Comments by Cob 2/26
D Tell Bournia to watch the introduction to sec II.B.2 - he puts equipment qualification in introduction but there is nothing of substance on EQ

cBour 2/27

12.0 RADIATION PROTECTION

The Final Safety Analysis Report provides information on the methods for radiation protection including the facility design and layout, equipment design, and a description of the health physics program. An estimate of occupational radiation exposure to plant personnel is included. Shielding is provided to reduce radiation levels. Ventilation is arranged to control the flow of potentially contaminated air. Radiation monitoring is employed to measure levels of radiation in potentially occupied areas and to measure airborne radioactivity throughout the plant. A health physics program is provided for plant personnel and visitors during reactor operation, maintenance, refueling, radwaste handling, and inservice inspections. We reviewed and evaluated the description and analysis of the radiation protection program included in the Final Safety Analysis Report.

The criteria used to determine acceptability of the program are that doses to personnel will be maintained within the established limits of 10 CFR Part 20 and that design and program features are consistent with the guidelines of Regulatory Guide 8.8, "Information Relevant to Ensuring Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," Revision 2. In response to our request, the applicant provided in the Final Safety Analysis Report extensive information concerning design improvements made for the purpose of assuring that occupational radiation exposures will be "as low as is reasonably achievable."

The applicant has considered means to keep external and internal radiation exposures to personnel, including both individual and total man-rem doses, "as low as is reasonably achievable." The facility has been designed to control radiation exposure such that doses to plant personnel will be "as low as is reasonably achievable" in accordance with the general guidance of Regulatory Guide 8.8. Since the construction permit was issued and much of the design completed long before Regulatory Guide 8.8 was issued, some of the specific guidance of Regulatory Guide 8.8 could not be met in a cost effective manner.

On the basis of our review, we conclude that the radiation protection program will provide reasonable assurance that doses to personnel will be less than the limits established by 10 CFR Part 20 and maintained "as low as is reasonably achievable" consistent with the intent of Regulatory Guide 8.8. The La Salle radiation protection program is, therefore, acceptable. Details are discussed in the following sections.

12.1 Assuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable

The applicant's management is committed that La Salle be designed, constructed, and operated such that occupational radiation exposures will be "as low as is reasonably achievable." The station Radiation/Chemistry Supervisor and the station Health Physicist are responsible for the implementaiton of this commitment. These policy considerations are consistent with the guidance of Regulatory Guide 8.8 and Regulatory Guide 1.8, "Personnel Selection and Training," Revision 1.

To minimize radiation exposures, the applicant has incorporated general considerations into the design to reduce: (1) the need to enter radiation fields, (2) the time of exposure when entry is necessary, and (3) the dose rate during exposure. These general considerations are implemented by detailed radiation protection design goals and guidelines. Also, information gained from the applicant's long experience with power reactors is factored into the design. Finally, design reviews are performed by radiation protection personnel to ensure that occupational radiation exposures will be "as low as is reasonably achievable." These design considerations are consistent with the guidance of Regulatory Guide 8.8 and are acceptable.

The applicant is also developing operational procedures to maintain exposures "as low as is reasonably achievable." The applicant has committed to include in these procedures measures for reducing exposure and the criteria for implementation of those measures consistent with the guidance of Regulatory Guide 8.8.

Based on the information provided, we conclude that the applicant intends to operate and maintain La Salle in such a manner that occupational radiation exposures will be "as low as is reasonably achievable."

12.2 Radiation Sources

The location and source terms of the contained radiation sources which must be shielded or included in the dose assessment are provided. The basis for the source terms meets our acceptance criteria as described below. The fission product source terms are based on an offgas rate of 100,000 microcuries per second after 30 minutes delay. The coolant and corrosion activation product source terms are based on measurements at operating boiling water reactors, and are consistent with American National Standard N.237-1976, "Source Term Specification." Neutron and prompt gamma source terms are based on reactor core physics calculations and operating reactor experience. The contained radiation source terms presented are comparable to estimates by other applicants with boiling water reactor designs and are acceptable.

The sources of airborne radioactivity and concentrations of airborne radioactivity in various parts of the station are provided. The methods used to calculate those airborne concentration estimates are consistent with commonly accepted methods and are, therefore, acceptable. The airborne radioactivity source terms presented are comparable to estimates by other applicants with boiling water reactor designs and are acceptable.

Based on the information provided, we conclude that the resulting radiation sources estimates are consistent with the acceptance criteria in Section 12.2 of the Standard Review Plan.

