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June 29, 1982

Mr. A. Schwencer, Chief
Licensing Branch #2
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: LaSalle County Station Units 1 and 2
Compliance with Regulatory Guide 1.97
NRC Docket Nos. 50-373 and 50-274

Reference (a): License NPF-11,. Condition 2.C.(18), dated April 17, 1982.

Dear Mr. Schwencer:

Reference (a) states, in part:

"By July 1, 1982, the licensee shall provide a plan for implementing modifications necessary to comply with Revision 2 of Regulatory Guide 1.97," "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," dated December 1980.

The purpose of this letter is to provide you with Commonwealth Edison Company's position on Regulatory Guide 1.97, for LaSalle County Station. This position is contained in the attachment to this letter. Commonwealth Edison Company believes it has completed the actions required by license NPF-11 Condition 2.C.(18) with this submittal.

To the best of my knowledge and belief the statements contained in this letter and in the attachment are true and correct. In some respects these statements are not based on my personal knowledge but upon information furnished by other Commonwealth Edison employees. Such information has been reviewed in accordance with Company practice and I believe it to be reliable.

If there are any further questions in this matter, please contact this office.

Very truly yours,

CW Schroeder 6/30/82

C. W. Schroeder
Nuclear Licensing Administrator

lm

cc: NRC Resident Inspector - LSCS

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ATTACHMENT

LaSalle County Station
License NPF-11 Condition 2.C.(18)
Compliance with Regulatory Guide 1.97

This Attachment presents the LaSalle County Station Units 1 and 2 position on Regulatory Guide 1.97 "Instrumentation For Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident". Some of the criteria of the Reg. Guide are discussed in the following paragraphs. A table is provided identifying Reg. Guide Table 1 variable and our position on that variable. Supporting information follows the table and a schedule for modifications to implement our position is discussed.

As explained in the text, the LaSalle as built equipment conforms with the intent of RG 1.97, Revision 2, which is to ensure that necessary and sufficient instrumentation exists at each nuclear power station for assessing equipment and plant conditions during and following an accident, as required by 10 CFR Part 50, Appendix A and General Design Criteria 13, 19, and 64. Implementation of RG 1.97 criteria is recommended except in those instances in which deviations from the letter of the guide are justified technically and when they can be implemented without disrupting the general intent of the Guide. Unless a particular paragraph or topic is discussed herein, our concurrance with the Revision 2 guidance is hereby acknowledged. LaSalle compliance as presented here, justifies the validity of several technical alternatives to the guidance of RG 1.97. Those variances are described herein and are considered workable alternatives.

In assessing RG 1.97, Edison has drawn upon information contained in several applicable documents, such as ANS 4.5,

NUREG/CR-2100, and the BWROG Emergency Procedures Guidelines, and on data derived from other analyses and studies. Edison believes that literal compliance with the provisions of the guide, because of their specific nature or their generic origin for PWR's, is not appropriate. Some RG 1.97 criteria call for excessive ranges or quality categories or both, others call for functions already available at LaSalle, and still others could adversely affect operator judgment under certain conditions. For example, detailed study sponsored by the BWR Owners Group shows that core thermocouples will provide conflicting information to reactor operators. Edison intends to follow the criteria used by the NRC for establishing Category 1, 2, and 3 instruments, although it should be noted that Category 2 instruments could vary widely between utilities, because of various plant-unique features.

A. General Statement

The following statement applies to the regulatory positions defined in RG 1.97, Revision 2 (the paragraph numbers cited correspond to those in RG 1.97).

1. Accident-Monitoring Instrumentation

Par. 1.3: Instruments used for accident monitoring to meet the provisions of RG 1.97 shall have the proper sensitivity, range, transient response, and accuracy to ensure that the control room operator is able to perform his role in bringing the plant to, and maintaining it in, a safe shutdown condition and in assessing actual or possible releases of radioactive material following an accident.

Accident-monitoring instruments that are required to be environmentally qualified will be qualified to the requirement of NUREG-0588. (and Memorandum and Order CLI-80-21) Seismic qualification of LaSalle equipment was completed to the IEEE 344-1975 Standards under the SQRT program.

Edison will comply with the quality assurance requirements, using its approved quality assurance program, as described in Topical Report CE-1 as revised.

Periodic checking, testing, calibrating, and calibration verification of accident-monitoring instrument channels (RG 1.118) shall be in accordance with the LaSalle Tech Specs.

Par. 1.4: Instruments designated as Categories 1 and 2 for variable types A, B, and C should be identified in such a manner as to optimize the human factors engineering and presentation of information to the control room operator. This position is taken to clarify the intent of RG 1.97, which specified that these instruments be easily discerned for use during accident conditions (see Issue 1, Sec. 5). LaSalle will evaluate methods to identify control room indicators during the Human Factors Engineering review of the control room. An implementation schedule will then be determined.

Par. 1.6: It is Edison's position that RG 1.97 Table 1 does not represent the minimum number of variables, ranges and categories needed for a BWR facility. LaSalle meets our alternative criteria which is bounded by RG 1.97. For that reason Table 1 is addressed herein.

Par. 2: Positions stated in Par. 1.3 and 1.6 above apply to the Type D & E variables.

B. LaSalle Type A Variables

Regulatory Guide 1.97, Revision 2, designates all Type A variables as plant-specific, thereby defining none in particular. The Guide defines Type A variables as

Those variables to be monitored that provide primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accident events.

Regulatory Guide 1.97 defines primary information as "information that is essential for the direct accomplishment of the specified safety functions." Variables associated with contingency actions that may be identified in written procedures are excluded from this definition of primary information.

The following is a list of Type A variables applicable to LaSalle.

Variable A1. Hydrogen Concentration Range: 0 - 10%

Operator action: If H_2 concentration in containment approaches the combustible limits, initiate combustible gas control systems.

