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| AUTH.NAME | AUTHOR AFFILIATION |
|---|---|
| MINECK,D.L. | Iowa Electric Light & Power Co. |
| RECIP.NAME | RECIPIENT AFFILIATION |
| HALL,J.R. | Office of Nuclear Reactor Regulation, Director (Post 870411 |

SUBJECT: Forwards evaluation of CRD line leakage identified during insp of drywell air gap in May 1990.

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Iowa Electric Light and Power Company
September 10, 1990

NG-90-2135

Mr. J. R. Hall
Licensing Project Manager-DAEC
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
CRD Line Leakage
File: A-101, C-11

Dear Mr. Hall:

Please find enclosed a copy of our evaluation of the CRD line leakage identified during an inspection of the drywell air gap in May 1990. This report summarizes our previous efforts to identify the leakage, the results of the destructive and non-destructive examinations of the affected piping, and the status of our root cause investigation. In addition, a summary of General Electric's Startup/Operational Assessment Report on Degraded Control Rod Drive Insert and Withdraw lines is included as an appendix to the report.

As stated in the report, we are implementing an on-line monitoring program to assist in our continuing effort to determine the root cause. We will keep you informed of any new or significant developments in this effort. In the unlikely event leakage is identified during Cycle 11 operation, appropriate actions will be taken in accordance with the the guidance provided in Generic Letter 90-05, "Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1,2 and 3 Piping", in addition to the preparation of any Justifications for Continued Operation which may be required.

Should you have any questions or concerns regarding this matter, please contact R. Browning at (319) 398-4485.

Very truly yours,

Daniel L. Mineck
Manager, Nuclear Division

DLM/PMB/pjv+

cc: P. Bessette
L. Liu
L. Root
R. McGaughy
J. R. Hall (NRC-NRR)
A. Bert Davis (Region III)
NRC Resident Office
Commitment Control

## ENGINEERING EVALUATION OF POTENTIALLY

## DEGRADED CRD PIPING

### INTRODUCTION

On 5/19/90, during a video probe inspection of the drywell air-gap (annulus), leakage was detected from a SW quadrant Control Rod Drive (CRD) insert and withdraw line. This document discusses the history, analysis and potential consequences of this leakage.

### HISTORY

Leakage was first documented at the torus downcomer penetration X-5E (Fig. 1) in August 1985. Water was found dripping down the wall from this penetration to the Torus Room basement. Initial investigations into the source of this leakage were unsuccessful. Shortly thereafter, the leakage stopped and investigations were suspended.

Leakage from this same penetration was again noted in September 1987. Chemical analysis of the water did not detect any radioisotopes. The sample size (10 ml) and counting time (1000 sec), however, were designed to detect relatively high activity in reactor water. Inspections of the air gap were performed using a fiberscope inserted through the spacing around penetration X-5E. These inspections were unsuccessful in identifying the leakage source.

In November, 1987, a log was initiated to quantify the leak rate. This log was designed to correlate the leak rate to drywell temperature, power level, work in the Reactor Building (RB), or to a particular system or component. Logging of the leak rate continued for about two months. During this period, the leakage was quantified at approximately 1 gph. However, no trends could be identified or correlations made between the leak rate and drywell temperature, power or specific equipment so the logging was discontinued.

Since no conclusive evidence had been obtained from the leak rate measurements, a video probe was purchased in 1988 to help pinpoint the source of leakage. A plan was devised to obtain a general sampling of the entire drywell air gap by inserting the probe through accessible penetrations. A total of 12 electrical and piping penetrations were inspected but no signs of water or leakage were detected.

In 1989, an engineer was assigned full-time to investigate and identify the leakage source. A project plan was formulated that included a review of NRC concerns, industry experience, DAEC experience and development of a root cause analysis program. As part of the root cause program, several possible sources were identified: condensation, drainage, and leakage. Further analysis eliminated condensation and drainage as likely sources so the investigation focused on leakage.

After researching plant drawings and conducting discussions with plant personnel, it was found that very little equipment is located adjacent to the concrete bioshield outside the drywell air gap. Of this equipment, a smaller amount has water that is not radioactive. Since no analytical explanation could be found, a full scale investigation and examination of drywell penetrations with the video probe was initiated.