12.3 Radiation Protection Design Features

The applicant has indicated that the dose accumulating functions performed by workers have been considered in the plant design. Features have been included in the design to help maintain exposures "as low as is reasonably achievable" in the performance of those functions. These features will facilitate access to work areas, reduce or allow the reduction of source intensity, reduce the time required in radiation fields, and provide for portable shielding and remote handling tools. The applicant's facility design features are consistent with the guidance of Regulatory Guide 8.8. Therefore, we conclude that the facility design features are acceptable.

The areas inside the restricted area are divided into a number of radiation dose rate zones for design purposes. The areas that will have to be occupied on a predictable basis during normal operations and anticipated occurrences are zoned such that exposures will be below the limits of 10 CFR Part 20 and "as low as is reasonably achievable." The zoning system and access control features also meet the posting and entry requirements of 10 CFR Part 20.203. Therefore, we conclude that the design dose rate zone system is acceptable.

Several of the recommendations of Regulatory Guide 8.8 are included in the plant design and operational program to minimize the buildup of activated corrosion products, a major contributor to occupational dose. The use of high cobalt, hard facing wear materials in the primary system has been limited to those places where it is necessary. Valves and pipe connections have been designed to minimize this buildup. Control of pH and flow velocity in the primary system will also help minimize this buildup. Finally, the applicant is considering filtering forward-pumped heater drains. The plant design was too far along to initiate all of the recommendations of Regulatory Guide 8.8 cost-effectively, such as low cobalt impurity specifications for primary system alloys. Therefore, we conclude that the applicant has given acceptable consideration to the inclusion of design features that minimize the buildup of activated corrosion products.

The applicant has provided sufficient shielding to maintain low radiation levels for the potentially occupied areas adjacent to the spent fuel pools during fuel transfer. This is done by using a portable fuel transfer shield. This shield will span the gap between the reactor vessel and the containment wall and will provide shielding between the fuel assemblies and the potentially occupied upper levels of the drywell during refueling operations. This temporary shielding will be used only during transfer of spent fuel between the reactor vessel and the fuel pool and will reduce the dose rate in the vicinity of the highest drywell grating to less than 50 millirem per hour. We find the shielding provided by this design acceptable.

The applicant has not included features in the radiation protection design specifically for the purpose of maintaining doses "as low as is reasonably achievable" during decommissioning. However, many of the features included in

the design to reduce doses during operation will also help reduce doses during decommissioning. The applicant estimates that the collective occupational dose due to decommissioning will be of the same order of magnitude as annual doses due to operation. Therefore, we conclude that the applicant has given acceptable consideration to the issue of personnel exposure during decommissioning.

The shielding was designed to meet the requirements of the radiation dose rate zone system discussed above. The applicant's shielding design methods, including the use of source terms, data cross section data, shield and source geometries, and radiation transport calculational schemes, are consistent with generally accepted practice. We checked the shielding drawing presented by the applicant to ensure that the shield design is acceptable. Also, the shield design and construction will be consistent with guidance of Regulatory Guide 8.8 and Regulatory Guide 1.69, "Concrete Radiation Shields for Nuclear Power Plants."

The ventilation system is designed to assure that airflow will be from areas of low potential for airborne radioactivity to areas of higher potential and then to filters or vents. Also, the system will maintain concentrations of airborne radioactivity in normally occupied areas within the limits of 10 CFR Part 20. The ventilation filter trains are designed to allow exposures to be maintained "as low as is reasonably achievable" during servicing, consistent with the guidance of Regulatory Guide 8.8. Therefore, we conclude that the ventilation system radiation protection design features are acceptable.

Detectors for the area radiation monitoring system will be located in normally occupied areas which have the potential for radiation fields in excess of the maximum design radiation dose rate. The detectors are designed to cover the expected and maximum design dose rates and dose rates due to anticipated operational occurrences. The monitors will have readout and annunciation in the control room and variable alarm setpoints. The detectors will be source checked routinely and calibrated at refueling intervals. The system deviates from our acceptance criteria in that not all of the monitors will have local alarms. However, those monitors which are located to warn personnel occupying an area of a sudden high increase in dose rate will be equipped with local alarms. Other alarms will be announced by the control room personnel. Therefore, we conclude that the area radiation monitoring system design is acceptable.

The applicant has provided area radiation monitors around the fuel storage areas to meet the requirements of 10 CFR Part 70.24 and to be consistent with the guidance of Regulatory Guide 8.12, "Criticality Accident Alarm Systems."

