ACCELERATED DOCUMENT DISTRIBUTION SYSTEM

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9304060401 DOC.DATE: 93/03/31 NOTARIZED: NO DOCKET #
FACIL:50-387 Susquehanna Steam Electric Station, Unit 1, Pennsylva 05000387

AUTH.NAME	AUTHOR AFFILIATION
LOCHBAUM,D.A.	Affiliation Not Assigned
PREVATTE,D.C.	Affiliation Not Assigned

RECIP.NAME	RECIPIENT AFFILIATION
MCCRACKEN,C.E.	Plant Systems Branch

SUBJECT: Part 21 rept consisting of comments on NRC/util 930318 meeting in Rockville,MD re failure of facility design to meet regulatory requirements for cooling of spent fuel pool under DBA conditions.

DISTRIBUTION CODE: IE19D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 16
TITLE: Part 21 Rept (50 DKT)

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD1-2 PD	1	1	CLARK,R	1	1
INTERNAL:	AEOD/DSP/ROAB	1	1	NRR/DRIL/RVIB	1	1
	REG FILE 01	1	1	RES/DSIR/EIB	1	1
	RGN1	1	1	RGN2	1	1
	RGN3	1	1	RGN4	1	1
	RGN5	1	1	SECY VANDERMEL	1	1
EXTERNAL:	INPO RECORD CTR	1	1	NRC PDR	1	1
	NSIC SILVER,E	1	1			

R I D S / A D D S

R I D S / A D D S

NOTE TO ALL "RIDS" RECIPIENTS:

PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM P1-37 (EXT. 504-2065) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR 15 ENCL 15

JA1

March 31, 1993

David A. Lochbaum
80 Tuttle Road
Watchung, NJ 07060
Telephone: 908 754-3577

Donald C. Prevatte
7924 Woodsbluff Run
Fogelsville, PA 18051
Telephone: 215 398-9277

Mr. Conrad E. McCracken
Chief - Plant Systems Branch
United States Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION
DOCKET NO. 50-387
LICENSE NO. NPF-14
10CFR21 REPORT OF SUBSTANTIAL SAFETY HAZARD
COMMENTS ON NRC/PP&L MEETING OF 3/18/93

Dear Mr. McCracken:

On March 18, 1993, Pennsylvania Power & Light Company (PP&L) representatives met with staff members of the U.S. Nuclear Regulatory Commission (NRC) at your White Flint offices to respond to NRC questions regarding the 10CFR21 report submitted by the undersigned. This report documented a substantial safety hazard that exists as a result of failure of the design of the Susquehanna Steam Electric Station (SSES) to meet numerous regulatory requirements with regard to the cooling of the spent fuel pool <u>under design basis accident (DBA) conditions</u> and the failure of PP&L to take appropriate actions at the times this condition was repeatedly brought to their attention. This meeting was also attended by the undersigned as observers. This letter is to bring to your attention the areas where the licensee failed to answer the NRC's questions or our 10CFR21 report concerns.

First, as we experienced in our months of discussions with PP&L before the 10CFR21 report was submitted, the licensee diverted the focus of attention from the unresolved DBA concerns to the relatively minor point for which analyses have been performed since our 10CFR21 report - the capacity of the non-safety-related fuel pool cooling system to handle <u>normal</u> refueling heat loads. Fully three-fourths of the presentation time was devoted to this relatively minor and uncontested point, and only a few minutes at the end of the presentation to the primary DBA concerns.

IE19
1/1

Next, in the portion of the presentation that was devoted to the DBA concerns, misleading, incomplete, and incorrect answers were provided to the NRC's questions. The following addresses these areas of disinformation point-by-point:

Licensing/Design Basis Requirements for Fuel Pool Cooling

<u>PP&L Position</u>: The only SSES design basis event in which normal spent fuel cooling is lost is a seismic event during refueling as described in FSAR Appendix 9A.

<u>Discrepancies with PP&L Position</u>: Contrary to PP&L's position that cooling of the fuel pool is only required to be accomplished under the non-accident condition described in FSAR, Appendix 9A, per the first paragraph of Reg. Guide 1.13, Spent Fuel Storage Facility Design Basis, facilities must be designed for cooling the fuel pool under "accident conditions". These same requirements are reflected in 10CFR50, Appendix A, Criterion 61 and NUREG 0800, Standard Review Plan, Section 9.1.3. All of these require the capability of maintaining the fuel pool water inventory and hence cooling under all normal operating and, contrary to the PP&L position, accident conditions.

