DISTRIBUTION:*w/encl.
[redacted]

LB#1 Rdg	bcc:
DEisenhut	TERA
JYoungblood	PDR*
LKintner	LPDR*
MRushbrook	NSIC
RTedesco	TIC
SHanauer	ACRS (16)
RVollmer	
TMurley	
RMattson	
RHartfield, MPA	
OELD*	
OIE(3)	

MAY 5 1981

Docket No.: 50-341

Mr. Harry Tauber
Vice President
Engineering & Construction
Detroit Edison Company
2000 Second Avenue
Detroit, Michigan 48226

Dear Mr. Tauber:

Subject: Safety Concerns Associated with Pipe Breaks in the BWR Scram System

On April 23, 1981, we discussed with your representatives the NRC's Office of Analysis and Evaluation of Operational Data (AEOD) report entitled, "Safety Concerns Associated with Pipe Break in the BWR Scram System." The Report describes a potential sequence of events which could result from a break in the BWR scram discharge piping during a scram condition concurrent with an inability to reclose the scram outlet valves. Concerns identified include the quality of the scram discharge volume piping, the ability to detect and isolate such a break, and potential water and steam degradation of available ECCS equipment as a result of the break. A number of recommendations were made in the report to remedy the potential concerns.

We are presently studying these issues and recommendations to determine whether the BWR design basis accidents should be modified and as a consequence whether appropriate actions should be taken for operating BWR plants. The purpose of this letter is to provide to you the AEOD report so that you can evaluate its applicability to your plant and determine appropriate remedial measures, and to request information from you concerning your evaluation in order to assist in determining an appropriate course of action for the NRC.

Therefore, please provide us within 45 days of your receipt of this letter, the following information:

1. A generic evaluation of the applicability of the indicated sequences of events in the REPORT to the BWR plant design, your estimate of the probability of occurrence of such sequences, and the bases for these conclusions,

2. A generic evaluation of the applicability of the indicated safety concerns and findings in the REPORT relative to BWR plant construction, design, and operation and the bases for these conclusions, and

RECEIVED
MAY 07 1981
U.S. NUCLEAR REGULATORY COMMISSION

8105120661

A

OFFICE							
SURNAME							
DATE							

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO 1980—329-824

3. A generic evaluation of the recommendations listed in the REPORT discussing the degree of which the recommendations are being or have been implemented with bases why the recommendations should or should not be completely implemented on BWRs.

In addition, provide the following information within 120 days of your receipt of this letter:

1. Provide an evaluation of the applicability of the 45 day generic evaluation to your plant. This evaluation should contain plant specific considerations related to system design, instrumentation, construction, operation, operator training, and emergency procedures for your plant.

2. In light of the AEOD report and the 45 day generic evaluation, provide a plant specific evaluation of your facility's Scram Discharge Volume System conformance to GDC 14, GDC 35, GDC 55, §50.2(v), 50.55a (including footnote 2), and §50.46 of the Commission's regulations. This evaluation should address which portions of the Scram Discharge Volume System are considered to be part of the reactor coolant pressure boundary, the quality group and safety class of the Scram Discharge Volume System, the codes and standards used for the design, fabrication and inservice inspection of this system, and your bases for the above classifications or groupings.

3. Provide by analysis or reference a demonstration that a break in the Scram Discharge Volume System meets the requirements of §50.46 of the Commission's regulations, taking into account the environmental and flooding aspects of such a break.

Sincerely,

Original signed by
Robert L. Tedesco
Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

Enclosure:
As stated

cc w/encl.: See next page

OFFICE	DL:LB#1	DL:LB#1	DL:AD/L				
SURNAME	LKintner/ls	BJYoungblood	RLTedesco				
DATE	5/2/81	5/7/81	5/4/81				

Mr. Harry Tauber
Vice President
Engineering & Construction
Detroit Edison Company
2000 Second Avenue
Detroit, Michigan 48226

cc: Eugene B. Thomas, Jr., Esq.
LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N. W.
Washington, D. C. 20036

Peter A. Marquardt, Esq.
Co-Counsel
The Detroit Edison Company
2000 Second Avenue
Detroit, Michigan 48226

Mr. William J. Fahrner
Project Manager - Fermi 2
The Detroit Edison Company
2000 Second Avenue
Detroit, Michigan 48226

Mr. Larry E. Schuerman
Detroit Edison Company
3331 West Big Beaver Road
Troy, Michigan 48084

David E. Howell, Esq.
21916 John R
Hazel Park, Michigan 48030

Mr. Bruce Little
U. S. Nuclear Regulatory Commission
Resident Inspector's Office
6450 W. Dixie Highway
Newport, Michigan 48166

Dr. Wayne Jens
Detroit Edison Company
2000 Second Avenue
Detroit, Michigan 48226

NUREG-0785
DRAFT

SAFETY CONCERNS
ASSOCIATED WITH PIPE BREAKS
IN THE
BWR SCRAM SYSTEM

by the

OFFICE FOR ANALYSIS AND EVALUATION

OF OPERATIONAL DATA

March 1981

Prepared by: Stuart D. Rubin
Lead Reactor
Systems Engineer

dupe of 8104240270 35pp

NOTE: This report documents results of studies performed by the Office for Analysis and Evaluation of Operational Data. The findings and recommendations contained in this report are provided in support of other ongoing NRC activities and do not represent the position or requirements of the responsible program offices of the Nuclear Regulatory Commission.

TABLE OF CONTENTS

List of Figures

Appendices

EXECUTIVE SUMMARY

Since the Browns Ferry 3 (BF-3) partial failure to scram of June 28, 1980, the scram discharge volume (SDV) subsystem of the BWR scram system has been extensively studied with respect to failure conditions which may cause a loss of scram capability or its protective function. At the same time, while the SDV system has reactor pressure boundary and primary containment boundary functions, little if any review effort has been expended to study the safety concerns associated with postulated pipe break failures within the SDV subsystem. Prompted by the serious and fundamental findings of deficiency, documented in our original BF-3 event case study investigation, AEOD undertook a more thorough safety review of the adequacy of the scram system design with regard to the reactor coolant boundary and primary containment functions. As a result of this further work, important additional issues and safety concerns have been raised with respect to isolation capabilities of the scram system and operation of the emergency core cooling systems for SDV pipe break situations.

We have found that, in the event of a SDV system pipe break attendant to a reactor scram, termination of the resultant reactor coolant blowdown outside primary containment would be dependent on successful closure of non-redundant (scram outlet) valves. The closure principle and design arrangement of these valves do not meet the important requirements for isolation valves described in GDC 54 and 55 of Appendix A to 10 CFR 50. Furthermore, while the break isolation involves a man-machine system, we have found that potentially less than adequate human factor preparation has been provided, given the importance to safety of isolating a break in the SDV system. Additionally, in the event that break isolation is not achieved, the current plant emergency operating procedures do not adequately address the potentially concurrent need for maintaining the core covered and protecting against the loss of ECCS equipment due to adverse environmental conditions including flooding.

We have found that failure to isolate a SDV system pipe break raises serious concerns regarding the assurance of long-term decay heat removal with emergency core cooling systems since the break itself potentially threatens operation of this equipment. At the same time, information found from our investigation for the mechanical integrity assurance basis of the SDV system piping indicates that the present level of assurance may not be commensurate with the risks associated with an accidental rupture of this piping.

In view of the deficiencies found and issues raised, we have recommended several corrective actions which should substantially reduce, although not eliminate, the perceived risks associated with a break in the SDV system piping attendant to a reactor scram.

In view of these perceived risks, we recommend that the regulatory need to postulate such pipe breaks as part of the BWR design basis be determined and standardized. To this end, we would recommend that a two-phase action plan be initiated. The first phase should immediately address and correct the presently inadequate mechanical integrity assurance basis of the SDV system components for operating BWRs. The second phase should incorporate a high priority safety issue review which will address the need to consider such breaks in the design basis and will develop and implement the needed corrective actions on a plant-by-plant basis if it is determined that SDV system breaks are to be included in the plant design basis.

1. INTRODUCTION

Immediately after the Browns Ferry partial failure to scram of June 28, 1980, the Office for Analysis and Evaluation of Operational Data (AEOD) initiated an independent investigation of the event, including the Browns Ferry 3 scram system design, operation and operating characteristics. The principal focus of this investigation centered on the Browns Ferry 3 (BF-3) scram discharge volume (SDV) system, including its hydraulic operating characteristics important to reactor scram capability and its protective function. The report which documented this review also touched upon the reactor coolant boundary isolation function of the SDV system. As a result of our independent investigation, AEOD identified several important deficiencies in the system design and hydraulic characteristics which related principally to the SDV system scram capability and protective functions. The serious and fundamental nature of these findings made it apparent to AEOD that less than an adequate system design review and regulatory safety review had been made when the SDV system design was originally developed and proposed for use in operating BWRs. Because of this perception, AEOD made the decision to extend its initial analysis and evaluation of the BF-3 scram system to include a more thorough safety assessment of the reactor coolant boundary and primary containment functions of the SDV system and its appendages.