The applicant will rely primarily on portable radiation monitoring instruments to assess the radiation hazard to personnel in areas which may be accessed during the course of an accident. The applicant does not consider the area radiation monitors to be part of the accident radiation monitoring system; therefore, they will not receive backup power from the diesel generators. The portable instruments will be placed to be readily accessible to personnel responding to an emergency, and will be designed with a sufficient instrument range for use in the event of an accident. We conclude that the accident radiation monitoring system is acceptable.

Our acceptance criterion in the Standard Review Plan for airborne radioactivity monitoring systems states that air should be sampled at normally occupied locations where airborne radioactivity may exist. The applicant has stated that there will be no area in the facility which has the potential for significant airborne radioactivity and which can be accessed by personnel without preaccess air monitoring. Significant airborne radioactivity would be levels on the order of a quarter of the maximum permissible concentrations in air specified in 10 CFR Part 20. Therefore, the applicant states that continuous airborne radioactivity monitoring is not required. The sources of airborne radioactivity are contained in cubicles which require airborne monitoring prior to and during access. The ventilation system is designed to prevent the migration of airborne radioactivity out of those cubicles, including unusual circumstances such as open doors. Surveys will be performed throughout the plant on a routine basis to ensure that the status of the plant airborne radioactivity is as expected. Also, air sampling systems will be used to detect unusual leakage from the sources of airborne radioactivity. The operating floor of the reactor building, including the spent fuel pool, does have the potential for significant airborne radioactivity. However, the applicant's experience at Quad Cities Station (a power reactor of very similar design which has been operated by the applicant for several years) indicates that significant airborne radioactivity only occurs in that area when work is being performed. Radiation protection inspection personnel from our Region III agreed that this has been the experience

at Quad Cities Station. Airborne monitoring will be provided whenever work is being performed in that area. Therefore, we conclude that the applicant will provide acceptable airborne radioactivity monitoring.

Based on the information provided, we conclude that the design of the radiation protection features are acceptable.

12.4 Dose Assessment

The applicant presented his estimate of the collective radiation dose equivalent workers will receive from the operation of the facility. The estimate is 650 man-rem per year for the station; this estimate includes the station staff and contractor personnel. The method used by the applicant is not consistent with the acceptance criteria in the Standard Review Plan. The applicant's assessment was not based on an analysis of the tasks involved in operation of the plant, the expected radiation dose rates, and the manpower required to perform those tasks. However, the applicant based the estimate on dose data from several years of operation at Quad Cities Station and other boiling water reactors. The design of Quad Cities Station is very similar to the design of La Salle, and Quad Cities Station is operated by the applicant. The applicant has carefully examined the operating experience at Quad Cities Station to determine those activities which contribute significantly to the collective occupational dose equivalent. The applicant then proceeded to examine possible design and procedure changes to reduce doses in those activities. Through this process the applicant has made several improvements in the La Salle design over the Quad Cities design to reduce doses. Examples of these improvements are a completely mechanized solid radwaste handling system and additional shielding around the spent fuel pool heat exchanger pumps. Also, as mentioned above, the applicant is examining the possibility of filtering the pumped-forward heater drains to reduce the radioactivity in the primary coolant system. Therefore, the method used by the applicant served the true purpose of a dose assessment, i.e., identification of instances where additional dose reduction features are justified. And in spite of the fact that the man-rem estimate was not made in a manner consistent with our acceptance criteria, we conclude that the applicant's dose assessment is acceptable.

The applicant has estimated the potential exposures of individual workers to airborne radioactivity in various parts of the station. These estimates are quite low; in most cases the estimates are only a few percent of the allowable exposures given in 10 CFR Part 20.103. Operating experience from boiling water reactors supports these estimates. Therefore, we conclude that the assessments of exposure to airborne radioactivity are acceptable.

The applicant has provided an estimate of 130 man-rems for the collective dose which the construction force will receive after Unit 1 begins operation. This estimate was based on the expected dose rates to which construction workers will be exposed and the construction manpower which will be expended on Unit 2 after Unit 1 begins operation. We have reviewed this assessment and conclude that the approach is reasonable. The estimate is comparable to estimates presented by other applicants with boiling water reactor designs. On these bases, we conclude that the construction worker dose assessment is acceptable.

Based on the information provided, we conclude that the dose assessment estimates are acceptable.