Safety function: Prevent hydrogen buildup to combustible levels and thus preserve containment integrity.

Variable A2. RPV Pressure Range: 0 - 1250 psig

Operator action: (1) Depressurize RPV and maintain safe cooldown rate by any of several systems, such as main turbine bypass valves, HPCS, RCIC, and RWCU: (2) manually open one SRV to reduce pressure to below SRV setpoint if an SRV is cycling.

Safety function: (1) Core cooling; (2) maintain reactor coolant system integrity.

Variable A3. Suppression Pool Water Temperature Range: 30 - 230^{O}F

Operator action: (1) Operate available suppression pool cooling system when pool temperature exceeds normal operating limits; (2) scram reactor if temperature reaches limit for scram; (3) if suppression pool temperature cannot be maintained below the heat capacity temperature limit, maintain RPV pressure below the corresponding limit; and (4) close any stuck-open relief valve.

Safety function: Maintain containment integrity.

Variable A4. Suppression Pool water Level Range: -18 - 14'

Operator action: Maintain suppression pool water level within normal operating limits: If suppression pool water level cannot be maintained below the suppression pool load limit, maintain RPV pressure below corresponding limit.

Safety function: Maintain containment integrity.

Variable A5. Drywell Pressure Range: -5 - 100 psig

Operator action: Control primary containment pressure by any of several systems, such as containment pressure control systems, standby gas treatment (SBGT), suppression pool sprays, drywell sprays.

Safety function: Maintain containment integrity.

C. Plant Variables for Accident Monitoring

Edison's positions on the implementation of the variables listed in Table 1 of RG 1.97 and on the fulfillment of design criteria and assignment of qualification categories for the instrumentation proposed for their measurement are summarized in the tabulation that follows.

In brief, the measurement of the five groups of variables provides the following kinds of information to plant operators during and after an accident: (1) Type A--plant pressure, barrier and heat sink information, on the basis of which operators can take specified manual control actions; (2) Type B--information about reactor shutdown, inventory control and isolation status; (3) Type C--information about the breaching of barriers to fission product release; (4) Type D--information about the operation of individual safety systems; and (5) Type E--information about the magnitude of the release of radioactive materials.

The categories are also related (in RG 1.97) to "key variables." Key variables are defined differently for the different variable types. For Type B and Type C variables, the key variables are those variables that most directly indicate the accomplishment of a safety function; instrumentation for these key variables is designated Category 1. Key variables that are Type D variables are defined as those variables that most directly indicate the operation of an emergency safety system; instrumentation for these key variables is usually Category 2. And key variables that are Type E variables are defined as those variables that most directly indicate the release of radioactive material; instrumentation for these key variables is also usually Category 2.

It should be noted that the Type A variables listed below are being proposed for inclusion in RG 1.97. Table 1 of RG 1.97 designates all Type A variables as plant specific and thus defines none in particular.

The variables are listed here in the same sequence used in Table 1, RG 1.97; however, for convenience in cross-referencing entries and supporting data, the variables are designated by letter and number. For example, the sixth B-type variable listed in RG 1.97 is denoted here as variable B6. (A copy of Table 1 from RG 1.97 is provided as an attachment).

Edison's position is shown for each variable. In general, there are three kinds of responses or recommendations: (1) Reg. Guide criteria are met; (2) existing system is justified; or (3) LaSalle will not implement. One exception is neutron flux (B1) which will be modified to meet our position.

As necessary, the positions are elaborated or substantiated in the "issues" section that follows the table.

Reg. Guide 1.97 Variables and LaSalle Positions

Type A Variables	Position
A1. H_2 concentration	LaSalle meets Reg. Guide criteria for all Type A variables
A2. Reactor pressure	
A3. Suppression pool water temperature	
A4. Suppression pool water level	
A5. Drywell pressure	
Type B Variables	
B1. Neutron flux	Will implement Category 2 system. See Issue 2.
B2. Control rod position	Criteria met
B3. RCS soluble boron concentration (sample)	Criteria met
B4. Coolant level in reactor	Criteria met for range of -160 - 60", see Issue 3. BWR Owners Group to address detection of core cooling.
B5. BWR core thermocouples	Will not implement. Reactor level instrumentation adequate with EPG's.

Type B Variables (cont'd)	Position
B6. RCS pressure	Criteria met, see Issue 3.
B7. Drywell pressure	Criteria met
B8. Drywell sump level	Criteria for Category 3 met, see Issue 4.
B9. Primary Containment Pressure	Criteria met
B10. P.C. Isol. Valve Position	Criteria met, redundant indication is not required on redundant isolation valves. Exclude check valves.

Type C Variables

C1. Radioactivity concentration or radioactivity level in circulating primary coolant.	Will not implement, see Issue 5.
C2. Analysis of primary coolant (gamma spectrum)	Criteria met
C3. BWR core thermocouples	Will not implement, see B.5.
C4. RCS pressure	Criteria met
C5. Pri. containment area radiation	Criteria met
C6. Drywell drain sumps level	Criteria met for Category 3, see Issue 4.