In the case study report[(1)] for the Browns Ferry 3 partial failure to scram event, we addressed deficiencies in the isolation capabilities of the BWR scram discharge volume system. We found that during a reactor scram a single active failure (to close) of an SDV system vent valve or drain valve would result in a blowdown of the reactor coolant system (RCS) outside primary containment. For this event, the RCS blowdown could be terminated only if all of the scram discharge valves could be reclosed. This is normally

accomplished from the control room by manually resetting the reactor protection system (RPS). However, as described in the BF-3 case study report and further expanded in this report, reclosure of the scram outlet valves may not always be possible. For example, many BWR reactor trip conditions do not readily clear or cannot be bypassed in either the SHUTDOWN or REFUELING mode. These are among many conditions that would normally prevent RPS reset. Thus, a sustained trip condition following a scram, such as caused by closure of the MSIVs, would normally prevent isolation of an RCS blowdown through a stuck open vent or drain valve. Thus it was noted in our report that closure of the scram outlet valves via RPS reset would be blocked by the trip condition itself (which cannot be bypassed in either the SHUTDOWN or REFUELING mode).

Since the time of our case study investigation of the BF-3 event and its cause, we have extended our review to include an assessment of safety concerns associated with single passive failures (i.e., pipe breaks) in the SDV system. It is postulated that attendant to a reactor scram a break may occur in the SDV system piping downstream of the scram outlet valves and upstream of the SDV system vent or drain valves. For this break location automatic closure of the vent or drain line isolation valves will not terminate the RCS blowdown since these valves are located downstream of the break location. In such an event, closure of all scram outlet valves would be the only available option to prevent an uncontrolled RCS blowdown outside primary containment.

2. DISCUSSION OF SAFETY CONCERNS

2.1 Break Location

When a BWR is not in a scrammed state, the scram valves are held closed by control air pressure and reactor coolant is retained on the upstream side of the closed valves. In this state, the scram valves perform reactor coolant boundary (RCB) and primary containment isolation (PCI) functions. Downstream of the closed scram outlet valves, the SDV headers are continuously drained (empty), unpressurized (open) and isolated from the RCS. The SDV headers in this state provide a scram capability function in that they provide the required free volume for the reactor water exhausted during a scram. Upon a reactor scram, the scram outlet valves open, the SDV drain and vent valves close and the SDV system piping fills and pressurizes as it accepts, contains, and limits the water exhausted from the reactor through the control rod drives (CRDs). Even after the control rods have fully inserted, (with the scram valves left open), reactor coolant continues to flow past the CRD seals, through the scram outlet valves and into the SDV system piping pressurizing it to full reactor pressure. Therefore, during and immediately following a scram the SDV system becomes the reactor coolant retaining boundary well outside of primary containment. After completion of a scram, therefore, the SDV system having fulfilled its scram capability function, assumes a reactor coolant boundary function and a primary containment isolation function. It is during this fully pressurized state of the SDV system that we have examined the potential safety concerns associated with a break in the SDV system piping. The pipe break is postulated to be a high energy break in any size line in the system and initiated by the pressure, temperature and other loadings attendant to the reactor scram but not, necessarily, considered in the mechanical design basis of the SDV system.

2.2 Break Isolation

From a system's viewpoint, the blowdown of the postulated break into the reactor building (secondary containment where the SDV system piping is located) could be terminated via manual control room operator action by initiating group closure of the scram outlet valves. This action requires the ability to manually reset the RPS (which requires RPS power and an absence of trip conditions) and the availability of control air supply. However, group closure of the scram outlet valves has not heretofore been defined as a required safety function. Accordingly, the systems (including control air supply) upon which operation of the scram outlet valves is dependent have not been designed to assure reliable closure of these valves. Thus, isolation of a postulated break in the SDV portion of the RCB which lies outside primary containment and downstream of the hydraulic control units (HCUs) cannot presently be reliably assured, at least to the degree inherent in other RCB pipes incorporating qualified isolation valve designs and arrangements. Although the scram outlet valves incorporate a relatively leak resistant design, there are numerous disabling conditions consequential to the trip condition or pipe break, as well as numerous disabling single failures in the RPS and control air systems, which could temporarily or permanently prevent successful reclosure of these valves following a scram. For example, such conditions as (1) a loss of control air pressure for any reason, (2) a trip condition which cannot be bypassed in either the SHUTDOWN or REFUELING mode or (3) a total loss of RPS power supply would prevent group reclosure of the scram outlet valves.

Also, unlike qualified RCB or PCI isolation valves, the scram outlet valves do not incorporate an automatic closure feature. The absence of an auto closure feature is clearly necessitated by the need for a reliable scram function which must not be automatically overridden under any circumstances.

The net effect is that scram valve group closure is a manual operation which must be remotely actuated by the operator from one of the control room consoles. Even under such circumstances, closure is precluded by a time delay relay for a minimum of ten seconds. This is to prevent the control room operator from interfering with, or prematurely terminating scram insertion of control rods. Thus, isolation of a break in the SDV system piping with the current design of the scram valve closure apparatus of necessity involves the human factor; that is, the isolation system for a postulated break in the SDV system piping can be characterized as a "man-machine" system.

A review of the "man" side of the man-machine SDV break isolation arrangement indicates potentially less than adequate human-factor preparation. There are no qualified SDV system break detection instruments for the operator to rely upon to quickly identify the presence of a break in the SDV system piping. Typically, BWRs like Browns Ferry-3 have reactor building radiation monitors located in the CRD-HCU areas. However, their operability and calibration are not presently included in plant Technical Specification requirements as are other radiation monitoring instruments in the plant. Additionally, depending on the sensor positions and their sensitivity, these instruments may annunciate for every reactor scram, regardless of whether a break were present or not. Furthermore, the control room operator has not been provided with special emergency operating procedures or training to quickly and appropriately respond to SDV system pipe break symptoms which would accompany normal post-reactor trip control room indications and activities. Additionally, should immediate reclosure of the scram valves not be possible there are no emergency operating procedures or operator training provided to aid the operator in diagnosing and correcting the source of failure in attaining RPS reset and/or recovering from a loss of control air supply. Continued blowdown of hot reactor water past the scram valves may also degrade and eventually disintegrate their teflon seating surface which could eventually eliminate the primary means of break isolation.

A local manual isolation valve is provided in series with each remote air-operated scram outlet valve on each HCU. However, dispatching an auxiliary operator to enter the reactor building to manually close each of these valves would be extremely unlikely, given the harsh environmental conditions including hot water blowdown, high radiation and possible loss of lighting or visibility in the area of the reactor building where the postulated break is located. Therefore, for both equipment-related and procedural-related reasons, isolation of a break in the SDV system attendant to a reactor scram may not be reliably assured.

2.3 Break Discharge Conditions

One should expect that failure to close the remote air-operated scram outlet valves or the local manual isolation valves would result in a considerable blowdown rate out of the reactor coolant system directly into the reactor building secondary containment. The blowdown rate would be limited only by either the combined control rod drive seal leakage from all drives manifolded by the SDV headers (via the 3/4 inch Schedule 80 scram exhaust risers on each drive) or by the postulated SDV system pipe break size and location. Currently, there is no Technical Specification limit for CRD seal leakage rate. However, seal leak rate (stall) testing at the BF-3 site after the June 28, 1980 control rod insertion failure indicated that the average CRD seal leak rate (with approximately 250 psi pressure differential across the seals) could be about a 3 gpm per drive. Furthermore, the General Electric Company technical manual[(2)] used for CRD operation, maintenance and testing recommends that seals be rebuilt when seal leakage exceeds 5 gpm. Thus, for 185 CRDs initial cumulative seal leakage could be anywhere from about 550 gpm to 900 gpm assuming a 250 psi pressure differential across the seals. Continued blowdown of hot reactor water through the CRDs would likely

degrade the CRD seals as a result of flashing and cavitation and seal heat-up caused by hot pressurized water flowing past the seals. (This effect might be similar to reactor coolant pump seal degradation following a loss of seal cooling injection flow.) Thus, the CRD blowdown rate, as initially limited by intact seals, might be expected to increase with time from the magnitudes cited above. Reactor system pressure, CRD seal condition, the actual differential pressure across the seals, line losses and the break size/location in the SDV piping system, would ultimately set the blowdown rate in the long term.

2.4 Potential Core Consequences

The anticipated cumulative seal leakage would be expected to be well within the makeup capacity of the high pressure coolant injection (HPCI) system or possibly the reactor core isolation cooling (RCIC) system. If the HPCI system was unavailable, the automatic depressurization system (ADS) in conjunction with either of the core spray (CS) systems or the low pressure coolant injection (LPCI) subsystem of the residual heat removal (RHR) system could provide ample alternate makeup. Thus, as far as peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, and coolable geometry criteria are concerned , an unisolated break in the SDV system may not be of concern during the initial mitigation phases of the event. It is, however, with respect to the continued long-term core cooling requirements and the availability of emergency makeup systems over the long term, that such an unisolated break provides unique ECCS challenges and uncertainties. Thus, it is with respect to long-term decay heat removal and maintaining the core covered that potentially serious public health and safety questions arise.

A break in the SDV system without isolation is equivalent to a small unisolated break in the bottom of the reactor vessel. For this case, the core shroud

and jet pump diffuser nozzles cannot provide their usual protection against a relatively rapid coolant loss and level drop above the core attendant to a temporary loss of makeup supply. This is unlike the case for even the largest postulated break in a recirculation line. Furthermore, even primary containment flooding (assuming water supply and pumps were available) would not assure long-term core coverage since the break would essentially be in the bottom of the vessel but located outside the primary containment structure. Accordingly, a source of makeup water and adequate pumping capability must be maintained available indefinitely or until such time that some means of break isolation can be provided. However, because of the unique location of this unisolated break, long term cooling may not be assured.

