shows drywell high pressure to be associated with contacts of C71A-K4A, B, C, D. (Hence the logic for K1 and K4 is reversed on more than one sheet.)

(4) FSAR Figure 7.3-11 refers to FSAR Figure 7.3-15, sheet 2, zone B-11 for the details of valve steam leakoff detection for valve E12-F009. Zone B-11 does not exist on the referenced IED (as determined by the response to Question 031.15).

Revise the final system design drawings to resolve all such discrepancies and submit the revised drawings in accordance with RG 1.70, Chapter 7.0.

031.266(RSP)
(Q031.137)
(Q031.240)

The responses to Parts 2 and 5 of Question 031.137 and Part 2 of Question 031.240 do not demonstrate that the devices which are used to protect the Class 1E nuclear instrumentation from the non-Class 1E control systems are suitably qualified. Therefore, the above cited responses are unacceptable. Provide amended responses to the cited questions which demonstrate that suitable (Class 1E) isolation is provided to protect the outputs from the nuclear instrumentation.

031.267(RSP)
(Q031.240)

The response to Question 031.240 Part 3 is incomplete and, therefore, unacceptable. Revise the response to include a description of the effect of Test 1 on the inputs and the bases for accepting such voltage transients in the nuclear instrumentation circuits which feed their signals to this type of isolator.

031.268
(Q031.137)
(Q031.240)

The meaning of your response to Question 031.137 Part 7 and the follow up Question 031.240 Part 4 is not clear. State that the isolation of the ESF status signals occurs within the ESF equipment via relay contacts and that the subject ESF status input cabinets contain no Class 1E wiring or provide a clear description and justification for any alternative design.

031.269
(Q031.256)

The response to Question 031.256 Part 3 is unclear. Describe the signal source for K62413 and K624C when the power supply which is common to K611, K612, K613, and K624A fails.

311057

031.270 (F9.4-1) (1E-0-4432AD) Clarify the discrepancy between FSAR Figure 9.4-1 Sheet 6 and FSAR Drawing 1E-0-4432AD with regard to the control of OVC16YA.

031.271 (1E0-4432AD) Justify your assertion that a failure of relay OREY-VC080XA, OXY-VC088X1 or OAE-VC090X1 does not constitute a single failure which will leave the control room unisolated when isolation is required.

120.0 MATERIALS ENGINEERING BRANCH

121.11 In Section 5.2.3.3.1.4, of the FSAR, "Operating LImits Based on Fracture Toughness," a factor of 2°F per ft-lb is used to convert longitudinal Charpy V-notch impact data obtained at the 30 ft-lb level to estimates of the data at the 50 ft-lb level. It is stated that these estimates plus a 30°F adjustment for specimen orientation are based on information tabulated in WRC Bulletin 217, "Properties of Heavy Section Nuclear Reactor Steels," and other fracture toughness tests. Explicitly state the procedures used to verify these factors, including a sample calculation and any data, other than that in WRC Bulletin 217, used as a basis for the estimates.

121.12 To demonstrate compliance with Section IV.A.3 of Appendix G, 10 CFR Part 50, provide the results from the C_V impact tests for materials for piping, pumps and valves.

121.13 It is stated in the FSAR that at 10°F the reactor vessel closure stud materials for Unit No. 1 have a minimum Charpy impact energy and lateral expansion of 43 ft-lb and 23 mils respectively. The reactor vessel stud materials in Unit No. 2 have a reported minimum Charpy impact energy of and a lateral expansion of 40 ft-lbs and 24 mils respectively. To demonstrate compliance with Section IV.A.4 of Appendix G, 10 CFR Part 50, provide the Charpy impact test results for bolting and other fastener materials in Unit Nos. 1 and 2.

121.14 Referring to Table 5.2-11 of the FSAR, the statement is made that for the nozzles, flange and shell regions near geometric discontinuities a RT_{NDT} of 40°F is used in lieu of a discontinuity analyses to demonstrate compliance with Section IV.A.2.b of Appendix G. Provide a calculation to demonstrate the adequacy of RT_{NDT} of 40°F in those regions and show how this value is used for the determination of the reactor vessel pressure temperature operation limits.

121.15 Section IV.B of Appendix G, 10 CFR Part 50, requires that the reactor vessel beltline material possess a minimum upper-shelf energy of 75 ft-lbs as determined from Charpy impact test data on unirradiated specimens conducted in accordance with paragraph NB-2322.2(a) of the ASME Code. To demonstrate compliance with Section IV.B, provide the upper-shelf Charpy impact test data for the reactor vessel beltline materials. If upper-shelf energies of less than 75 ft-lbs were obtained, analyses and data also must be submitted to demonstrate adequate margins of safety from deterioration by neutron irradiation.

121.16 Section II.B of Appendix H of 10 CFR Part 50 requires that materials from the reactor beltline region be monitored by a surveillance program complying with ASTM E 185-73, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels." The FSAR indicates that the specimens selected for the surveillance program are from representative materials from orientation and location not in conformance to ASTM E 185-73. Provide the following information for the specimens in the surveillance program:

(1) Identify the surveillance specimens taken from the base and weld materials in the beltline region of the reactor vessel, including the plate and weld identification number, specimen orientation, and the location from which the specimens were taken. Also identify the base and weld metal surveillance specimens that were not taken from the actual base and weld materials in the reactor vessel beltline region. The identification should include:

(a) Materials specification, heat number and material identification number

(b) Weld wire

(c) Weld flux

(d) Weld process

(e) Heat treatment

(f) C_V impact energy test results

(g) RT_{NDT}

(h) Copper content

(2) Provide technical justification for deviation from the requirement of Section II.C.1 of Appendix H that the test specimens in the surveillance program be taken from alongside the fracture toughness test specimens which were required by Section III.A of Appendix G of 10 CFR Part 50. The information should demonstrate that the test specimens are fully representative of the materials and processes used for the fabrication of the beltline region of the reactor vessel.

(3) In response to Question 121.3, the statement is made that based on experience at the General Electric Company the amount of shift measured on irradiated longitudinal test

specimens will be essentially the same as the shift in equivalent transverse specimens. Provide the Charpy impact test data to demonstrate that the RT_{NDT} shift and the decrease in the upper-shelf energy level due to neutron irradiation are equivalent regardless of the specimen orientation.

330.0 RADIOLOGICAL ASSESSMENT

331.23 (12.3) Describe permanent shielding provided to assure acceptable radiation levels in potentially occupied areas in the vicinity of the spent fuel transfer process. If very high radiation areas are projected, describe precautions taken to prevent inadvertent personnel access during fuel transfer.