Type C Variables (cont'd)	Position
C7. Suppression pool water level	Criteria met
C8. Drywell pressure	Criteria met
C9. RCS pressure	Criteria met
C10. Pri. containment pressure	Criteria met for 0-100 psig, one channel to 135 psig.
C11. Containment and drywell H_2 conc.	Criteria met
C12. Containment and drywell O_2 conc.	Criteria met
C13. Containment effluent radioactivity-noble gases	Criteria met
C14. Radiation exposure rate	Current instrumentation is adequate, see Issue 6
C15. Effluent radioactivity-noble gases	Criteria met

Type D Variables

D1. Main feedwater flow	Criteria met
D2. Condensate storage tank level	Criteria met

Type D Variables (cont'd)	Position
D3. Suppression spray flow	Current design is adequate, see Issue 7.
D4. Drywell pressure	Criteria met
D5. Suppression pool water level	Criteria met
D6. Suppression pool water temperature	Criteria met
D7. Drywell atmosphere temperature	Criteria met
D8. Drywell spray flow	Current design is adequate, see Issue 7.
D9. MSIV leakage control system pressure	Criteria met
D10. SRV position	Criteria met
D11. Isolation condenser system shell-side water level	N/A
D12. Isol. condenser system valve position	N/A
D13. RCIC flow	Criteria met, see Issue 9.
D14. HPCI flow	Criteria met for HPCS, see Issue 9.
D15. Core spray system flow	Criteria met, see Issue 9.
D16. LPCI system flow	Criteria met, see Issue 9.

Type D Variables (cont'd)	Position
D17. SLCS flow	Alternate criteria met for category 3, see Issue 9.
D18. SLCS storage tank level	Criteria met for category 3, see issue 10.
D19. RHR system flow	Criteria met
D20. RHR heat exchanger outlet temperature	Criteria met
D21. Cooling water temp. to ESF components	Criteria met for main system flow
D22. Cooling water flow to ESF components	Criteria met for main system flow
D23. High radioactivity liquid tank level	Criteria met
D24. Emergency ventilation damper position	Criteria met for dampers actuated under accident conditions whose failure could result in radioactive release.
D25. Status of standby power and other energy sources	Criteria met, on-site sources only.
Type E Variables	
E1. Primary containment area radiation-high range	Criteria met

Type E Variables	Position
E2. Reactor building or secondary containment area radiation	Existing instrumentation is justified, see Issue 11.
E3. Radiation exposure rate	Existing instrumentation is justified, see Issue 12.
E4. Noble gases and vent flow rate	Criteria met
E5. Particulates and halogens	Criteria met
E6. Radiation exposure meters	Deleted per NRC errata July 1981
E7. Airborne radiohalogens and particulates	Criteria met for portable units, on-site analysis.
E8. Plant environs radiation	Existing instrumentation justified, see Issue 14.
E9. Plant and environs radioactivity (MCA)	Criteria met
E10. Wind direction	Criteria met
E11. Wind speed	Criteria met
E12. Estimation of atmospheric stability	Criteria met
E13. Primary coolant and sump (grab sample)	Criteria met for primary coolant and containment sump, see Issue 13.
E14. Containment air sample	Criteria met

D. Issues

ISSUE 1. INSTRUMENT IDENTIFICATION

Issue Definition

Regulatory Guide 1.97 specifies, in par. 1.4.b, the following: "The instruments designated as Types A, B, and C and Categories 1 and 2 should be specifically identified on the control panels so that the operator can easily discern that they are intended for use under accident conditions."

Discussion

The objective of this regulatory position is the achievement of good human factors engineering in the presentation of information to the control room operator. This objective is primarily achieved by good indication-control relationship, training and experience. The proper specific indication of what indicators are intended for use under accident conditions is best evaluated in a task analysis to determine what indication is used and an evaluation of the main control panels as a whole. Improper indication could distract an operator and cause confusion.

Conclusion

Instruments designated as Categories 1 and 2 for monitoring variable types A, B, and C should be identified in such a manner as to optimize applicable human factors engineering and presentation of information to the control room operator. LaSalle will determine an acceptable method of identifying RG 1.97 parameters as part of the Human Factors Engineering review of control room design.

ISSUE 2. VARIABLE B1

B1: Neutron Flux

Issue Definition

The measurement of neutron flux is specified as the key variable in monitoring the status of reactivity. Neutron flux is classified as a Type B variable, Category 1. The specified range is 10^{-6} percent to 100 percent full power (SRM, APRM). The stated purpose is "Function detection; accomplishment of mitigation."

Discussion

The lower end of the specified range, 10^{-6} percent full power, is intended to allow detection of an approach to criticality by some undefined and noncontrollable mechanism after shutdown.

In attempting to analyze the performance of the neutron-flux monitoring systems, a scenario was postulated to obtain the required approach to criticality. Basically, it assumes an increase in reactivity from loss of boron in the reactor water after SLCS actuation.

The accident scenario incorporates the following factors:

1. The control rods fail (completely or partially) to insert, and the operator actuates the standby liquid control system (SLCS).

2. The SLCS shuts the reactor down.

3. A slow leak in the primary system results in an outgo of borated water and its replacement by water that contains no boron.

4. A range of leak rates up to 20 gpm was considered (see Table 1).

Calculations were made to evaluate the rise in neutron population as a function of different leak rates. The calculations were made for a shutdown neutron level of 5×10^{-8} percent of full power. The choice of 5×10^{-8} was based on measurements at two BWR plants. The shutdown level was assumed to have a negative reactivity of 10 dollars, an assumption that is representative of a shutdown with all rods inserted. The results of the calculations are presented in Table 1. The numbers in the table refer to the time in hours required to increase the flux by 1 decade. For example, with a leak of 5 gpm, it takes 100 hr to increase the power from 5×10^{-8} percent to 5×10^{-7} percent, and 10 hr to increase it from 5×10^{-7} percent to 5×10^{-6} percent.

The reactor is subcritical and the neutron level is given by

$$\text{Neutron level} = S \times M,$$

where S is the source strength and M is th[illegible]iplication, which is given by

$$M = 1/(1-k).$$

For $k = 0.9$, M is 10; for $k = 0.99$, M is 100 and so forth. For criticality, the denominator approaches 0, as k approaches 1.0. Thus, the above equation was used to calculate relative neutron flux levels for a subcritical reactor until the reactor was near critical; then the critical equation of power with excess reactivity was used. Reactor power is directly proportional to neutron level.