12.5 Health Physics Program

The applicant's organization will include health physics professionals and technicians. The Radiation/Chemistry Supervisor (Radiation Protection Manager) and the Health Physicist will have the responsibility for implementing the health physics program and maintaining exposures "as low as is reasonably achievable." Meetings and correspondence with the applicant indicate that La Salle's recently appointed Rad/Chem Supervisor does not meet the criteria of Regulatory Guide 1.8, "Personnel Selection and Training," for "applied radiation protection" experience. The applicant contends that the depth of La Salle's radiation protection organization outweighs what deficiencies may exist in the Radiation/~~Protection Manager~~ Chemistry Supervisor candidate's qualification. Specifically, the Health Physics Coordinator at La Salle, who reports directly to the Rad/Chem Supervisor, meets the requirements of Regulatory Guide 1.8 which references ANSI 18.1. This individual will assist the ~~Radiation Protection Manager~~ Rad/Chem Supervisor in the daily conduct of radiation protection activities and will serve as a backup to the Rad/Chem Supervisor in his absence. The Technical

Staff Supervisor at La Salle is also a qualified Radiation Protection Manager and can assist the Rad/Chem Supervisor if necessary. The applicant will hire a full time engineer to monitor the ALARA conformance at La Salle and participate in the development of the La Salle as low as is reasonably achievable (ALARA) program. In addition to site support, the applicant has an offsite technical liaison office (which includes individuals qualified to ANSI 18.1) which can provide health physics assistance and provides assurance that the onsite Radiation Protection Manager is aware of potential program deficiencies. Based on the support of these other health physics trained individuals onsite and the health physics support from corporate, we find the technical qualifications of the Health Physics organization at La Salle to be acceptable. The organizational aspects of the program are consistent with the guidance of Regulatory Guide 8.8.

12.5.1 Equipment, Instrumentation and Facilities

The applicant's radiation protection facilities will include portable instrument calibration and storage areas, personnel and equipment calibration and storage areas, personnel and equipment decontamination areas, change room, radiochemical laboratory, and a whole-body counting area. A variety of counting room and portable radiation detection and measurement instrumentation will be provided. The instrumentation will provide the capability to deal with all the types of ionizing radiation which may be encountered at the station. A variety of personnel monitoring devices and personnel protection equipment will also be provided. Protective clothing, respiratory protection devices, and personnel dosimeters will be included in the available equipment. However, the applicant has indicated that he does not intend to have personnel air samplers which can be worn on protective clothing; this is a deviation from our acceptance criteria. Although these devices can be useful in special circumstances, acceptable protection of personnel and assessment of exposure to airborne radioactivity can be accomplished without them. The health physics equipment, instrumentation and facilities are consistent with the guidance of Regulatory Guide 8.8 and, are, therefore, acceptable.

I.B Support Personnel

I.B.1.2 Organization and Management

Position

Corporate management of the utility-owner of a nuclear power plant shall be sufficiently involved in the operational phase activities, including plant modifications, to assure a continual understanding of plant conditions and safety considerations. Corporate management shall establish safety standards for the operation and maintenance of the nuclear power plant. To these ends, each utility-owner shall establish an organization, parts of which shall be located onsite, to: perform independent review and audits of plant activities; provide technical support to the plant staff for maintenance, modifications, operational problems, and operational analysis; and aid in the establishment of programmatic requirements for plant activities.

The licensee shall establish an integrated organizational arrangement to provide for the overall management of nuclear power plant operations. This organization shall provide for clear management control and effective lines of authority and communication between the organizational units involved in the management, technical support, and operation of the nuclear unit. The key characteristics of a typical organization arrangements are:

(1) Integration of all necessary functional responsibilities under a single responsible head.

(2) The assignment of responsibility for the safe operation of the nuclear power plant(s) to an upper-level executive position.

Utility management shall establish a group, independent of the plant staff, but assigned onsite, to perform independent reviews of plant operational activities. The main functions of this group will be to evaluate the technical adequacy of all procedures and changes important to the safe operation of the facility, and to provide continuing evaluation and assessment of the plant's operating experience and performance.

Discussion and Conclusion

As part of our evaluation of the organization and management improvements for operation of the La Salle Nuclear Station, an NRC team visited the applicant's corporate office of May 13 and 14, 1980, and the La Salle Station on September 9-11, 1980, where the team discussed the extent to which the applicant met the current versions of the draft document that has subsequently been published as NUREG-0731, "Guidelines for Utility Management Structure and Technical Resources - Draft Report for Interim Use and Comment." Our visit to the applicant's corporate office on May 13-14, 1980, was to evaluate the corporate management and offsite technical support in conjunction with a review of the Zion station. The results of this evaluation are documented in a June 23, 1980 letter to the applicant from R. F. Heishman, Region III, NRC Office of Inspection and Enforcement. The team found that:

(1) The applicant's corporate organization has been altered to strengthen the nuclear operations management under a senior, knowledgeable management official. This official, Byron Lee, holds the current title Executive Vice President and is responsible for all engineers, construction and operations of the companies' generating facilities, including fossil units. Reporting to Mr. Lee is Cordell Reed, Vice President of Nuclear Operation, who is responsible for the management of nuclear stations, station nuclear engineering, nuclear fuel services, environmental affairs, and nuclear licensing activities.