For an unisolated break in the SDV system, reactor coolant would continue to be lost out the reactor system without accumulating in the drywell-torus which is the normal reservoir for water for long term cooling. Reactor water discharged directly into the reactor building would collect on the floor and be carried down through the open floor drains and other open passageways of the reactor building to the basement of the building. Once there, it would collect in the dirty radwaste (DRW) sumps located in the reactor building basement corner rooms. Water collected there would normally be pumped out of the secondary containment by two small capacity, (50 gpm) sump pumps and enter the DRW liquid waste collection system tanks. This water lost from the reactor would not normally be suitable or available for return to the reactor.

2.5 Potential Consequences to the Mitigation Systems

The reactor building layout for BF-3 incorporates large stairwell openings (identified by circles in Figure 2-1) in three of the four corners of the

565-foot elevation, where the SDV headers are located. The stair steps are open-lattice metal gratings which would permit hot water to cascade directly down to the basement floor. There are no curbs at the stairwell entrances. Any water not removed by the floor drains on the 565-feet elevation floor will run over to the stairwells and flow directly into the basement. Located in the basement at these corners (see circles in Figure 2-2) are the RHR system pumps and the CS system pumps. Thus these low pressure makeup systems might be quickly disabled by the effects of water cascading into the corner rooms and by the flashing of hot water. In this way, a break in the SDV system could result in the loss of most if not all of the low pressure emergency core cooling pumps shortly after the break occurred. Qualification of this equipment for operation under such environmental conditions clearly would be questionable. Additionally, the RCIC pump is located in the same room with one train of the CS pumps and the HPCI pump is located in a room which is adjacent to one train of the RHR pumps and would, therefore, also be subject to severe environmental conditions including flooding. The control rod drive pumps are located on a platform above one train of the CS pumps and would be similarly involved in the adverse environmental conditions. The fourth corner of the reactor building basement contains an elevator shaft instead of a stairwell which should provide temporary protection against immediate damage to one train of the residual heat removal system, although the environment would degrade quickly.

If break isolation is not successful, the blowdown rate into the reactor building (which could be in excess of 1,000 gpm) would substantially exceed the total capacity of the sump pumps (which is approximately 100 gpm). Even if the sump pumps initially were capable of removing the reactor water being collected in the sumps, assurance of continued water removal from the sump

cannot be provided indefinitely for continued SDV system blowdown. An unarrested blowdown would eventually challenge the operability of the sump pumps and their electrical circuits with environmental conditions for which they were not designed. For example, for BF-3 the sump pumps are powered by the 3C 480V reactor building MOV boards which are immediately adjacent to the HCUs on the 565 feet elevation. Furthermore, these pumps and their power supplies would not be readily accessible by maintenance personnel given the harsh environmental conditions in the reactor building. The pumps are not supplied with emergency onsite power.

Thus it appears likely that all of the ECCS pumps in the basement would eventually be lost by flooding if the break were not isolated. Clearly, the unavailability of either qualified high or low pressure makeup coupled with an unisolated break in the bottom of the vessel would result in a continuing drop in water level over the core and eventual core uncovery.

An integrated pictorial overview of the concerns expressed in this section is provided in Figure 2-3. Appendix A contains an estimate of the risk associated with a pipe break in the SDV system.

3. FINDINGS

3.1 <u>During a BWR reactor scram, the SDV system piping becomes an extension of the reactor coolant boundary outside primary containment. During this (scram) condi 'on, only non-redundant (scram outlet) valves protect against an uncontrolled blowdown of the reactor coolant which could arise from a postulated pipe break in the SDV system piping.</u>

As discussed previously, during a reactor scram the boundary of the reactor coolant system is extended beyond the scram outlet valves to the SDV system piping which accepts, contains, and limits the high pressure reactor water exhausted during a scram. The SDV system piping would normally pressurize to full reactor pressure unless the scram outlet valves are reclosed immediately after full control rod insertion. Isolation of a postulated break in the SDV piping during a reactor scram would depend upon successful reclosure of each of the scram outlet valves. There is only one such valve in the flow path from each of the 185 control rod drives to the postulated break.

This single "isolation" valve arrangement appears to violate those portions of General Design Criteria 54 and 55 of Appendix A to 10 CFR 50 which require that reactor coolant pressure boundary piping systems penetrating primary containment be provided with <u>redundant</u> isolation and containment capabilities which reflect the importance to safety of isolating these piping systems. Clearly, the use of a single isolation (scram) valve does not meet these criteria for the containment isolation function. It is equally clear, however,

that the use of an additional redundant automatic "isolation" valve in the scram discharge (riser) line would adversely impact the reliability of the scram function aspect of the lines. Thus, while opening only a single valve (to cause a rod to scram) is clearly desirable from a scram function reliability viewpoint, the availability of only a single valve (to isolate a break in the SDV system piping) is clearly equally undesirable (if not unacceptable) from a containment isolation function reliability viewpoint. Implicitly, it may be concluded from the single scram outlet valve arrangement that the overriding need for a highly reliable scram function has taken precedence over (and at the expense of) the reliability of the containment (and break) isolation function.

3.2 The non-redundant (scram outlet) valves do not utilize a closure principle or provide a design arrangement with a reliability reflecting the importance of isolating a postulated pipe break.

The use of scram outlet valves for reliable isolation of a postulated break in the SDV system piping attendant to a reactor scram appears to violate those portions of General Design Criteria 54 and 55 of Appendix A to 10 CFR 50 which require that reactor coolant pressure boundary piping systems penetrating primary reactor containment be provided with reliable isolation and containment capabilities which reflect the importance to safety of isolating these systems. As noted earlier, group closure of the scram outlet valves has not heretofore been defined as a required safety function. Accordingly, the systems upon which scram outlet valve operation is dependent have not been designed with features to assure reliable closure of these valves.

Reliable group opening of these valves has been established as a required safety function, to assure a reliable scram function. Because of the need for a reliable scram, the reactor protection and control air systems have been designed such that the numerous possible failure states of either of these systems would cause the scram outlet valves to open, which is in the "fail safe" direction for scram function reliability. Conversely, the same possible failure (loss of) modes of these two systems have the opposite impact on the reliability of the valves in the group closure sense. That is, the list of possible active and passive failure states of the reactor protection and control air systems which will cause the scram valves to open also represents the list of possible common failure modes which would prevent group closure of the scram outlet valves when reactor coolant boundary integrity and containment isolation are needed.

Some of these common failure causes are readily correctable thereby permitting relatively prompt remote manual group reclosure of these valves, e.g., a reactor trip condition which can be quickly bypassed in either the SHUTDOWN or REFUELING mode. Other causes would not be correctable even in the long term, e.g., rupture of a copper tubing control air line caused by a postulated high energy (pipe whip) type break in the SDV system piping or a seismic event. Access to the source of failure for repair likely would be precluded by the harsh environmental conditions created by the break. Thus, the reactor coolant blowdown would not be considered terminatable by reclosure of the scram outlet valves.

3.3 The reliability of equipment currently installed and the capability of SDV system pipe break detection is neither commensurate with the needed reliability for break isolation nor reflective of the potential consequences of a rupture of the SDV system piping.

Typically, BWR plants like BF-3 have radiation monitors located in each of the CRD-HCU areas of the reactor building. However, this instrumentation is not safety grade nor is it supported by Technical Specification operability and trip setpoint (calibration check) requirements. These instruments are also of a single channel design. The reactor building does have reliable high radiation monitors in the various zones of the ventilation system exhaust duct work. These zone radiation monitors are used for automatic zone isolation of the reactor building and for automatic initiation of the standby gas treatment system. The operability and trip set point of these instruments are covered by Technical Specification operability and calibration check requirements. However, these instruments are not sufficiently close to the CRD-HCUs and SDV headers to provide reliable and unambiguous detection of breaks in this equipment. Accordingly, we find that the reliability of the current break detection function of the overall "man-machine" arrangement for SDV break isolation cannot be assured to the degree which would normally be required of a primary containment or a reactor coolant pressure boundary isolation system. Operator action to initiate manual reclosure of the scram outlet valves in the event of an SDV system break would be uncertain.

3.4 A postulated break in the SDV system piping during a reactor scram with a failure to reclose the scram outlet valves would result in an uncontrolled reactor coolant blowdown outside primary containment which could threaten the ECC systems and the availability of makeup water required for long-term core cooling.

As previously discussed, since the SDV system piping is located in the reactor building and outside primary containment, a postulated break there would result in a reactor coolant blowdown outside primary containment (unless the scram outlet valves are reclosed). Furthermore, since the SDV piping is below the level of the core and drains from inside the core shroud, reactor hot water could continuously drain out of the reactor vessel and onto the floor of the reactor building. Additionally, an unisolated SDV break inside the reactor building would also, sooner or certainly later, threaten the operability of the emergency core cooling systems required for mitigation since the ECC system pumps are located in the basement of the building. The adverse environmental conditions created by the hot water break, together with potential flooding conditions, would make operability of this equipment questionable before very long. Moreover, the water lost from the reactor coolant system would be unavailable to the normal heat removal recirculation flow path (i.e., torus, low pressure ECC system and return to vessel) required for long-term cooling. Accordingly, unless the water which is lost from the RCS can be returned to the condensate storage tank (for return to the vessel), all normal ECCS inventory eventually will be depleted. At this point, an alternate makeup source would have to be provided if pumps were still available to deliver the water to the reactor vessel.

3.5 A break in one or more control rod drive scram exhaust lines located upstream of the scram outlet valves and outside primary containment would result in an unisolatable blowdown of reactor coolant outside of primary containment even if all scram outlet valves were closed.