The increase in reactivity toward criticality can be turned around by actuating the SLCS. A second actuation of the SLCS would cause a decrease in reactivity because of the high concentration of boron in the injected SLCS fluid relative to that in the leaking fluid (nominally 400 ppm). The sensitivity of the detector must allow adequate time for the operator to act. For a scaling evaluation, ten minutes

was considered sufficient time for operator action for accident prevention and mitigation.

Table 1 shows that the detector sensitivity (i.e., lower range) requirement is a function of leak rate and therefore of reactivity-addition rate. On the basis of a 20-gpm leak rate, Table 1 shows that a detector that is on scale (i.e. about 10^{-5}) within 3 decades of the shutdown power (10^{-8}) would allow 0.18 hr (10.8 min) for operator action before reactor power increased another decade. A total of 0.36 hr (21.6 min) would be available for operator action from the time the indicator comes on scale to the time reactor power reaches 0.5 percent of full power. An alarm would be provided to warn the operator when the neutron flux reaches some plant-specific set-point.

The 20-gpm leak rate, which was assumed to continue for 27.75 hr, was used to define the sensitivity of the detector. It should be noted that the assumed leak rate, extended over the 27.75-hr period, would result in a loss of inventory so large that it could not in reality go undetected by the operator. Moreover, reactivity-addition caused by this gradual boron depletion is unlikely, since boron concentration is sampled and measured periodically. Again, the improbable 20-gpm leak rate was used only to obtain a mechanistic and conservative approach for selection of instrument sensitivity.

An absolute criterion for the lower range must include consideration of the neutron source level. The use of the neutron level 100 days after shutdown is conservative. There is high probability that conditions would be stable and controllable 2 days after the emergency shutdown, for the core-decay heat is at a low level and the boron monitoring system should be functioning by that time. The actual neutron level will vary with fuel design, fuel history, and shutdown control strength. Measurements of shutdown neutron flux (with all rods inserted) at two BWR reactors show readings of 30 to 80 counts/sec (1000 counts/sec corresponds to 10^{-8} of full power). Measurements on other BWR reactors and for different fuel histories would show some variation, but those

variations would be small compared with a criterion that is concerned with units of decades.

Neutron flux is the key variable for measuring reactivity control. The degree to which this variable is important to safety is another consideration. The large number of detectors (i.e., source-range monitors and intermediate-range monitors) that are driven into the core soon after shutdown makes it highly probable that one or more of the existing NMS detectors will be inserted. On the other hand, there is little probability that there would be, simultaneously, a need for this measurement (in terms of operator action to be taken) and an accident environment in which the NMS would be rendered inoperable. Further, the operator can always actuate the SLCS on loss of instrumentation.

Although some upgrading of the current NMS is appropriate to improve system reliability and its ability to survive a spectrum of accidents, a rigorous Category 1 requirement is not justified relative to the criterion of "importance to safety." A Category 2 classification of this measurement fully meets the intent of RG 1.97 for neutron flux indication.

Four options have been identified to meet the LaSalle neutron flux measurement criteria. All four options could provide indication over the range recommended. The principal features of the four alternatives are presented below.

<u>Option 1.</u> The first option provides for upgrading two or more of the source-range monitors (SRM's). The upgrading includes the connecting cable inside the drywell and the power source for the SRM drives. At least two SRM's would have dual roles of accident instrumentation and normal start-up; these two SRM's would be withdrawn a lesser distance from the core than the SRM in the current design. It is estimated that in its fully withdrawn position, the current SRM will detect about 10^{-3} or 10^{-5} percent of full power. This sensitivity can be increased by using an intermediate position that is less than the

present 2-2.5 ft from the core. An intermediate position that produces 10 percent depletion in 5 years was used as a guide to the maximum allowed burn-up of the sensor. This position below the core would give the SRM a detection capability of about 2×10^{-7} percent of full power. The SRM drives need not be upgraded, because the upgraded detector system would be adequate, even if the drive did not move the SRM detector. (An upgraded power source for the drives improves the probability of insertion.) The success of this alternative--which uses the four SRM's for normal start-up--depends on a design modification to accommodate the new cable (the key concern is the flexibility of the cable, for the detector moves about 10 ft; this movement is accommodated in the cable loop) and on the design of a limit switch or a detent mechanism to hold the drive tube in the required intermediate position.

Option 2. The second option is to replace two or more SRM systems with upgraded systems. The full SRM system, including the drives, would be upgraded. This approach would require input from a potential equipment supplier in order to estimate schedules, cost, and overall effect of the upgrading. Whereas the first alternative uses upgraded cables and power supply (which are commercially available), this approach would require additional engineering to achieve an upgraded drive system as well. A Category 1 drive system is a developmental item.

Option 3. In the third option, fixed in-core detectors are used. The system uses SRM-type detectors as stationary detectors positioned close enough (as discussed above) to the core to meet the lower range requirements. New cables are needed to meet the requirements of the accident environment. This system would provide dedicated "accident monitors" in two of the intermediate-range monitor (IRM) tubes or in two local-power range-monitor (LPRM) tubes. It may be feasible to put five detectors in the LPRM tube or, if space is limited, the bottom detector of the LPRM string could be replaced with the "accident" detector. With this approach the four movable SRM's would still continue to be available for normal functions.

Option 4. In the final option, out-of-core detectors, which are being qualified for use in pressurized water reactors (PWR's) may be used. Considerations of this ongoing PWR qualification program for Cateogry 1 instrumentation and the freedom from any effect on the current neutron monitoring system (NMS) make this option an attractive one. The key question is whether these out-of-core detectors can meet the lower range requirement, for the detectors are positioned outside the biological shield. Based on calculations of neutron flux made for a BWR at full power (see Fig. 1) and on current detector design practices, the out-of-core detector may be feasible. Other effects, such as attenuation by water that is at a lower temperature (than the full-power operating temperature) and by boron in the water, need to be considered.