(2) The applicant has an active program directed at reducing operational error.

(3) A formal program for review of Licensee Event Reports from other than the applicant's plants was in place.

(4) The applicant has implemented an extensive offsite review function.

(5) The licensee made no commitments during our May 13 and 14 visit to meet the draft criteria regarding maximum overtime limits. (The licensee

subsequently, in Amendment 54 of the Final Safety Analysis Report, committed to meet the staff requirements for maximum overtime limits as discussed under Item I.A.1.3 of this report).

(6) The corporate management of Commonwealth Edison Company is sufficiently involved in matters affecting the Zion Station and other applicant's plants to assure a continual understanding of plant conditions and safety considerations. Frequent corporate level meetings are held to assure that corporate management is aware of the status of, and any problems that have developed at, the Zion Nuclear Station and other power plants. While there is not a documented procedure covering these meetings and formal meeting minutes are not maintained, these frequent management meetings appear to accomplish the functions of senior management oversight desired by us.

(7) The licensee's current offsite staff exceeds the minimum required staff qualifications and technical capabilities.

The review of the applicant's corporate management was performed, for the most part, by discussing and considering the extent to which it conformed with our paper entitled "Draft Criteria for Utility Management and Technical Competence" dated February 25, 1980. The required staff qualifications and technical capabilities alluded to in item (7) above are those listed in this draft paper. We expect that the applicant's corporate management will be similarly involved in matters affecting the La Salle Station.

During our September 9-11, 1980 site visit to La Salle, we held discussions with both La Salle's management and staff and with the applicant's corporate managers concerning the organization and management improvements both at La Salle and at the corporate office. We interviewed a number of station personnel including the Plant Superintendent, the Operating Assistant Superintendent, the Technical Staff Supervisor, an Operating Engineer, a Shift Engineer, and the Training Supervisor. We also held discussions with the Vice President, Nuclear Stations and the Director of Nuclear Safety, who came from the corporate office to meet with us.

Based on discussions with the station personnel, we conclude that there will be a system in place at La Salle prior to fuel load that will assure that information concerning important safety matters and operational experiences at other nuclear plants as well as at La Salle will be disseminated to the operating personnel who need it. We also determined that appropriate direct oral communication channels exist between the plant management staff and the corporate office. We also observed, based on watching a taped training session in which corporate managers instructed the plant operators the importance of taking safe, conservative actions whenever a question arises concerning Technical Specification operation limits, that the applicant's corporate management is concerned with safe operation of its stations and is taking action to enhance it.

We learned in our discussions at the site that (1) the Division Manager, Nuclear Stations title has been changed to Vice President, Nuclear Stations, and (2) the applicant has established a new organization in its corporate office called the Nuclear Safety Department. R. Jortberg is the Director of this new department, and he reports directly to the Chairman and President and receives day-by-day functional direction from the Vice President, Nuclear Operations.

In Amendment 54 to the Final Safety Analysis Report, the applicant indicates that reporting directly to the new Director of Nuclear Safety are a Supervisor of Off-Site Review and a Supervisor of Safety Engineering Groups. The independent onsite safety engineering groups, that are required by Item I.B.1.2 to be located at each station, report to this Supervisor of Safety Engineering Groups.

This group assigned to La Salle is called the Safety Engineering Group-La Salle Station. It will consist of four dedicated full-time engineers. These four full-time personnel will be augmented on a part-time basis by personnel from other parts of the applicant's organization to provide expertise in disciplines not represented within the onsite group. The applicant has stated that personnel assigned to the group shall meet the qualification requirements described in Section 4.7 of Draft ANSI/ANS 3.1-1979. We find this approach to the independent

engineering review and evaluation effort to be acceptable at this time. However, we intend to review this activity at La Salle in about a year, as we plan to do at other recently licensed plants (for which this requirement was a licensed condition), to assure that the onsite group is functioning properly and to determine if some changes are needed to make it more effective. We will include requirements for the functioning of this onsite group in La Salle's Technical Specifications.