Except for the manual isolation valves immediately upstream and downstream of the scram outlet valves, there are no valves in the scram exhaust lines between the CRDs and the SDV which could be closed to isolate a break. Thus, should one or more of the 3/4 inch Schedule 80 exhaust lines rupture upstream of the scram outlet valves and outside primary containment, closing these valves would not isolate the break. Furthermore, since the subject piping is below the level of the core and drains from inside the core shroud, hot reactor water would continuously drain out of the reactor vessel and onto the floor of the reactor building.

It should be noted that this situation is different, for example, from the small diameter BWR transversing incore probe (TIP) system instrument lines which also penetrate the bottom of the reactor vessel. The TIP lines do incorporate redundant and diverse isolation valves immediately outside the drywell to provide isolation protection. Break isolation of the scram exhaust lines is also different from the situation for ruptured PWR steam generator tubes. For this case, leaks through the ruptured tubes (which would place the lost reactor coolant outside containment) can be conveniently terminated by draining the primary system down to a level exposing the break elevation of the tubes. The lowest elevation of the tubes is still well above the top of the core; thus, the break flow can always be terminated eventually. Since all of the BWR scram exhaust piping (and SDV system piping) is well below the core elevation, draining the RCS to uncover and thereby terminate the break flow from the bottom of the reactor vessel would not be possible.

The CRD seal leakage flow passing through a single scram exhaust line could range between 3 gpm and 5 gpm immediately after the break to about 12 gpm after CRD seal degradation (assuming a 250 psi pressure differential). The flow would be considerably higher for a larger pressure differential which might be the case for breaks immediately outside primary containment. Thus, rupturing only a few of these lines could quickly result in a cumulative break flow which would exceed the capacity of the two 50 gpm sump pumps in the reactor building basement.

Although a single passive failure might legitimately be postulated for any pipe in the reactor coolant boundary (including a scram exhaust line), no SDV system pipe break is thought to concurrently involve the rupture of several exhaust lines. Multiple line failures might occur, however, due to such causes as large high energy pipe breaks, sabotage or interaction with heavy equipment (e.g., fuel shipping railroad cars) in the vicinity of the hydraulic control units in the reactor building.

3.6 <u>The assurance provided by the industry codes and vendor quality assurance programs for the mechanical design, fabrication, installation, testing and inspection of the SDV system piping do not appear to be commensurate with the risks associated with an accidental rupture of this piping without isolation.</u>

As discussed previously, a break in the SDV system piping without isolation could result in severe consequences including possible core uncovery since the break might threaten continued operability of the emergency core cooling systems and the availability of makeup water. Additionally, the reliability of the break isolation arrangement upon which prompt mitigation of the event would be dependent, is considered to be less than adequate. Under such circumstances it would appear to be appropriate to compensate, in part,

for these systems-related deficiencies and safety concerns by providing a higher degree of assurance for the mechanical integrity of the SDV system piping during the life of the plant. A review of the current basis for assuring mechanical integrity of the SDV system piping shows that this assurance is not commensurate with the possible consequences associated with a postulated break in this piping.

For most of the operating BWRs (i.e., those for which the SDV system mechanical design was initiated before about 1971), the SDV piping system was probably designed, fabricated, installed and inspected to the requirements of USA Standard Code for Pressure Piping-Power Piping,USAS, B31.1. This code did not provide for a detailed quality assurance program for design, fabrication and construction. Also, piping systems for use in water service and built in accordance with B31.1 were not required to have volumetric examinations of welds except for those with nominal wall thickness greater than 1-5/8 inches. Pipes of one to two inches in diameter such as drain, vent and instrument lines were not required to have examinations.

The Section III ASME B&PV Code rules for Class 2 components were available in 1971. Plants granted a construction permit from 1971 through 1973 would probably have been specified to construct the SDV system piping to the Class 2 rules rather than B31.1, but it could vary depending upon the order date for the component. The B31.1 and Class 2 rules are similar and nether requires a thermal fatigue analysis (thermal expansion fatigue by anchors is included).

The Browns Ferry-3 SDV system was constructed by Reactor Controls, Inc. (RCI) of San Jose, California. From conversations with RCI representatives, it has been learned that most operating BWR/3 and BWR/4 SDV systems (including the CRD-HCU piping networks) were constructed by RCI. More recently, RCI has expanded its scope of supply to include the mechanical engineering design and analysis of the SDV systems. The SDV systems for BWR plants now under construction would be built to the ASME B&PV Code, Section III, Subsection NC rules for Class 2 Components. The Code requires that this work be done in accordance with the quality assurance requirements of ASME Section III Article NCA-4000. However, examination of the construction deficiency report for LaSalle County Station (see Appendix B) shows that contrary to these requirements, "Reactor Controls, Inc., (designer and installer of portions of the Control Rod Drive System) did not have a QA/QC program that addressed the areas of ... design control, ... and detailed implementing procedures for design, installation, and inspection activities." From this inspection report it may be inferred that most operating BWR SDV systems were not constructed to the high quality assurance standards now considered to be appropriate and reflective of the potential consequences associated with an accidental rupture of this piping without isolation.

Finally, inservice inspection of SDV components built to Section III would be conducted in accordance with the ASME B&PV Code, Section XI, Subsection IWC rules for Class 2 components. Section XI rules would, most likely, also be followed for SDV components constructed to B31.1 rules because Section 50.55a of 10 CFR Part 50 requires periodic updating of inservice inspection programs for each plant. The CRD scram exhaust risers and the SDV vent and drain lines could be exempted from examination because they are smaller

than the 4" diameter exemption provided in the Code. The SDV header should not be exempted on either size or pressure considerations, but it is not apparent that all plants include the header in their inservice inspection program. One argument that might be used to explain why the header is not included is that there is no need to examine the larger pipe because the maximum break flow is limited by the flow from a single 3/4 inch scram exhaust riser. If the header is exempted by this reasoning, then the only inservice inspection required by the Code would be the system pressure test once every 3-1/3 years and the system hydrostatic test once every ten years.

4. RECOMMENDATIONS

1. Require that the CRD-HCU exhaust lines and SDV system piping meet the highest standards for design, fabrication, installation, testing, inservice inspection and quality assurance which can be reasonably attained.

In view of the potentially serious consequences associated with pipe breaks in the SDV system without isolation and the significant difficulty and issues involved in improving break isolation reliability, it would appear most appropriate to first assure that the probability of an SDV system pipe break has been adequately minimized. However, from our investigation we found that the level of mechanical integrity assurance presently provided for the life of the plant is significantly deficient. We, therefore, recommend that a thorough re-review of the mechanical design, fabrication, installation, testing, inservice inspection and quality assurance standards and requirements which were applied to the existing CRD-HCU and SDV systems be undertaken with the intention of evaluating their adequacy and upgrading as necessary and practicable. Requiring a complete fatigue analysis and a more extensive and frequent inservice inspection of the small diameter piping welds for the existing SDV systems are examples of possible improvements in these areas. We also recommend that the results of the actual work performed in these areas for all operating BWRs be thoroughly re-reviewed and re-performed as necessary to assure that the mechanical integrity requirements are met and that the current bases are acceptable. Finally we recommend that these standards be applied to future BWR CRD-HCU exhaust and SDV systems.

2. Assure that reliable and redundant break detection instruments such as temperature, humidity, or radiation monitors are provided in the immediate vicinity of the HCUs and SDV system piping.

An important component of the SDV system "man-machine" break isolation arrangement is reliable break detection. Accordingly, it is recommended that reliable (safety grade) break detection instruments be installed in the immediate area of the control rod drive HCUs and SDV system piping. Detection based on high radiation, temperature, and/or humidity conditions may be used for this purpose. These instruments should be covered by Technical Specification setpoint and operability requirements and should be annunciated in the control room. They should be redundant. To preclude a single failure from disabling the detection link in the man-machine isolation arrangement. Appropriate consideration should be given to adequate environmental qualification. Only with such break detection instruments can reliable and timely break diagnosis and actions by the operator be assured.

3. Develop and implement appropriate emergency operating procedures and operator training for postulated breaks in the CRD insert or exhaust piping or the SDV system piping.

Training provided should familiarize the control room operator with SDV break symptoms, indications, and diagnosis. The emergency procedures developed should require immediate reclosure of the scram outlet valves upon a detected break in the SDV system piping. Emergency operating procedures should include all available mitigation steps if timely reclosure of the scram outlet valves cannot be accomplished. The procedures should be supported by appropriate analyses to demonstrate the most appropriate course of action (e.g., possibly

depressurizing the reactor via the SRVs to reduce the CRD blowdown rate). Subsequent actions required to reclose the scram outlet valves should be developed and provided. Procedures and training required for long-term recovery if the scram outlet valves cannot be reclosed for an indefinite period should be developed and implemented. These procedures should include steps to prevent or delay the possible eventual loss of all ECCS by flooding or environmental damage. Finally, consideration should be given to any special emergency procedures and training which may be required to terminate a reactor coolant blowdown which cannot be isolated by the scram outlet or manual isolation valves because of break location, environmental conditions or valve failure.

4. Consider improving the closure reliability of the scram outlet valves. Various ways should be studied for improving the closure reliability of the scram outlet valves. Such studies should examine concepts for improving the reliability of control air supply (e.g., accumulators) and AC power supply (e.g., individual alternate temporary emergency power supply hookups) to the solenoid scram pilot valves. Any proposed improvements in closure reliability should carefully consider the possible negative impacts on scram reliability.