Conclusion

A range from 5×10^{-5} percent of full power (within 3 decades of the neutron flux level 100 days after shutdown) to 100 percent of full power is recommended. An alarm is also recommended that would alert the operator of a rise in neutron flux. It is concluded that a Category 2 classification is responsive to the intent of RG 1.97, and that one of the four options can provide the needed neutron flux measurement.

TABLE 1. RELATIVE NEUTRON FLUX VERSUS TIME[a]

Percent of power	Leakage rate, gpm (ramp rate, ¢/min)[b]					
	1(0.03)		5(0.15)		20(0.60)	
	Σ	Δ	Σ	Δ	Σ	Δ
5×10^{-8}	-555	500	-111	100	-27.75	25
5×10^{-7}	-55	50	-11	10	-2.75	2.5
5×10^{-6}	-5	5	-1	1	-0.25	0.25
5×10^{-5}	0		0		0	
5×10^{-4}	0.8	0.8	0.36	0.36	0.18	0.18
5×10^{-3}	1.33	0.53	0.51	0.15	0.25	0.07
5×10^{-2}	1.59	0.26	0.62	0.11	0.31	0.06
5×10^{-1}	1.80	0.21	0.72	0.10	0.36	0.05
5×10^{0}	1.89	0.09	0.80	0.08	0.40	0.04

[a]Shutdown flux = 5×10^{-8} percent of power.

[b]Σ = total number of hours; Δ = hours for neutron flux to increase by one decade.

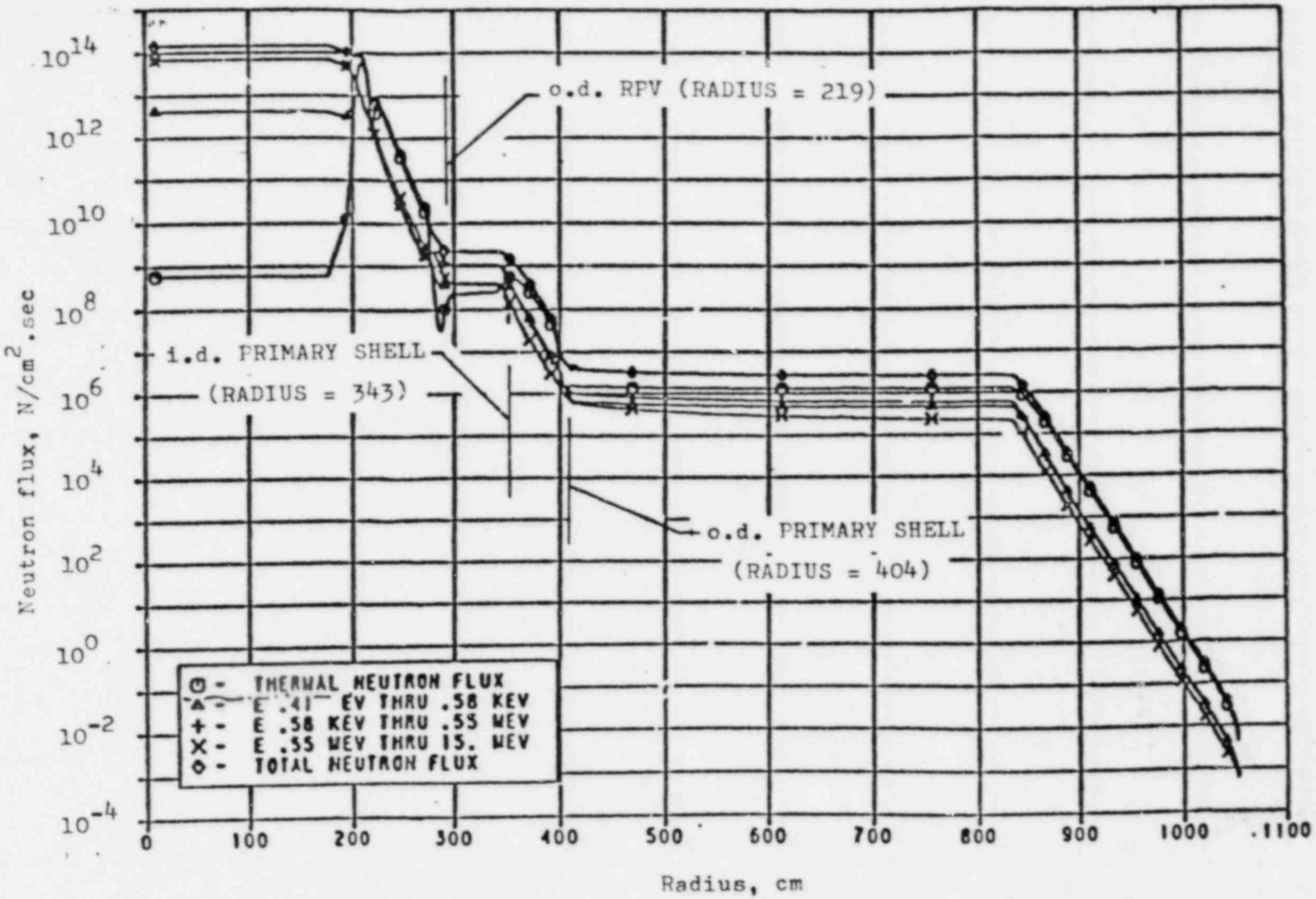

Figure 1. Radial distribution of neutron fluxes: full power.

ISSUE 3. TREND RECORDING

Issue Definition

The purpose of addressing Issue 3 is to determine which variables set forth in RG 1.97 need trend recording.