In addition, in order to clarify some concerns that we had on the Health Physics organization at La Salle, we performed a site visit on January 26 and 27, 1981. As a result of the visit, we resolved the following issues:

(1) In the Commonwealth Edison's Quality Assurance Topical Report dated December 29, 1980, the applicant has revised the Rad/Chem Supervisor's reporting requirements to comply with the criteria of Item I.B.1.2. The new Quality Assurance Manual provides the Rad/Chem Supervisor with direct recourse to the Superintendent in order to resolve questions related to the conduct of the Radiation Protection Program. This is acceptable to us.

(2) The radiation protection section and chemistry section are combined as one section at La Salle. Health Physics Appraisal findings from other applicant's plants having similar health physics/chemistry structures have shown that weaknesses do exist in this type of joint organization. In order to resolve these weaknesses and assure proper operation of its radiation protection organization, the applicant has commissioned a consultant to perform a management assessment of its organizational structure. The applicant has committed to implement appropriate changes, as necessary, based on the final recommendations of this study, to assure the proper functioning of the radiation protection program at La Salle, and other applicant's plants. These changes will be implemented at La Salle. We find the review of La Salle health physics organization an acceptable approach to resolution of our concerns in this area. We find the La Salle health physics/chemistry organization acceptable subject to implementation of the organization study recommendations at all Commonwealth Edison stations.

On the basis of our review as discussed above, we conclude that the applicant's organization and management improvements related to TMI lessons learned are substantial and provide reasonable assurance that appropriate and due concern for safety will be exercised in the operation of La Salle and therefore the applicant complies with the requirements of Item I.B.1.2.

II.B.2 Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation

Position

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors" and Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors" (i.e., the equivalent of 50 percent of the core radioiodine, 100 percent of the core noble gas inventory, and 1 percent 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 postaccident operations of these systems.

Each licensee shall provide for adequate access to vital areas of 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

The purpose of this item is to ensure that licensees examine their plants to determine what actions can be taken over the short-term to reduce radiation levels and increase the capability of operators to control and mitigate the consequences of an accident. The actions should be taken pending conclusions resulting in the long-term degraded core rulemaking, which may result in a need to consider additional sources.

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. For the purposes of this evaluation, vital areas and equipment are not necessarily the same vital areas or equipment defined in 10 CFR Part 73.2 for security purposes. The security center is listed as an area to be considered as potentially vital, since access to this area may be necessary to take action to give access to other areas in the plant.

The control room, technical support center (TSC), sampling station, and sample analysis area must be included among those areas where access is considered vital after an accident. (Refer to Section III.A.1.2 of this report for discussion of the TSC and emergency operations facility.) The evaluation to determine the necessary vital areas should also include, but not be limited to, consideration of the post-loss-of-coolant accident hydrogen control system, containment isolation reset control area, manual emergency core cooling system alignment area (if any), motor control centers, instrument panels, emergency power supplies, security center, and radwaste control panels. Dose rate determinations need not be for these areas if they are determined not to be vital.

As a minimum, necessary modification must be sufficient to provide for vital system operation and for occupancy of the control room, TSC, sampling station, and sample analysis area.

In order to assure that personnel can perform necessary post-accident operations in the vital areas, the following guidance is 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, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and Standard Review Plan 15.6.5 with appropriate decay times based on plant design (i.e., assuming the radioactive decay that occurs before fission products can be transported to various systems).

(a) Liquid-Containing Systems: 100 percent of the core equilibrium noble gas inventory, 50 percent of the core equilibrium halogen inventory, and 1 percent of all others are assumed to be mixed in the reactor coolant and liquids recirculated by residual heat removal, high-pressure coolant injection, and low-pressure coolant injection, or the equivalent of these systems. In determining the source term for recirculated, depressurized cooling water, assuming that the water contains no noble gases.

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

(2) Systems Containing the Source

Systems assumed in your analysis to contain high levels of radioactivity in a post-accident situation should include, but not be limited to, containment, residual heat removal system, safety injection systems, chemical and volume control system, containment spray recirculation system, sample lines, gaseous radwaste systems, and standby gas treatment systems (or equivalent of these systems). If any of these systems or others that could contain high levels of radioactivity were excluded, you should explain why such systems were excluded. Radiation from leakage of systems located outside of containment need not be considered for this analysis. Leakage measurement and reduction is treated under Section III.D.1.1, "Integrity of Systems Outside Containment Likely to Contain Radioactive Material for PWRs and BWRs." Liquid waste systems need not be included in this analysis. Modifications to liquid waste systems will be considered after completion of Section III.D.1.4, "Radwaste System Design Features To Aid in Accident Recovery and Decontamination."