5. Prior to the initiation of any pressure boundary maintenance on the SDV system pipings, require the manual isolation valve for each scram exhaust riser be closed; and before subsequent startup, require appropriate verification that the manual valves are reopened.

SDV pressure boundary maintenance or modification activities may not be precluded by Technical Specifications from being performed in any reactor mode. However, such activities would normally be expected to take place during periods when the reactor is in either SHUTDOWN or REFUELING mode. Activities which result in a loss of SDV pressure boundary integrity might be performed with only the scram outlet valves closed to isolate the SDV system piping from the

reactor coolant. Maintenance or modification procedures may not require that the HCU manual isolation valves also be closed. If the manual valves are not closed, the scram outlet valves would be maintained closed with both RPS channels energized and control air pressure applied to each of the scram valve actuators. Under such circumstances, should a RPS trip condition (or loss of RPS power) or a loss of control air occur, an uncontrolled loss of reactor coolant outside primary containment would result if the SDV pressure boundary were open at that time. Depending upon the circumstances, reclosure of the scram outlet valves may not be readily achievable. Accordingly, to protect against such an uncontrolled loss of coolant, it is essential that manual closure of the manual isolation valves be required. It should also be noted that opening the SDV system manual flush valves without an operator remaining on standby to assure immediate reclosing, if needed, is another pressure boundary maintenance which requires similar treatment.

6. <u>For plants to be constructed consider locating the SDV system headers and HCUs at an elevation in the reactor building which would place them above the top of the reactor core.</u>

By routing the CRD piping to and from the HCUs and SDV headers to a level above the top of the reactor core, the possibility of an unisolatable break which could drain reactor coolant from below the core would be substantially reduced. It would still be possible for an individual CRD insert or withdraw (scram outlet) line to break below the core level inside the primary containment. However, only a break outside containment above the level of the top of the core could be cross connected by the flow contribution of all of the scram exhaust lines. Thus, with this arrangement it would be possible to terminate a break in the SDV system by bringing reactor system pressure down to atmospheric conditions. Reactor water would not be able to drain outside primary containment to below the level of the top of the core.

5. REFERENCES

1. "Report on the Browns Ferry 3 Partial Failure to Scram Event on June 28, 1980," July 30, 1980, Office for Analysis and Evaluation of Operational Data, USNRC.

2. "Operation and Maintenance Instructions Control Rod Drive System for Browns Ferry Nuclear Plant," GER-9585/9586, June 1971.