The licensing basis for the fuel pool must be considered in the context of all the licensing requirements for the plant, not just one section of the FSAR. It must be considered along with all the design basis events and conditions described in the FSAR which can mechanistically cause failure of the normal fuel pool cooling system during a DBA LOCA.

The accident conditions described in the FSAR include the DBA LOCA itself which produces hydrodynamic loads and environmental conditions for which the fuel pool cooling system is not designed, the design basis seismic event (SSE), loss of offsite power (LOOP) which deenergizes the system's electrical components (pumps, valves, etc.) as well as those for the non-safety-related service water system which supplies the cooling water for the system's heat exchangers, and random failure of any component in the fuel pool cooling or service water systems since they are both non-safety-related and therefore not designed for single failure. It is failure of the fuel pool cooling system during a LOCA due to any of these design basis accompanying conditions for which we are most concerned since the plant's equipment, analyses, training, and procedures had not accounted for these conditions at the time of our 10CFR21 report, and appear not to be complete even today.

Additionally, in 1988, the emergency operating procedures for the plant were changed due to EQ temperature problems in the reactor building to require that non-1E power to the building be deenergized at 24 hours into a LOCA to reduce the building heat load. This would provide yet another "failure" mechanism for the fuel pool cooling system during a DBA LOCA.

And finally, even for the seismically induced failure event without LOCA described in FSAR Appendix 9A which PP&L contends is their only required licensing basis, only the radiological consequences of a boiling spent fuel pool had been considered in the plant design at the time of the 10CFR21 report. No analyses had been performed of the effects on the systems and equipment in the reactor building due to the additional heat and water (condensation and overflow) from a boiling spent fuel pool, and it does not appear that these analyses are complete and comprehensive even today.

Reactor Building Access Post-LOCA

PP&L Positions: PP&L maintains that the reactor building will be totally accessible for the operators post-LOCA to perform whatever actions are necessary to cope with a failure of the spent fuel pool cooling system, including opening of the manual ESW valves which supply makeup water to the boiling fuel pool. This position is based on a recent 10CFR50, Appendix K mechanistic core thermal-hydraulic analysis for LOCA conditions which is currently under review, but not yet approved by the NRC, which shows that no fuel failure will occur. With no fuel failure, there is no release of radionuclides, and the reactor building remains clean.

PP&L also maintains that even if they use Reg. Guide 1.3 assumptions, (1) they are not required to consider the airborne radiation effects in the reactor building, (2) they will place shielding around the extremely high radiation sources close to the ESW valves that have to be manipulated, and (3) they will have access to the non-accident unit in which they can effect the necessary valve manipulation to provide makeup water to the fuel pools.

Discrepancies with PP&L Positions: The licensing basis requirements for fuel failure which must be considered as a result of a LOCA and the requirements for operator access to areas containing equipment which must be operated post-LOCA have been clearly and unambiguously stated in numerous NRC documents, including Reg. Guide 1.3, Reg. Guide 1.4, NUREG 0737, an NRC letter to all operating nuclear power plants dated October 30, 1979 (copy attached), and 10CFR50, Appendix A, Criterion 19. These all require personnel access analyses to be based on non-mechanistic assumptions of fuel failure resulting in releases of 50% of the radioiodine, 100% of the noble gases, and 1% of the core solids inventory into the reactor coolant. They also require that the design maximum personnel radiation exposure for accident conditions not exceed 5 rem whole body equivalent. PP&L has acknowledged and committed to these requirements in numerous licensing/design basis documents including the FSAR and the design basis calculations in their files. These all show radiation levels at the ESW valves which are required to be operated in thousands of rem per hour, and airborne radiation levels on the refueling floor in hundreds of rem per hour. The FSAR concludes "that the reactor building is generally inaccessible for several days after the accident due to contained sources". This conclusion is made even without considering the contribution from airborne sources.

And yet, for this issue, PP&L appears to have adopted a new licensing/design basis without the benefit of review and approval by the NRC. (Using these new bases, it could also be argued that there is no need for primary containment, secondary containment, the standby gas treatment system, the control room CREOASS system, or the meteorological monitoring system. However, to suggest that these features are not required is equally as absurd as PP&L's position regarding reactor building access.)

PP&L contends that their 10CFR50, Appendix K analysis showing no fuel failure is more realistic than their licensing basis, and therefore, they are using it instead. However, in the only large LOCA that has been experienced in a commercial reactor in this country, the TMI accident, the actual fuel failure was virtually total. For this reason, the requirements of Reg. Guide 1.3 were reiterated by the NRC in the October 30, 1979 NRC letter and in NUREG 0737.