(3) Dose Rate Criteria

The design dose rate for personnel in a vital area should be such that the guidelines of Criterion 19 of the General Design Criteria will not be exceeded during the course of the accident. GDC 19 requires that adequate radiation protection be provided such that the dose to personnel should not be in excess of 5 rem whole body, or its equivalent to any part of the body for the duration of the accident. When determining the dose to an operator, care must be taken to determine the necessary occupancy times 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. 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: <15 mrem/hr (averaged over 30 days). 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 Standard Review Plan 6.4.

(b) Areas Requiring Infrequent Access: Criterion 19 of General Design Criteria. These areas may require access on an irregular basis, 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 of sites where occupancy may be needed often, but not continuously.

(4) Radiation Qualification of Safety-Related Equipment

The review of safety-related equipment which may be unduly degraded by radiation during post-accident operation of this equipment relates to equipment inside and outside of the primary containment. Radiation source terms calculated to determine environmental qualification of safety-related equipment consider the following:

(a) Loss-of-coolant accident (LOCA) events which completely depressurize the primary system should consider releases of the source term (100 percent noble gases, 50 percent iodines, and 1 percent particulates) to the containment atmosphere.

(b) LOCA events in which the primary system may not depressurize should consider the source term (100 percent noble gases, 50 percent iodines, and 1 percent particulates) to remain in the primary coolant. This method is used to determine the qualification doses for equipment in close proximity to recirculating fluid systems inside and outside of containment. Non-LOCA events both inside and outside of containment should use 10 percent noble gases, 10 percent iodines, and 0 percent particulate as a source term. The following table summarizes these considerations:

Containment	LOCA Source Term (Noble Gas/Iodine/ Particulate)	Non-LOCA High-Energy Line Break Source Term (Noble Gas/Iodine/Particulate)
Outside	Percent (100/50/1) in reactor coolant system	Percent (10/10/0) in reactor coolant system
Inside	Larger of (100/50/1) in containment or (100/50/1) in reactor coolant system	(10/10/0) In ~~RCS~~ reactor coolant system

Discussion and Conclusions

The La Salle radiation and shielding review utilized our prescribed post-accident source terms described in Regulatory Guides 1.3 and 1.7, as specified in NUREG-0737. Using these source terms, the applicant derived spacial-and-time-dependent dose rate maps for La Salle which were then used to calculate post-accident integrated doses to vital areas. The vital areas (areas critical for plant shutdown) at La Salle have dose rates which will allow continuous occupancy following an accident. Access between vital areas will be controlled in accordance with the Health Physics Program. The applicant has provided a listing of post-accident paths. Areas designated as "vital areas" at La Salle are the control room, the auxiliary electric equipment room, where the remote shutdown panels are located, the Technical Support Center, and the sampling stations. The integrated dose to the first three areas for the duration of an accident will be less than 5 rem whole body, as per Criterion 19 of the General Design Criteria. The integrated dose to individuals performing sampling operations in the sampling stations will be less than 3 rems whole body or 18-3/4 rems to the extremities, as per Criterion 19 of the General Design Criteria.

Areas of highly restricted access following an accident at La Salle include the entire reactor building, emergency core cooling system equipment located outside primary containment, and the area in the vicinity of the standby gas treatment system charcoal filter. Restricted access areas following an accident include the post-accident sampling system in the heating, ventilation, and air-conditioning system and waste tank rooms, and the east end of the radwaste tunnel. Other areas may be designated as restricted areas, depending on the severity of the accident. Access to all restricted access areas will be controlled in accordance with the Health Physics Program. The post-accident sampling room will be an area of extended occupancy. Its design will permit an operator to take the first set of post-accident samples while limiting his integrated dose to 1 rem (which is within the exposure limits set by 10 CFR Part 20). Shielding and other modifications made as a result of the shielding design review include the following. The main stack monitoring panel will transfer its monitoring functions to the well-shielded high-range stack monitor

when it measures a prescribed radiation level. The applicant has added shield doors and additional shielding walls to the reactor building in order to reduce radiation streaming and improve access to various plant areas (including the facilities adjacent to the control room) following an accident.

The applicant has performed a design review of plant radiation and shielding for post-accident operations. This review has shown that La Salle meets the post-accident shielding requirements. We therefore find La Salle response to Item II.B.2 acceptable.