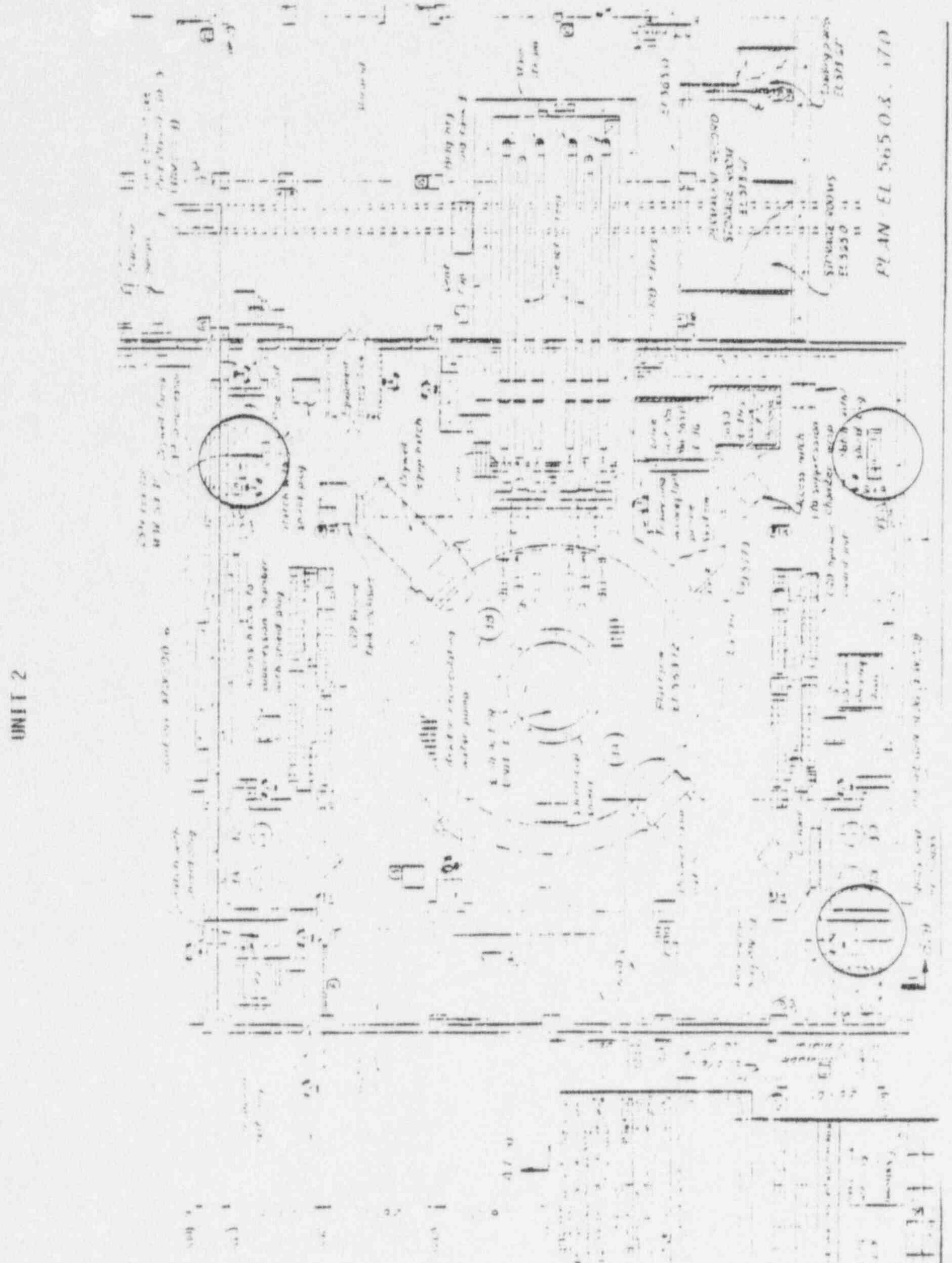

Figure 2-1 Floor Plan -565' Elevation

UNIT 2

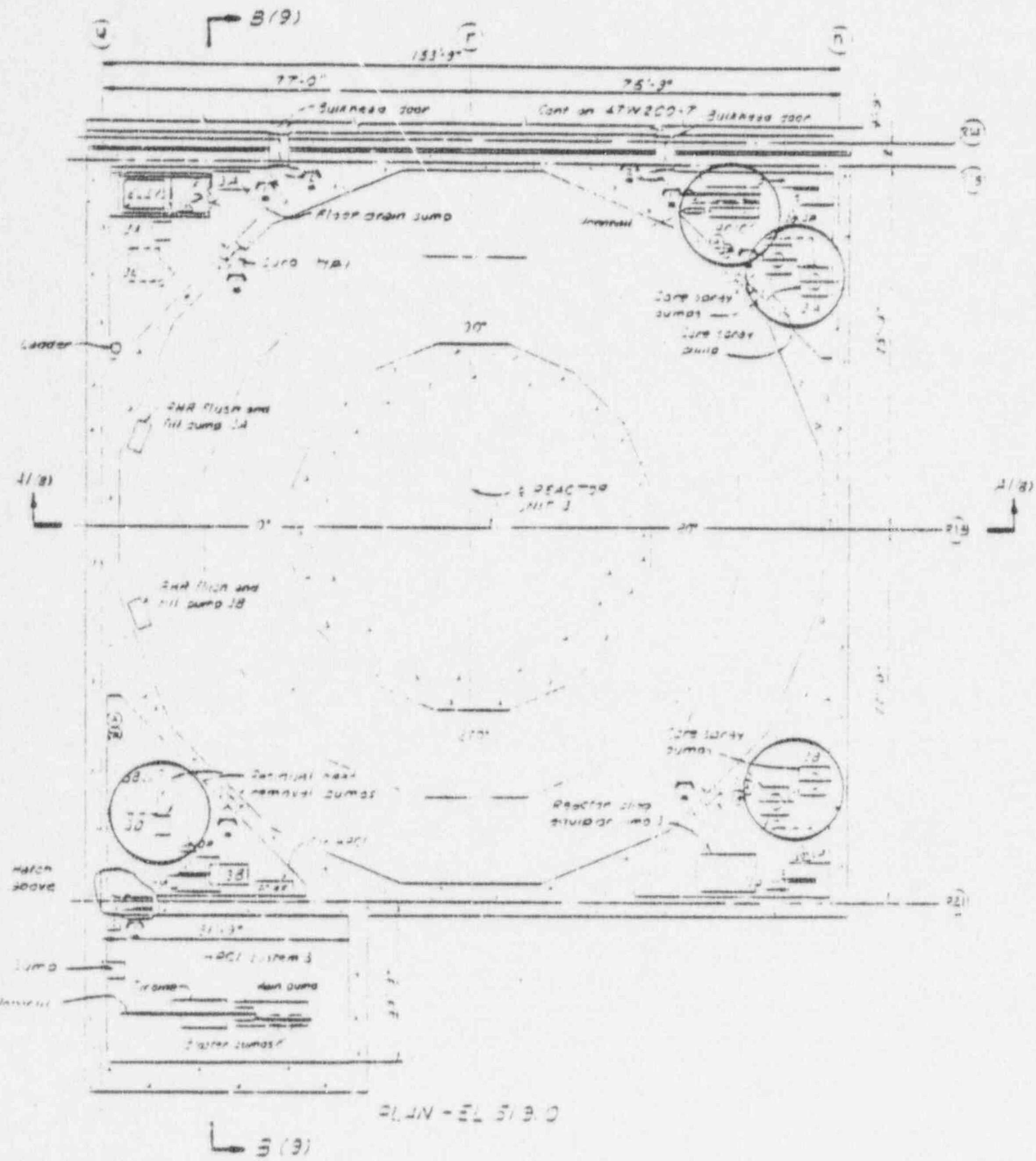

Figure 2-2 Floor Plan - 519' Elevation

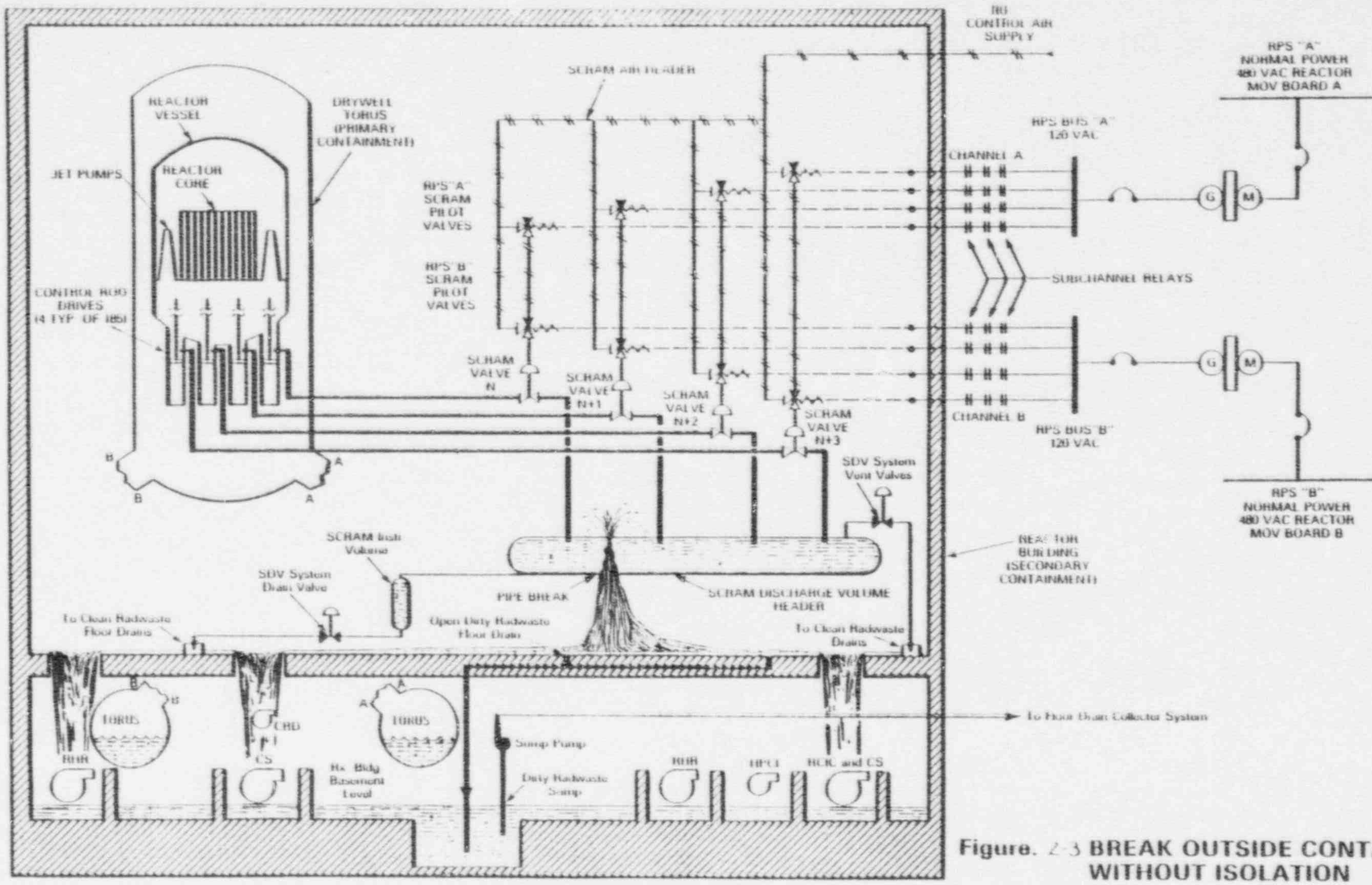

Figure. 2-3 BREAK OUTSIDE CONTAINMENT WITHOUT ISOLATION

(CONTROL AIR FAILURE)

RISK ASSESSMENT

An estimate of the core uncovery risk from a break in the SDV system piping (at a plant like BF-3) might be calculated as follows:

$$P_0 = P_1 \times P_2$$

where,

P_0 = Probability of Core Uncovery/Rx/Yr

P_1 = Probability of an unisolated SDV break/Rx/Yr

P_2 = Probability of core uncovery following an unisolated SDV break

where,

$$P_1 = (N \times P_{11}) \times (P_{12} + P_{13})$$

N = Number of Rx scrams/Rx/Yr

P_{11} = Probability of an SDV Break (>> sump pump cap)/Rx scram

P_{12} = Probability of not being able (RPS or control air condition) to immediately reclose scram valves after a Rx scram/Rx scram

P_{13} = Probability of not reclosing (human or procedural) or being unable to reclose (break consequences) scram valves after an SDV break.

If we assume:

$N = 2$

$P_{11} = 10^{-4}$

$P_{12} = 10^{-1}$

$P_{13} = 10^{-1}$

$P_2 = 0.25$

then $P_0 = 10^{-5}$

Discussion

Based on BWR operating experience it would not be unreasonable to assume that at least two reactor scrams (from full pressure and temperature) occur every year at each plant. It might also be assumed that a break in a small line in the SDV system (downstream of the scram outlet valves and upstream of the SDV system vent and drain valves), resulting in a substantial blowdown rate*, (>> 100 gpm) can occur once in every 10,000 BWR reactor scrams. (A blowdown rate of this magnitude could result in eventual loss of the emergency makeup systems if not isolated.) For BWRs it also seems reasonable to assume that out of every ten reactor scrams, one would involve a RPS trip condition or power supply failure or a loss of control air supply such that the scam outlet valves would not be able to be reclosed for an indefinite period of time. Furthermore, should a break in the SDV system occur, the additional abnormal plant symptoms and reactor system process conditions indicated in the control room could divert and continue to occupy the control room operator's time and attention (e.g., reactor water level drop) which could result in the scram valves being left open. The break itself may also introduce additional failure modes to the break isolation arrangements (e.g., air line failure due a postulated pipe whip of a ruptured SDV system line, environmental damage to the detection equipment, damage to the scram valve teflon seating surfaces caused by prolonged blowdown). We would estimate that considerations such as these could contribute an additional one chance in ten of not isolating a break in the SDV system.

---

* Note: A break from a one inch Schedule 160 vent line is capable of passing approximately 400 gpm at 1,000 psi, while a two-inch Schedule 160 drain line is capable of passing approximately 1,500 gpm.

Finally, in the event of such an unisolated break in the SDV system, we would assume that there is a 75% chance that at least some ECCS equipment in the Reactor Building basement and emergency makeup inventory will be available to keep the core covered continuously and indefinitely even though none of the equipment is qualified for environmental conditions including flooding.

Although the above point estimate is considered to be 10^{-5} /Rx/Yr, which would make this event a significant contributor to risk, the uncertainty range may be such that the uncovery probability most likely lies within the range of 10^{-3} /Rx/Yr to 10^{-9}/Rx/Yr. Consequently it is difficult to conclude on the basis of these numbers alone that the existing plant design configuration is safe, i.e., less than 10^{-6}/Rx/Yr.

If from these convolutions one were to conclude that the SDV pipe break is a significant contributor to BWR core uncovery risk, it is believed that the risk can best be reduced by decreasing the likelihood of a break in the SDV system piping by an appropriate upgrading of the SDV system mechanical integrity assurance basis. The risk can also be reduced in a significant although less favorable or desirable way by improving the reliability of the break isolation arrangements.

Appendix B

INSPECTION REPORT FOR LaSALLE COUNTY STATION

AEOD

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

MAR 3 1981

Docket No. 50-373
Docket No. 50-374

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Vice President
Post Office Box 767
Chicago, IL 60690

Gentlemen:

Thank you for your letter dated February 3, 1981, informing us of the steps you have taken to correct the noncompliance which we brought to your attention in Inspection Report No. 50-373/80-48; 50-374/80-30 forwarded by our letter dated January 9, 1981. We will examine these matters during a subsequent inspection.

In your letter you requested us to reconsider (1) whether the meeting of January 29, 1981 should be classified as an Enforcement Conference and (2) the Severity Level of the noncompliance. We have reconsidered the matter and continue to believe the Severity Level selection is correct and the meeting was an Enforcement Conference.

The Severity Level of these violations was not increased for repeating a previous violation. It was our determination that the problems related to control rod drive pipe suspension systems resulted from degradation of management control systems designed to assure proper plant construction (Severity Level IV). Although a close call, we believed it was not a Severity Level III violation, i.e., lack of quality assurance program implementation related to a single work activity as shown by multiple program implementation violations that were not identified and corrected by more than one quality assurance/quality control checkpoint relied upon to identify such violations.