In addition to PP&L's current position being outside the licensing basis, we contend that their current analysis is not necessarily more realistic, even though it may be more sophisticated and more precise than previous analyses. But, precision does not necessarily equate to realism. We believe that the actual experience of TMI is the best example of realism.

Concerning PP&L's position that they are not required to consider airborne radiation in the reactor building, it is based on their not having addressed these effects in the FSAR and this deficiency not having been caught in the original NRC reviews. Their position is, in effect, that if the NRC didn't catch it, it's acceptable to ignore it.

First of all, that attitude is not acceptable. And secondly, the NRC did catch it, later, in their review of PP&L's SPING system. PP&L was informed by the NRC in a letter dated 4/23/84 that due to airborne radiation post-LOCA, the reactor building would be inaccessible for SPING sampling. PP&L acknowledged this in their letters PLA-2133 and PLA-2219, and they made changes in the SPING system design accordingly. These changes were acknowledged in an NRC letter dated 9/19/85. PP&L's present position is totally inconsistent with this case.

PP&L's contention that even if they did have Reg. Guide 1.3 levels of fuel failure, it would be no problem, since they would place shielding around the pipes, is ludicrous. First, they have no contingency plans, procedures, training, or staged materials and equipment to execute this evolution. Second, such an evolution would require considerable time just due to the size and weight of the shielding that would be required for such very large, extremely radioactive sources. The size of this job would preclude timely action to perform the necessary valve operations. And finally, the placing of the shielding would entail significantly more close range exposure to these extremely high radiation sources than would the manipulation of the valves without the shielding.

PP&L's position that they would have access to the non-accident unit to effect the necessary valve manipulation is a new argument. At the time of the 10CFR21 report, the reactor building HVAC system design was such that for LOCA/LOOP, all three HVAC zones would automatically cross-connect and go into the recirculation mode, effectively creating uniform airborne radiation conditions in both the accident and non-accident units.

Loss-of-Offsite-Power

PP&L Position: PP&L maintains that the probability of LOOP is very low, and that even if it should occur, it would be of such short duration that the pool would not boil before power was restored. Therefore, they argue, they do not have to consider the effects of long-term LOOP in their design for fuel pool cooling.

Discrepancies with PP&L Position: PP&L's position is based purely on probabilistic arguments and does not conform with regulatory requirements. Probability arguments are sometimes acceptable as justifications for interim operation, which is not the case here, but they are not legitimate substitutes for the required design bases. The design requirements for long-term LOOP are contained in 10CFR50, Appendix A, Criterion 17 which requires that LOOP be considered in the plant design, Reg. Guide 1.137 which requires that emergency diesels have sufficient fuel for operation for at least seven days at full load, and in numerous other regulations requiring 30-day capability for safety-related systems for coping with accident conditions, e.g., ultimate heat sink capacity.

PP&L's probability arguments also do not hold with common sense and recent experience in the industry. Long-term LOOPs can result from a seismic event, a tornado, or other natural phenomenon for which 10CFR50, Appendix A, Criterion 2 requires the plant to be designed. An example of where such an event caused a LOOP of many days is the experience of the Turkey Point Plant with Hurricane Andrew. It can also result from sabotage or from human error as was the case of the recent Plant Vogtle LOOP.

PP&L's probability arguments rely heavily on the demonstrated reliability of the PP&L grid. By analogy, they could also argue that they need not consider a LOCA in their design basis since SSES has never experienced one. Such a position does not conform with the non-mechanistic philosophy the NRC has consistently taken regarding licensing requirements for LOCA or for LOOP.

Additionally, as described earlier, PP&L's emergency procedures have them deenergize non-1E power to the reactor building at 24-hours into an accident if temperatures in the building approach calculated values. This, in effect, imposes a 100% probability of a long-term LOOP.

Another weakness of their probability arguments is their contention that power can be restored within the calculated twenty-five hour time to boil. For certain refueling conditions, the time to boil may actually be significantly less. Additionally, even the twenty-five hour time to boil is not realistic for evaluating the EQ threat to reactor building equipment for the following reason. The current reactor building temperature analysis for EQ uses 125 degrees F for the fuel pool temperature. As the actual pool temperature increases above this value, it begins to affect the reactor building environment long before it reaches the boiling point by producing heat and condensing vapor conditions which have not been analyzed and for which the safety-related equipment in the building has not been designed.