Discussion

Regulatory Guide 1.97, par. 1.3.2f, states the general requirement for trend recording as follows: "Where direct and immediate trend or transient information is essential for operator information or action, the recording should be continuously available for dedicated recorders." Using the BWR Owners Group Emergency Procedures Guidelines (EPG's) as a basis, the only trended variables required for operator action are reactor water level and reactor vessel pressure.

Conclusion

For LaSalle only reactor water level (variable B4) and reactor vessel pressure (variable B6) require trend recording.

ISSUE 4. VARIABLES B8 and C6

B8: Drywell Sump Level
C6: Drywell Drain Sumps Level

Issue Definition

Regulatory Guide 1.97 specifies Category 1 instrumentation to monitor drywell sump level (variable B8) and drywell drain sumps level (variable C6). These designations refer to the drywell equipment drain tank and floor drain tank levels. Category 1 instrumentation indicates that the variable being monitored is a key variable. In RG 1.97, a key variable is defined as ". . . that single variable (or minimum number of variables) that most directly indicates the accomplishment of a safety function. . . ." The following discussion supports the BWR Owners Group alternative position that drywell sump level and drywell drain-sumps levels should be qualified to Category 3 instrumentation requirements.

Discussion

The LaSalle drywell has two drain sumps. One drain is the equipment drain sump, which collects identified leakage; the other is the floor drain sump, which collects unidentified leakage.

Although the level of the drain sumps can be a direct indication of breach of the reactor coolant system pressure boundary, the indication is not unambiguous, because there can be water in those sumps during normal operation. Other instrumentation specified in RG 1.97 that would also indicate leakage in the drywell:

1. Drywell pressure--variable B7, Category 1
2. Drywell temperature--variable D7, Category 2
3. Primary containment area radiation--variable C5, Category 1

The drywell-sump levels signal neither automatic-protection control circuitry nor the operator to take safety related actions. Both sumps have level detectors that provide only the following nonsafety indications:

1. Continuous level indication
2. High-level alarm
3. High-high-level alarm

Regulatory Guide 1.97 specifies instrumentation to function during and after an accident. The drywell sump systems are deliberately isolated at the primary containment penetration upon receipt of an accident signal to establish containment integrity. This fact renders the drywell-sump-level signal irrelevant. Therefore, by design, drywell-level instrumentation serves no useful accident-monitoring function.

The Emergency Procedure Guidelines use the RPV level and the drywell pressure as entry conditions for the Level Control Guideline. A small line break will cause the drywell pressure to increase before a noticeable increase in the sump level. Therefore, the drywell sumps will provide a "lagging" versus "early" indication of a leak.

Conclusion

Based on the above facts, Edison believes that Category 3, "high-quality off-the-shelf instrumentation" is appropriate for drywell-sump level and drywell-drain-sumps level instrumentation.

ISSUE 5. VARIABLE C1

C1: Radioactivity Concentration or Radiation Level in Circulating Primary Coolant

Issue Definition

Regulatory Guide 1.97 specifies that the status of the fuel cladding be monitored during and after an accident. The specified variable to accomplish this monitoring is variable C1--radioactivity concentration or radiation level in circulating primary coolant. The range is given as "1/2 Tech Spec Limit to 100 times Tech Spec Limit, R/hr." In Table 1 of RG 1.97, instrumentation for measuring variable C1 is designated as Category 1. The purpose for monitoring this variable is given as "detection of breach," referring, in this case, to breach of fuel cladding.

Discussion

The usefulness of the information obtained by monitoring variable C1, in terms of helping the operator in his efforts to prevent and mitigate accidents, has not been substantiated. No particular planned operator action to be taken based on monitoring this variable is specified in the current version of the Emergency Procedure Guidelines (EPG's). The critical actions that must be taken to prevent and mitigate a gross breach of fuel cladding in a BWR are (1) shutdown the reactor and (2) maintain water level. Monitoring variable C1, as directed in RG 1.97, will have no influence on either of these actions. Any usefulness from this monitored variable falls in the category of "information that the barriers to release of radioactive material are being challenged" and "identification of degraded conditions and their magnitude, so the operator can take actions that are available to mitigate the consequences." There are no additional operator actions to mitigate the consequences of fuel barriers being challenged, other than those based on Type A and B variables.

Regulatory Guide 1.97 specifies measurement of the radioactivity of the circulating primary coolant as the key variable in monitoring fuel cladding status during isolation of the NSSS. The words "circulating primary coolant" are interpreted to mean coolant, or a representative sample of such coolant, that flows past the core. A basic criterion for a valid measurement of the specified variable is that the coolant being monitored is coolant that is in active contact with the fuel, that is, flowing past the failed fuel. Monitoring the active coolant (or a sample thereof) is the dominant consideration. The High Radiation Sampling System (HRSS) provides a representative sample which can be monitored in the post accident time frame.

The subject of concern in the RG 1.97 requirement is assumed to apply when a shutdown reactor is isolated. This is the only assumption that's justified for a BWR because on-line radiation monitors in the condenser off-gas and main steam lines provide reliable and accurate information on the status of fuel cladding when the plant is not isolated. Further, the High Radiation Sampling System (HRSS) will provide an accurate status of coolant radioactivity, and hence cladding status. Also, monitoring of the primary containment radiation and containment hydrogen concentration provides information on the status of fuel cladding integrity.

Later in the cooling sequence, the HRSS operation is augmented by area radiation monitors when the RHR system is being used to remove core decay heat.

Conclusion

The designation of instrumentation for measuring variable C1 should be Category 3, because no planned operator actions are identified and no operator actions are anticipated based on this variable. It does not serve as a key variable. Existing Category 3 instrumentation is adequate for monitoring fuel cladding status under all circumstances.