The meeting is considered an Enforcement Conference because of your noncompliance history related to large and small bore pipe suspension systems. Had the new enforcement policy not been in effect at the time of this inspection, these items would have been infractions and your history would have prompted an Enforcement Conference. Under the new policy we continue to look at past history, so the same conclusion was reached. Although we took the position that the "clock started" at the time of issuance of the revised enforcement policy with respect to counting multiple violations of Severity Level I, II, or III items of noncompliance, it is necessary that the history before issuance of the Policy be considered in the determination of when to hold an Enforcement Conference.

dupe of 8103110375 2pp

You have stated a desire to meet with us to discuss enforcement. We will contact you in the near future to arrange such a meeting.

Sincerely,

James G. Keppler
Director

cc w/ltr dtd 2/3/81:

cc w/encl:
J. S. Abel, Director of Nuclear Licensing
L. J. Burke, Site Construction Superintendent
T. E. Quaka, Quality Assurance Supervisor
R. H. Holyoak, Station Superintendent
B. B. Stephenson Project Manager
Central Files
Reproduction Unit NRC 20b
AEOD
Resident Inspector, RIII
PDR
Local PDR
NSIC
TIC
Dean Hansell, Office of Assistant Attorney General

I.T. Yin 8 38-42610

RIII	RIII	RIII	RIII	RIII	RIII	RIII
Yin/jp 2/25/81	Danielson	Spessard	Reimann 2-26-81	Norelius	Davis	Keppler

Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

February 3, 1981

Mr. James G. Keppler, Director
Directorate of Inspection and
Enforcement - Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

Subject: LaSalle County Station
NRC Inspection Report
50-373/80-48 and 50-374/80-30
NRC Docket Nos. 50-373/374

Dear Mr. Keppler:

In response to the subject inspection report transmitted by your letter dated January 9, 1981, attached are replies to the apparent items of noncompliance in the Notice of Violation. The attached replies include our evaluation of quality assurance program and management control system improvements which will be implemented to preclude further violations of this type.

The primary reason for the violation was inadequate followup of corrective actions identified in our reply to your previous inspection report 50-373/80-20 and 50-374/80-13. This inadequate followup occurred because the LaSalle County Project Construction Management did not recognize their responsibility to followup their contractor's design control corrective actions. This was the only LaSalle County Construction Management controlled contractor with extensive design and analysis responsibility. Design and analysis are normally handled by contractors controlled by the LaSalle County Project Engineering organization; therefore, Construction Management incorrectly assumed the design and analysis corrective actions would be followed by Project Engineering. This lack of responsibility for control of contractor design activities is unique to this specific contractor.

We agree that our followup was not adequate to assure timely corrective actions to deficiencies identified in the vendor quality assurance program by the NRC. As we stated in our meeting on January 29, 1981, Commonwealth Edison had performed an audit of the vendor in May, 1980, in which deficiencies were identified and had scheduled a reaudit of the vendor in November, 1980 to take steps to correct his inadequate response to date. Although our followup was not timely, it did not represent a breakdown in our Quality Assurance program.

dupe of 8103110390 11pp

FEB 0 6, 1981

Your Inspection Report does not discuss the basis you used to determine the severity level of this violation; however, in a meeting on January 29, 1981, you explained the severity level was increased for repeating a previous violation. We stated in the January 29, 1981 meeting that in our opinion the noncompliance cited should not be considered a repeat noncompliance under the new enforcement policy because it was the first occurrence during the period of applicability of the new enforcement policy. Therefore, we respectfully request your reconsideration of considering the January 29, 1981, meeting as an enforcement meeting and the appropriate reassignment of severity level.

Very truly yours,

C. Reed

C. Reed
Vice President

0279B

Enclosure

Response to Notice of Violation

The items of apparent noncompliance identified in Appendix A of the NRC letter dated January 9, 1981, are responded to in the following paragraphs.

ITEM 1

10 CFR 50, Appendix B, Criterion II, states that, "The applicant shall establish at the earliest practicable time consistent with the schedule for accomplishing the activities, a quality assurance program . . ." and Criterion I, states that, "The applicant may delegate to others, such as contractors, . . . the work of establishing and executing the quality assurance program . . ., but shall retain responsibility therefore."

Commonwealth Edison Company Topical Report CE-1A, "Quality Assurance Program for Nuclear Generating Station," Revision 14, dated September 9, 1980, states in Section 2 that, "The quality assurance programs of Commonwealth Edison Company, Architect Engineers and Nuclear Steam Supply System vendors include the requirements of ASME Section III Article NCA-4000, the quality assurance criteria for nuclear power plants for Appendix B to 10 CFR 50 "Quality Assurance Criteria for Nuclear Power Plant," and the mandatory requirements of ANSI N45.2, "Quality Assurance Program Requirements for Nuclear Power Plants" and ANSI N18.7, "Standards for Administrative Control for Nuclear Power Plants." The requirements are implemented by means of detailed quality procedures delineating the means of detailed quality procedures delineating the specific methodology to be used. In addition, individual contractor's, fabricator's and vendor's Quality Assurance programs will include the applicable portions of the Code Standards and Appendix B as they affect the total program."

Contrary to the above, Reactor Control, Inc., (designer and installer of portions of the Control Rod Drive System) did not have a QA/QC program that addressed the areas of organization, interfaces, design control, and document control. In addition, the program also lacked detailed implementing procedures for design, installation, and inspection activities.

CORRECTIVE ACTION TAKEN AND RESULTS ACHIEVED

Based on Audit 1-80-95 (performed November 11, 1980 and November 12, 1980) by CECo QA and CECo Construction review of as-built drawings, a stop work letter dated November 12, 1980 was issued by Project Construction to Reactor Controls, Inc., covering the installation and inspection of safety related CRD piping supports. An expanded stop work letter was written November 13, 1980 to Reactor Controls, Inc., covering all safety related

engineering work since further review of the deficiencies noted in the November 12, 1980 Stop Work Letter were determined to be the responsibility of Reactor Controls, Inc., San Jose Engineering organization. Subsequently, a letter from W. H. Donaldson to J. Millett was written on November 17, 1980 to identify all the open items requiring resolution. The "Action Item List" encompassed the NRC findings and open items, CECo audit findings, the B. R. Shelton letter dated November 6, 1980, and the CECo QA trend analysis letter dated November 14, 1980.

In response to the Stop Work lettes and the action item list, Reactor Controls, Inc., has totally reviewed their QA/QC program. As a result, implementation instruction and a QA manual addenda were written addressing areas where their QA/QC program needed improvement. The QA Manual addenda contains an index which indicates where each point of the 18 point criteria are addressed. The instruction book is indexed to provide a cross reference to the Reactor Control, Inc., QA Manual and the 18 point criteria. Specific items identified in the noncompliance report are discussed below:

1. Organizational Interface:

   Reactor Controls, Inc., has prepared the following procedures to identify various organizational interfaces:

   1) QA 1 3-1 Instruction For Interfaces Between Engineering and Stress Analysis.

   2) QA 1 6-3, Instructions for Document Transmittal for Approval.

   3) RSDA-1, Procedure for Review of Design or Stress Analysis Reports Submitted by Vendors or Subcontractors.

   Additionally, it has been established that the responsibility for the transmittal of engineering and design information will be vested with the Engineering and Construction Manager for Reactor Controls, Inc., and the cognizant LaSalle County Project Construction Engineer.

2. Design Control:

   Reactor Controls, Inc., has developed the following procedures to control design:

   1) QA 1 3-1, Interfaces between Engineering and Stress Analysis; QA 1 3-2, Drawing Changes.

2) RSDA-1, Procedure for Review of Design or Stress Analysis Reports Submitted by Vendors or Subcontractors.

3) QA 1 5-2, Engineering Drawings and Engineering Change Notices; QA 1 6-2, ECCL control.

3. Document Control:

Reactor Controls, Inc., has recently instituted a computerized system for controlling documents which have been reviewed and approved for use by their Project Engineer. All documents which constitute the Engineering Controlled Checklist (ECCL) will now be included in the computerized system. The following procedures implement Reactor Control's document control system.

1) QA 1 3-2 Drawing changes

2) QA 1 5-2 Engineering drawings and engineering change notices

3) QA 1 5-1 Procedure control

4) QA 1 6-1 Document control headquarters

5) QA 1 6-2 ECCL control

6) QA 1 6-3 Document transmittal for approval.

7) QA 1 6-4 Document control site/shop

8) QA 1 6-5 Document control system (computer)

4. Installation and Inspection:

Reactor Controls, Inc., has developed QA 1 3-2, Installation of component supports to further cover the installation and inspection of the CRD piping supports. Reactor Controls, Inc., is also developing a final walkdown procedure to be used for final inspection and verification of the as-built CRD piping and support system. This procedure will encompass the requirements of IE Bulletin 79-14.

The LaSalle County Project Construction Engineer and the Site QA Supervisor reviewed the preliminary drafts of the implementation procedures and the QA Manual addenda in San Jose December 9, 1980

through December 12, 1980, in order to determine that all open items were being addressed. Comments on these procedures and their response to the action item list were given to Reactor Controls at that time. The formal transmittal of these procedures was received on site January 12, 1981, and are currently being processed through the formal review by CECo and S&L.