Use of RHR Fuel Pool Cooling Mode Post-LOCA

PP&L Position: PP&L contends that even if there is failure of the normal fuel pool cooling system in a DBA LOCA, they can cool the pool using the RHR system in the fuel pool cooling mode.

Discrepancies in the PP&L Position: There are many glaring discrepancies in this position.

First, the use of this mode of RHR also requires operator access to the reactor building to manipulate manual valves which, as discussed previously, is not possible due to radiation levels.

Second, analyses show that this mode of operation cannot be achieved. Although the RHR heat exchangers may have sufficient heat transfer capability, there is insufficient NPSH at the RHR pumps to deliver the required flow.

Third, this mode of operation for RHR has never been successfully tested. Preoperational tests of the system in this mode were abandoned when the required flow could not be achieved, and the failure was written off because this was not a safety-related mode of operation of the system.

Fourth, in a DBA LOCA, if there is a single failure of one of the RHR divisions as is required to be considered in the design, there is no RHR division available to perform fuel pool cooling.

Fifth, in the RHR fuel pool cooling mode, the heat from the fuel pool is rejected to the spray pond ultimate heat sink (UHS) instead of the cooling tower as it is during normal operation. The UHS has not been analyzed for this additional heat load for a DBA LOCA, and current analyses indicate that its temperature would substantially exceed the design allowable under this condition.

Sixth, the valves for this mode have been removed from the inservice inspection program, and this mode is not safety-related as required by Reg. Guide 1.13 for the fuel pool cooling backup.

Seventh, use of RHR post-LOCA in the fuel pool cooling mode will transmit the fission product laden reactor coolant to the spent fuel pool. Current plant shielding analyses and offsite and control room dose analyses do not consider source terms of that magnitude in that location.

Environmental Effects of the Boiling Spent Fuel Pool

PP&L Position: PP&L has presented very little in addressing this question. Their meeting presentation only addressed the condensation from the boiling spent fuel pools, saying it would be collected in the reactor building sumps where it would pose no threat to safety related equipment, and it would be pumped to radwaste to be processed.

Discrepancies in PP&L's Position: On the only environmental point addressed by PP&L, there are many discrepancies.

First, the volume of water cited in the meeting that would have to be dealt with is considerably less than what the design basis analyses show. PP&L indicated there would be approximately one million gallons going to the sumps, which they were capable of holding. This volume is based on an assumption that ESW makeup flow can be exactly matched to the boiloff rate. However, since ESW makeup would be accomplished by manual, unmodulated valve positioning, and since the boiloff rate would be constantly changing, it would be impossible for the flow rate to be perfectly matched.

The flow rate must therefore must be set to at least the design flow rate of 60 gpm to each pool, for a total of 5.2 million gallons that would have to be handled over the thirty day accident duration. It is doubtful the sumps have the capacity to hold this volume.

Second, the sump pumps may not be available because they are non-safety-related, non-1E powered, and not environmentally qualified.

Third, even if the sump pumps are available, radwaste is also non-safety-related, non-1E powered, and it is not designed to handle this volume of water or water laden with DBA LOCA core fission products.

And finally, the purpose of secondary containment as stated in the FSAR is to keep accident products inside the reactor building. To introduce procedures contrary to that intent creates an very serious unreviewed safety question.

PP&L did not address other issues associated with the moisture from the boiling spent fuel pool. Some of these are: What are the effects of condensation in the safety-related HVAC ductwork? Are the ducts designed for the weight of condensate accumulation, and if not (they are not), what would be the effects of structural failures of the ductwork on other safety-related equipment? What would be the effects of water pouring out of ductwork on safety-related electrical equipment? What are the effects of moisture condensing inside safety-related electrical equipment? What are the effects on the safety-related air flow in the ducts for accumulations of water even if structural failure did not occur? What are the effects of condensation in the safety-related HVAC coolers in the building which are not designed for latent heat removal?

PP&L also did not address the other environmental effects of the boiling spent fuel pool, such as the addition of approximately 20.9 million BTU/hour of heat load to the building whose equipment is currently environmentally qualified for only 5.2 million BTU/hour, or the actuation setpoint of the standby gas treatment system fire dampers versus the temperature of the reactor building, and other effects.

Qualifications of Monitoring Instrumentation

PP&L Position: PP&L presented virtually no justification for the fuel pool level and temperature monitoring instrumentation not being environmentally or seismically qualified, or 1E powered.