ISSUE 6. VARIABLE C14

C14: Radiation Exposure Rate

Issue Definition

Variable C14 is defined in Table 1 of RG 1.97 as follows: "Radiation exposure rate (inside buildings or areas, e.g., auxiliary building, fuel handling building, secondary containment, which are in direct contact with primary containment where penetrations and hatches are located." The reason for monitoring variable C14 is given as "Indication of breach."

Discussion

The use of local radiation exposure rate monitors to detect breach or leakage through primary containment penetrations is impractical and unnecesary. In general, radiation exposure rate in the secondary containment will be largely a function of radioactivity in primary containment and in the fluids flowing in ECCS piping, which will cause direct radiation shine on the areas of concern where radioactive fluids are piped. Also, because of the amount of piping and the number of electrical penetrations and hatches and their widely scattered locations, local radiation exposure rate monitors could give only local indications. The proper way to detect breach of containment is by using the plant noble gas effluent monitors provided for that specific purpose.

Conclusion

Using radiation exposure rate monitors to detect primary containment breach is neither feasible nor necessary. Other means of breach detection that are better suited to this function (as described above), are available. The existing monitors which indicate habitability are adequate for habitability and the stack monitor is adequate to monitor effluent from the stack.

ISSUE 7. VARIABLES D3 AND D8

D3: Suppression Spray Flow

D8: Drywell Spray Flow

Issue Definition

Regulatory Guide 1.97 specifies flow measurements of suppression chamber spray (SCS) (variable D3) and drywell spray (variable D8) for monitoring the operation of the primary containment-related systems. Instrumentation for measuring these variables is designated Category 2, with a range of 0 to 110 percent of design flow. These flows relate to spray flow for controlling pressure and temperature of the drywell and suppression chamber.

Discussion

The drywell sprays can be used to control the pressure and temperature of the drywell. The residual heat removal (RHR) system flow element is used for measuring drywell flow.

The pressure suppression chamber sprays can be used to control the pressure and temperature in the suppression chamber. From the control room the operator controls pressure and temperature by adjusting suppression chamber spray flow. The RHR system flow element is used for flow indication. The suppression chamber spray operates in parallel with the drywell spray and is regulated with a throttling valve. The flow is determined by RHR flow indication. The effectiveness of spray flow can be verified by pressure and temperature changes of the drywell and the suppression chamber as indicated in the control room.

Conclusion

The current plant equipment, in conjunction with operating practice, meet performance requirements of accuracy and reliability for measurement of spray flows into the drywell and suppression chamber.

ISSUE 9. VARIABLES D13-D17

D13: RCIC Flow

D14: HPCS Flow

D15: Core Spray System Flow (LPCS)

D16: LPCI System Flow

D17: SLCS Flow

Issue Definition

Regulatory Guide 1.97 specifies flow measurements of the following systems: reactor core isolation cooling (RCIC) (variable D13), high-pressure core spray (HPCS) (variable D14), low pressure (LPCS) (variable D15), low-pressure coolant injection (LPCI) (variable D16), and standby liquid control (SLC) (variable D17). The purpose is for monitoring the operation of individual safety systems. Instrumentation for measureing these variables is designated as Category 2; the range is specified as 0 to 110 percent of design flow. These variables cover the water flow rate into the reactor pressure vessel (RPV).

Discussion

The RCIC, HPCS, and LPCS systems each have one branch line--the test line--downstream of the flow-measuring element. The test line is provided with a motor-operated valve that is normally closed (two valves in series in the case of the HPCS). Further, this mo-valve in the test line closes automatically when the emergency system is actuated, thereby ensuring that indicated flow is not being diverted into the test line. Proper valve position can be verified by a direct indication in the control room of test valve position.

Although the three LPCI loops have several branch lines located downstream of each flow-measuring element, each of those lines is either normally closed or automatically aligned. Upon initiation of a LPCI

loop, the valves in that loop automatically line up for proper operation and prevent flow diversion into branch lines. Proper valve alignment can be verified by a direct indication of valve position in the control room.

For all of the above systems, there are valid primary indicators other than flow measurement to verify the performance of the emergency system, primarily vessel water level.

The SLC system is manually initiated. Flow-measuring devices were not provided for this system. The pump-discharge header pressure, which is indicated in the control room, indicates SLC pump operation. Besides the discharge header pressure observation, the operator can verify the proper functioning of the SLCS by monitoring the following:

1. The decrease in the level of the boric acid storage tank
2. The reactivity change in the reactor as measured by neutron flux which responds to the concentration of boron in the reactor core
3. The SLC motor contactor indicating lights (or motor current); the use of these indications is a valid alternative to SLCS flow indication
4. Squib valve continuity indicating lights
5. The open/close position indicators for the SLC check valves

Conclusion

The flow-measurement schemes for the RCIC, HPCS, LPCS, and LPCI are adequate in that they meet the requirements of RG 1.97. Monitoring the SLCS can be adequately done by measuring variables other than the injection flow.

ISSUE 10. VARIABLE D18

D18: SLCS Storage Tank Level

Issue Definition

Regulatory Guide 1.97 lists standby liquid-control system (SLCS) storage-tank level as a Type D variable with Cateogry 2 design and qualification criteria.

Discussion

The symptomatic Emergency Procedure Guidelines (EPG), Revision 1, as presently approved do not consider ATWS conditions; however, the EPG committee of the BWR Owners Group has been developing a draft reactivity control guideline in which procedures are described for raising the reactor water level based on the amount of boron injected into the vessel, as indicated by the SLC tank level. Additionally, the operator would be required to trip the SLC pumps before a low SLC tank level is reached, thereby preventing damage to the pumps that would render them useless for future injections during the scenario. The usefulness of this draft is not established yet, nor is the need or credibility of the ATWS scenario.