A preliminary review and follow-up of implementation procedures and the QA Manual was performed by the NRC Region III and Region IV inspectors between January 12, 1981, and January 15, 1981, in San Jose and at Earthquake Engineering Systems (EES), Reactor Control's analysis subcontractor, in San Francisco. It was explained to the inspectors that we had not yet initiated formal review and, therefore, no approval of any Reactor Control's procedures had been given. Some procedures were still being developed. The NRC inspectors acknowledged this and indicated their review was solely to keep abreast of the Reactor Control, Inc.,/CECo corrective action progress.

All items raised during this NRC inspection were either in progress or were being reviewed and resolved. The design and acceptance criteria for stiffness, deflection, frequency, loading combinations, are currently being reviewed by S&L and Reactor Controls. Reactor Controls is doing physical testing of clamps and unistrut material. These test results will be compared to the calculated values used by EES in the CRD pipe support analysis. Sargent & Lundy is revising specification J-2922 to incorporate ECNs M-283-LS and M-285-LS in an Amendment. These previously transmitted ECNs contained in the design information necessary for RCI to complete the analysis.

CORRECTIVE ACTION TO AVOID FURTHER NONCOMPLIANCE

The contract with Reactor Controls is unique. No other on-site contractor has extensive design and analysis responsibility coupled with the normal material supply and erection contract. The division of responsibility within CECo, that is, Engineering is responsible for design whereas Construction is responsible for administration of contracts which contain major field erection, lead to ambiguous control of the design portion of Reactor Controls scope of work. As a result some of the open items from NRC Report 50-373/80-20; 50-374/80-13 were not adequately followed up to assure successful corrective action prior to the NRC inspection recorded in Report 50-373/80-48; 50-373/80-30. To resolve this problem, LaSalle County Project Construction has been given the responsibility for the overall administration of Reactor Control's contract. Project Engineering and S&L will provide assistance and information

as necessary but all design and engineering information transmitted to Reactor Controls will be transmitted with the knowledge of the LSC Project Construction Engineer to the Reactor Control, Inc., Engineering and Construction Manager. Similarly, Reactor Control's engineering and design information will be transmitted from the Reactor Control Engineering and Construction Manager to the LSC Project Construction Engineer.

The previously discussed Amendment to specification J-2922 will include all outstanding ECNs, thus incorporating all design and technical information in one package. The establishment of the single line responsibility and interface between Reactor Controls, Inc., Engineering and Construction Manager and LSCS Project Construction Engineer combined with the amended specification encompassing outstanding ECNs should improve design control. In addition, review, approval, and implementation of Reactor Control procedures previously referenced will provide the QA/QC controls necessary for design, document control, installation and inspection.

DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED

Full compliance is expected to be achieved generally in accordance with the following schedule:

1. Submittal, Review, and Approval of Procedures
   2/2/81 - 2/6/81

2. Reactor Control, Inc., Training and Implementation
   2/2/81 - 2/6/81 (off site) 2/9/81 - 2/13/81 (on site)

3. Partial Life - Document Control, QC Inspection, HCU Bracing Detailing and Material Purchase.
   2/6/81

4. Partial Life - CEA Installation
   2/13/81

5. Partial Life - CRD HCU Bracing Erection
   2/13/81

6. Implementation Audit in San Jose

7. Implementation Audit - Site

8. Lift Stop Work
   2/20/81

In this regard, we shall provide a copy of the RCI documentation package after final CECo approval has been given in order to expedite your review. We request, therefore, that your verification review be timely so that work can be reinitiated on this project on the schedule defined above.

ITEM 2

10 CFR 50, Appendix B, Criterion XVIII, states that, "A comprehensive system of planned and periodic audits shall be carried out to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program."

Commonwealth Edison Company Topical Report CE-1-A, "Quality Assurance Program for Nuclear Generating Stations", Revision 14, dated September 9, 1980, states in Section 18 that, "Audits will be performed by Commonwealth Edison Company and/or its contractors, subcontractors and vendors to verify the implementation and effectiveness of quality programs under their cognizance" and "Audits will be performed selectively at various stages of contracts on a varying frequency, based on the nature and safety significance of the work being done to verify compliance and determine the effectiveness of procedures, inspections, tests, process controls and documentation."

Contrary to the above, audits of Reactor Controls, Inc., appeared to be inadequate in that there was no systematic evaluation of contractor performance and audit findings were not resolved in a timely manner.

CORRECTIVE ACTION TAKEN AND RESULTS ACHIEVED

1. As indicated in Commonwealth Edison's letter of June 6, 1980, responding to noncompliance items in report 50-373/80-20 and 50-374/80-13, an established program of Audits and surveillances does exist for RCI on-site and off-site activities. RCI's off-site activities had been periodically reviewed during scheduled audits in May, 1977, with follow up and close out June, 1977; in March, 1979, with follow up and close out June, 1979; in March, 1980, with follow up and close out June through August, 1980. This planned evaluation process for off-site activities was in addition to 4 on-site audits of RCI in 1977 4 in 1978, 8 in 1979, and 10 in 1980, as well as numerous surveillance of on-site activities. The structure of the RCI organization is such that many on-site reviews necessitate evaluation of documents prepared off-site and as such, our on-site audits and surveillances were indirectly reviewing off-site activities.

2. The CECo audit of RCI and Earthquake Engineering Systems (EES) conducted on march 25, 26, and 27, 1980, reviewed in detail the RCI design, design control, and design personnel qualifications for the control rod drive (CRD) piping hangers. Four items of noncopmpliance were identified and later closed out through review of RCI management commitments and documents transmitted to the site. Commonwealth Edison has always had an established program for monitoring corrective action and ultimately closing out the audit noncompliances when resolved to our satisfaction. Commonwealth Edison believes that this program was complied with during the close out of this audit.

3. Commonwealth Edison QA does acknowledge the fact that QA did not follow up and verify effective close out of the items identified during NRC inspector Yin's audit of RCI (San Jose) in April, 1980 (NRC Report 50-373/80-20 and 50-374/80-13). For deficiencies identified by the NRC at off-site vendor locations, it has been the practice, for engineering related items, that the Commonwealth Edison Engineering organization respond to, and be responsible for, follow up and close out of the deficient item. Commonwealth Edison engineering responded to the NRC citations indicating satisfactory resolution had been achieved. In these cases, Quality Assurance would not have initiated any follow-up action to assure satisfactory resolution. This problem is now resolved with the clear identification of the cognizant Construction Engineer as overall contract administrator.

4. In light of RCI's failure to initiate and complete adequate corrective actions as committed in CECo's response of June 6, 1980, QA recognizes the need to establish a system to track the corrective action commitments for NRC Region III Off-Site vendor inspections and verify proper resolution. This would be in addition to our normal practice of monitoring follow up progress for on-site deficiencies. In an effort to provide this coverage, the Quality Assurance Department has established by Memorandum #17 dated January 14, 1981, a program which requires site QA track all NRC items with a monthly status report submitted to the Manager of QA. This monitoring process is expected to assure timely completion of committed corrective action and should improve the effectiveness of the Commonwealth Edison QA program in this area.

5. Relative to the specific matters of concern identified by Mr. Yin during his November, 1980, audit of RCI, San Jose, immediate action was taken by the Commonwealth Edison Engineering organization when it was determined that follow up action was not adequately completed. Separately, Site QA and Project Construction had been pursuing resolution of on-site audit deficiencies prior to Mr. Yin's trip to RCI. On October 21, 1980

site QA scheduled an audit of RCI's on-site organization for the week of November 10. This audit was to include formal review of corrective action taken by RCI in response to earlier CECo on-site audits. That audit identified inadequate corrective action by RCI on CECo items. As a result, installation and inspection for all Safety Related CRD Pipe Supports was stopped on November 12, 1980. This "stop work" was later expanded to include all related Engineering activities in San Jose. The stop work will remain in place until Project Construction, with the concurrence of Commonwealth Edison QA, is satisfied that adequate corrective action has been completed.

6. When Commonwealth Edison was advised by RCI that they had prepared, in draft form, what they considered the majority of procedures necessary to resolve Commonwealth Edison and NRC concerns, the Site QA Superintendent and the cognizant Project Construction Engineer performed an intensive review of the draft documents at San Jose. Comments were provided and in the case of the design interface document, total rewrite of procedure was recommended. The incorporation of all comments has been completed and submittal of required documents began the second week of January. Following review and approval of the necessary procedures, site QA plans to review the corrective action on site and in San Jose prior to allowing RCI to return to work. This will be followed by an extensive audit of RCI's implementation both on site and off site promptly after returning to work.

CORRECTIVE ACTION TO AVOID FURTHER NONCOMPLIANCE

In addition to the Commonwealth Edison QA/QC Program changes addressed in ITEM 1, and the implementation of Quality Assurance Department Memorandum #17 which was discussed above, the Commonwealth Edison QA Department has been reorganized to improve the effectiveness of QA management levels in addressing Quality concerns. Each of the construction sites now has three supervisory level personnel, 2 QA Supervisors and a QA Superintendent rather than a QA Supervisor as in the past. This change should allow the Site QA organization to follow on-site and off-site Quality Items more closely. More management attention to significant quality matters and consequently quicker resolution of Quality Related Problems is expected.

This focusing of the attention of the responsible CECo Field Engineer on the QA/QC activities associated with a project as well as the administrative changes made in the conduct of activities by the CECo QA Department will prevent recurrence of the deficiencies identified.

DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED

The administrative changes in the conduct of review of on-site contractor QA activities has been implemented, including the addition of a Site QA Superintendent. Final review and acceptance of the RCI QA/QC Program changes will be completed as defined in ITEM 1. The CECo QA verification audit of RCI promptly after the stop work order has been lifted.