ATTACHMENT 3, Containment High-Range Radiation Monitor

Position

In-containment radiation-level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be developed and qualified to function in an accident environment.

Clarification

(1) Provide two radiation monitor systems in containment which are documented to meet the requirements of Table II.F.1-4.

(2) The specification of 10^8 rad/hr in the above position was based on a calculation of post-accident containment radiation levels that included both particulate (beta) and photon (gamma) radiation. A radiation detector that responds to both beta and gamma radiation cannot be qualified to post-LOCA (loss-of-coolant accident) containment environments but gamma-sensitive instruments can be so qualified. In order to follow the course of an accident, a containment monitor that measures only gamma radiation is adequate. The requirement was revised in the October 30, 1979 letter to provide for a photon-only measurement with an upper range of 10^7 R/hr.

(3) The monitors shall be located in containment(s) in a manner as to provide a reasonable assessment of area radiation conditions inside containment. The monitors shall be widely separated so as to provide independent measurements and shall "view" a large fraction of the containment volume. Monitors should not be placed in areas which are protected by massive shielding and should be reasonably accessible for replacement, maintenance, or calibration. Placement high in a reactor building dome is not recommended because of potential maintenance difficulties.

(4) For BWR Mark III containments, two such monitoring systems should be inside both the primary containment (drywell) and the secondary containment.

III.D.3.3 Improved In-Plant Iodine Instrumentation Under Accident Conditions

Position

(1) Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

(2) Each applicant for a fuel loading license to be issued prior to January 1, 1981 shall provide the equipment, training, and procedures necessary to accurately determine the presence of airborne radioiodine in areas within the plant where plant personnel may be present during an accident.

Clarification

Effective monitoring of increasing iodine levels in the buildings under accident conditions must include the use of portable instruments using sample media that will collect iodine selectively over xenon (e.g., silver zeolite) for the following reasons:

(1) The physical size of the auxiliary and/or fuel handling building precludes locating stationary monitoring instrumentation at all areas where airborne iodine concentration data might be required.

(2) Unanticipated isolated "hot spots" may occur in locations where no stationary monitoring instrumentation is located.

(3) Unexpectedly high background radiation levels near stationary monitoring instrumentation after an accident may interfere with filter radiation readings.

(4) The time required to retrieve samples after an accident may result in high personnel exposures if these filters are located in high-dose-rate areas.

After January 1, 1981, each applicant and licensee shall have the capability to remove the sampling cartridge to a low-background, low-contamination area for further analysis. Normally, counting rooms in auxiliary buildings will not have sufficiently low backgrounds for such analyses following an accident. In the low background area, the sample should first be purged of any entrapped noble gases using nitrogen gas or clean air free of noble gases. The licensee shall have the capability to measure accurately the iodine concentrations present on these samples under accident conditions. There should be sufficient samplers to sample all vital areas.

For applicants with fuel loading dates prior to January 1, 1981, provide by fuel loading (until January 1, 1981) the capability to accurately detect the presence of iodine in the region of interest following an accident. This can be accomplished by using a portable or cart-mounted iodine sampler with attached single-channel analyzer (SCA). The SCA window should be calibrated to the 365 KeV of iodine-131 using the SCA. This will give an initial conservative estimate of presence of iodine and can be used to determine if respiratory protection is required. Care must be taken to assure that the counting system is not saturated as a result of too much activity collected on the sampling cartridge.

Discussion and Conclusion

The applicant will have the capability of measuring post-accident concentrations of radioiodine in the control room and other areas of the plant. To perform this measurement, the applicant will have five Eberline Instrument Corporation PING-3 (2A special) particulate, iodine, and noble gas air monitoring systems for air sampling plant areas for the presence of radioiodine. These systems are cart mounted with battery-powered backup. Radioiodine analysis will also be provided using an Eberline Instrument Corporation SAM-2 iodine monitoring system. These systems will be used with silver zeolite cartridges (approximately 100 of which will be stored onsite) during accident conditions.

La Salle has station procedures for obtaining and evaluating routine and nonroutine air samples. In addition, initial training and periodic drills are

conducted for the radiation/chemistry personnel in the use of these monitors following an accident. The applicant will analyze iodine cartridges in a low background, low contamination area following an accident. The lower storeroom elevation of the service building or the radwaste control room were two areas identified for this purpose. Iodine cartridges will be purged of any entrapped noble gases using station service or bottle nitrogen prior to analysis.

The applicant has adequately addressed the criteria of Item III.D.3.3 and his response meets the positions set forth in our requirements.