Discrepancies in PP&L Position: The current fuel pool monitoring instrumentation does not have remote readout capability, but requires reactor building entry to take readings. This presents the same problems of operator access as were discussed above.

10CFR50, Appendix A, Criterion 63, Regulatory Guide 1.97, and NUREG 0800, Sections 7.1 and 9.1.3 require that fuel pool level and temperature measuring instrumentation be environmentally and seismically qualified, and 1E powered. The PP&L instrumentation does not meet these criteria.

Reliance on Emergency Organization

PP&L Position: PP&L also took great credit in their presentation, as in our past conversations, for the capabilities of their emergency organization to cope with a loss of normal fuel pool cooling during a DBA LOCA.

Discrepancies in PP&L Position: PP&L seems to lack a valid understanding of the purpose of an emergency organization. Such an organization is intended to provide one of the outer lines of defense in the concept of defense-in-depth, the first element being correct design and procedures. If this first element succeeds in its purpose, then the emergency organization should have to take no action. PP&L's arguments substitute the emergency organization for proper design and procedures, thereby sacrificing the first line of defense. This is not, and has never been, the NRC's intent.

Additionally, the conditions cited in our 10CFR21 report had never even been recognized by the emergency organization, and no preparations, procedures, or analyses had been performed on possible contingencies. We find it difficult to envision that the emergency organization would have been able to formulate the appropriate actions to cope with a loss of fuel pool cooling while simultaneously coping with a DBA LOCA, when the overall PP&L organization working for many months under relatively unstressful conditions has not been able to formulate the appropriate actions.

The regulations covering nuclear power essentially pose a question: Will the system perform its function under all conditions for which it is required without adversely affecting the functions of other systems? When the answer to this question is "yes", for all systems needed to mitigate design basis postulated events, the health and safety of the public is assured.

We submitted a 10CFR21 report for the loss-of-fuel-pool cooling event at SSES because the answers to several applicable questions were either "no" or "maybe". Since our report, PP&L has performed extensive evaluations, yet there remain many "no" and "maybe" answers. PP&L is justifying its position by condensing the question when necessary to permit a "yes" response. For example, PP&L proposes to use the RHR fuel pool cooling mode in the event normal fuel pool cooling is lost. This can be done - sometimes. However, this cannot be done under all conditions and without adversely affecting other aspects of the plant design, as previously discussed. With the health and safety of the public in the balance, it is imperative that all the questions be fully addressed, and all the answers be "yes".

PP&L's current awareness, procedures, modifications, training, etc with regard to these issues, although inadequate, are better than at the time of our 10CFR21 report. But their progress to-date, such as it is, has not been spontaneous, as the law, their procedures, common sense, and ethical responsibilities dictate it should have been when these issues were first raised. Their current progress has come about essentially only since, and we believe, only as a direct result of, our decision to report these issues to the NRC after their steadfast refusal to do so. This is not how the system is supposed to work.

We see this as a test case where the licensee has blatantly refused to do the right things, relying on the complexity of the issues and the difficulties in opposing the Organization to prevail in their favor. If this attitude is left to stand, then the system will have failed, and this case will become yet another example feeding the cynicism of the working level people in the PP&L organization - an example of what happens when tough safety issues are raised. All hope for a credible system of reporting such issues will have been lost.

But our hope is that this will not happen. We believe in the system. Otherwise, we would not have come this far.

Yours very truly,

Donald C. Prevatte for
David A. Lochbaum

Donald C. Prevatte
Donald C. Prevatte

cc:

US Nuclear Regulatory Commission
Attention: Document Control Clerk
Mail Station P1-137
Washington, DC 20555

Director, Office of Nuclear Reactor Regulation
US Nuclear Regulatory Commission
Washington, DC 20555

Mr. Thomas T. Martin
Regional Administrator, Region I
US Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

Mr. David Shum
US Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation, SPLB
Washington, DC 20555

Mr. Lawrence E. Kokajko
US Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation, DSSA/SPLB
Washington, DC 20555

Mr. Jay Lee
US Nuclear Regulatory Commission, DRSS/PRPB
Washington, DC 20555

Mr. Charles Miller
US Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation, PDI-2
Washington, DC 20555

Mr. Morton Fleishman
US Nuclear Regulatory Commission, OCMKR
Washington, DC 20555

Mr. John R. White
US Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

Mr. Richard Clark
US Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, DC 20555