Regarding the instrumentation category requirement for variable D18, RG 1.97 indicates that it is a key variable in monitoring SLC system operation. Regulatory Guide 1.97 also states that in general, key Type D variables be designed and qualified to Category 2 requirements.

In applying these requirements of the Guide to this instrumentation, the following are noted:

1. The current design basis for the SLCS assumes a need for an alternative method of reactivity control without a concurrent loss-of-coolant accident or high-energy line break. The environment in

which the SLCS instrumentation must work is therefore a "mild" environment for qualification purposes.

2. The current design basis for the SLCS is recognized as considerably less than the importance to safety of the reactor protection system and the engineered safeguards systems. Therefore, in accordance with the graded approach to quality assurance specified in RG 1.97, it is unnecessary to apply a full quality-assurance program to this instrumentation.

Conclusion

SLCS storage-tank-level instrumentation should meet Category 3 design and qualification criteria as required by RG 1.97.

It is realized that the resolution of the ATWS issue may include substantial changes to the SLCS design criteria. At that time, the SLCS instrumentation should be reevaluated to ensure adequacy.

ISSUE 11. VARIABLE E2

E2: Reactor Building or Secondary Containment Radiation

Issue Definition

Regulatory Guide 1.97 specifies that "Reactor building or secondary containment area radiation" (variable E2) should be monitored over the range of 10^{-1} to 10^{4} R/h for Mark I and II containments, and over the range of 1 to 10^{7} R/hr for Mark III containments. The classification for Mark I and II is Category 2; for Mark III, the classification is Category I.

Discussion

As discussed in the variable C14 position statement (Issue 6), Secondary Containment Area Radiation is an inappropriate parameter to use to detect or assess primary containment leakage.

Conclusion

It is Edison's position that the specified Reactor Building area radiation monitors should not be required for LaSalle secondary containments.

ISSUE 12. VARIABLE E3

E3: Radiation Exposure Rate

Issue Definition

Regulatory Guide 1.97 specifies in Table 1, variable E3, that radiation exposure rate (inside buildings or areas where access is required to service equipment important to safety) be monitored over the range of 10^{-1} to 10^{4} R/hr for detection of significant releases, for release assessment, and for long-term surveillance.

Discussion

In general, access is not required to any area of the secondary containment to service equipment important to safety in a post-accident situation. If and when accessibility is reestablished in the long term, it will be done by a combination of portable radiation survey instruments and post-accident sampling of the secondary containment atmosphere. The existing lower-range (typically 3 decades lower than the RG 1.97 range) area radiation monitors would be used only in those instances in which radiation levels were very mild.

Conclusion

Because the LaSalle design does not require access to a harsh environment area to service safety-related equipment during an accident, this variable is provided from existing area radiation monitors. That is, this parameter should be reclassified as Category 3 with the existing range capability.

ISSUE 13. VARIABLE E13

E13: Primary Coolant and Sump

Issue Definition

Regulatory Guide 1.97 specifies installation of the capability for obtaining grab samples (variable E13) of the containment sump, ECCS pump-room sumps, and other similar auxiliary building sumps for the purpose of release assessment, verification, and analysis.

Discussion

The High Radiation Sample System can obtain grab samples of reactor coolant containment sumps, RHR, and RWCU.

- A reactor coolant grab sample is a conservative estimate of ECCS room sump radioactivity. Reactor coolant sample information would be available in a much more timely manner than sump activity. The reactor building floor drain sumps are isolatable by locally tripping the pumps or their power supply.

- Sampling as close to the source of radiation as possible, ECCS and auxiliary building sump radiation can be conservatively estimated.

Conclusion

Conservative estimates of sump activity can be made without sampling, LaSalle will not implement sample capability of ECCS room floor drain sumps or auxiliary building sumps.

ISSUE 14: VARIABLE E8

E8: Plant and Environs Radiation

Issue Definition

Regulatory Guide 1.97 specifies that "Plant and Environs Radiation" (variable E8) should be monitored over the range of 10^{-3} R/hr to 10^{4} R/hr, photons and 10^{-3} rads/hr to 10^{4} rads/hr, beta radiations and low energy photons. The classification is category 3.

Discussion

Portable plant radiation detectors have ranges from 10^{-3} r/hr to 10^{3} R/hr. Conservative calculation predict most areas of the plant will be less than 500 R/hr after a worst case accident. To monitor an area greater than the range of our existing monitors would not be practical because the technician involved would expose himself to unnecessarily high levels of radiation. No decisions regarding habitability are made with readings above the presently monitored range.

Conclusion

A range of monitoring to 10^{4} R/hr would not enhance plant and environs radiation monitoring. LaSalle meets the intention and the purpose of the Reg. Guide criteria.

E. Conclusion

LaSalle has control room indication of all Reg. Guide 1.97, Table 1 parameters except core thermocouples and primary coolant radioactivity. Alternate methods provide information relative to the exceptions. A tabulation of what variables are indicated and what criteria are met summarizes the LaSalle position. Justifications for the differences between the LaSalle design and Reg. Guide criteria are in the issues section following the table. Two actions discussed below will implement our position on Reg. Guide 1.97.

The Reg. Guide criteria that indicators of RG 1.97 parameters be easily discerned for use during accident conditions should be implemented as part of a control room design review rather than addressing it as a single issue. An acceptable method of identifying RG 1.97 parameters in the control room will be investigated as part of the detailed control room design review to be performed.

Adequacy of reliably powered, qualified neutron monitor was justified in Issue 2. The existing system does not meet proposed criteria, therefore LaSalle will upgrade neutron monitoring capability to category 2 criteria. Because time will be required for hardware development and installation, completion of this plant modification is scheduled prior to the third fuel cycle.

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