Richert/Lambach

Dec 11/14/79 (B)
11/16/79 (A)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D C 20555

October 30, 1979

IF-79-148

(TO ALL OPERATING NUCLEAR POWER PLANTS)

REF JAF NRC DOC

Gentlemen:

SUBJECT: DISCUSSION OF LESSONS LEARNED SHORT TERM REQUIREMENTS

On September 13, 1979, a letter was issued to each power reactor licensee which defined a set of "short term" requirements resulting from the NRC staff investigations of the TMI accident. Since the letter was issued, the staff has attempted to further define these requirements. During the week of September 24, 1979, seminars were held in four regions of the country to encourage industry feedback and dialogue on each short term requirement. As a result of these discussions, four topical meetings were held in Bethesda to discuss certain issues in further detail.

Enclosure 1 provides additional clarification of the NRC staff requirements. It should be noted that the intent of these requirements have not changed throughout this process and are restated in Enclosure 1.

Enclosure 2 is a chart of the NUREG-0578 items and their corresponding implementation schedules. The chart indicates which of the items require prior NRC review and approval and those for which post implementation NRC review is acceptable.

(A) For those items requiring prior NRC approval, your design details should be submitted in a timely manner so that this approval and your implementation of the item can be completed by the required date. For those items which do not require prior NRC approval, you must document your method of implementation by the required completion date. These schedules assume that your methods are in complete agreement with the staff's requirements as previously documented in NUREG-0578, our September 13, 1979 letter, and clarified herein. Where your methods are not in complete agreement with the staff's requirements, a detailed description of your proposed methods along with justification for the differences, is required. Please provide this description and justification as soon as possible but no later than 15 days following receipt of this letter.

DSR # 38917
OF PAGES ~~84~~ 75

DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSEMS WHICH MAY BE USED IN POST ACCIDENT OPERATIONS (2.1.6.b)

POSITION

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory, and 1% of the core solids, are contained in the primary coolant), each licensee shall perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

CLARIFICATION

Any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is designated as a vital area. In order to assure that personnel can perform necessary post-accident operations in the vital areas, we are providing the following guidance to be used by licensees to evaluate the adequacy of radiation protection to the operators:

1. Source Term

   The minimum radioactive source term should be equivalent to the source terms recommended, in Regulatory Guides 1.3, 1.4, 1.7 and Standard Review Plant 15.6.5. with appropriate decay times based on plant design.

a. Liquid Containing Systems: 100% of the core equilibrium noble gas inventory, 50% of the core equilibrium halogen inventory and 1% of all others are assumed to be mixed in the reactor coolant and liquids injected by HPCI and LPCI.

b. Gas Containing Systems: 100% of the core equilibrium noble gas inventory and 25% of the core equilibrium halogen activity are assumed to be mixed in the containment atmosphere. For gas containing lines connected to the primary system (e.g., BWR steam lines) the concentration of radioactivity shall be determined assuming the activity is contained in the gas space in the primary coolant system.

2. Dose Rate Criteria

The dose rate for personnel in a vital area should be such that the guidelines of GDC 19 should not be exceeded during the course of the accident. GDC 19 limits the dose to an operator to 5 Rem whole body or its equivalent to any part of the body. When determining the dose to an operator, care must be taken to determine the necessary occupancy time in a specific area. For example, areas requiring continuous occupancy will require much lower dose rates than areas where minimal occupancy is required. Therefore, allowable dose rates will be based upon expected occupancy, as well as the radioactive source terms and shielding. However, in order to provide a general design objective, we are providing the following dose rate criteria with alternatives to be documented on a case-by-case basis. The recommended dose rates are average rates in the area. Local hot spots may exceed the dose rate guidelines provided occupancy is not required at the location of the hot spot. These doses are design objectives and are not to be used to limit access in the event of an accident.

a. Areas Requiring Continuous Occupancy: $\leq$15mr/hr. These areas will require full time occupancy during the course of the accident. The Control Room and onsite technical support center are areas where continuous occupancy will be required. The dose rate for these areas is based on the control room occupancy factors contained in SRP 6.4.

b. Areas Requiring Infrequent Access: GDC 19. These areas may require access on a regular basis, but not continuous occupancy. Shielding should be provided to allow access at a frequency and duration estimated by the licensee. The plant Radiochemical/Chemical Analysis Laboratory, radwaste panel, motor control center, instrumentation locations, and reactor coolant and containment gas sample stations are examples where occupancy may be needed often but not continuously.