In December 1989, the video probe identified water running down the drywell shell in the SW quadrant. Attempts were made to follow the small streams upward but it was impossible due to the limited flexibility of the probe. Inspections of other penetrations in the area were unsuccessful in finding any water or the source of the streams.

In January 1990, a two-liter sample of leakage water was chemically analyzed. This time, the sample was counted longer and radiotropic analysis identified low levels of Cobalt 60 which indicated condensate demineralized effluent water (CRD cooling supply) as a likely source.

The inspection efforts were temporarily suspended in January 1990 when the video probe was damaged. The effort was started again in May 1990 with a new, more advanced articulating video probe.

On 5/17/90, an inspection of the external drywell shell using the video probe detected moisture around the CRD insert and withdraw lines located in the SW quadrant. As it was possible to manipulate the video inspection probe around only a few CRD lines at this time, it was not known which specific lines were wet and the source of the water could not be determined. Although the containment wall and a few CRD lines were damp, no leaking lines were observed.

On 5/19/90 more detailed examinations were performed by inserting the video probe into the sleeving around the individual CRD lines in the top row of the SW bundle. The probe was inserted at CRD 26-07W (upper left corner of the bundle), and water was observed along the containment wall and on the CRD piping; again, no leaking lines were discovered. The probe was removed and re-inserted into the pipe sleeve for CRD 22-11W, and a pinhole leak was seen emanating from one of the pipes in this general area. The leak was a steady stream at an estimated rate of 1 to 2 gallons per hour. To determine which line was actually leaking, several of the CRDs in this area of the bundle were exercised. A change in the water leakage rate and direction was observed when CRD 30-07 was given a withdraw signal. This indicated that the withdraw line of CRD 30-07 was the source of the leak. The leak was identified at the toe of the fillet weld attaching the pipe to the containment wall as shown in Figure 2.

The probe was moved within the same bundle to different CRD lines for individual examinations. Another pipe had what appeared to be a thru-wall circumferential crack of at least 200°. The crack was also located at the toe of the fillet weld. Water was not streaming out of this line but there appeared to be some minor weeping within the area of apparent cracking. Upon further inspection, roughly half of the SW bundle lines (50) were found to be visibly wet and others had mineral streaks indicating they had been wet in the past. Some corrosion and discoloration was seen on the pipes and drywell shell.

A Non-Conformance Report (NCR90-016) was initiated to document the situation. An engineering evaluation of the degraded CRD insert and withdraw lines in the SW bundle was prepared in response to the NCR. This evaluation was used to support continued operation of the DAEC until the Cycle 10/11 Refuel Outage scheduled to begin 6/28/90.

A detailed visual inspection of the individual CRD lines using the video probe was conducted in the NE, SE and NW bundles. One hundred percent of the lines in the remaining three bundles were visually inspected. These inspections showed the lines to be dry with no evidence of cracking.

## NDE EXAMINATION OF THE SW BUNDLE

Following plant shutdown, the CRD line to drywell shell welds in the SW bundle were inspected using a 45° mounted, 1/4", 5 megahertz transducer on a stud probe. The results of this initial inspection of the annulus (air gap side) fillet welds showed 4 lines with thru-wall indications, 44 lines with outside diameter (O.D.) indications only and 2 lines with no indications at all. UT results for the drywell side fillet welds showed indications of cracking (again O.D.) in 22 of the lines.

Four of the CRD lines (including the fillet welds to the drywell) in the SW bundle were cored out and shipped to Packer Engineering for destructive analysis and metallurgical evaluation. The samples were chosen based on the visual and ultrasonic examination results, as well as the need for obtaining information for both insert and withdraw lines. The results of this initial metallurgical examination of CRD lines 30-07W, 30-07I, 26-07W and 26-07I are as follows:

- All displayed cracks in the 2:00 to 5:00 sector and striations indicative of high-cycle fatigue.
- Metallographic sections of the fractures exhibited a transgranular fracture consistent with a fatigue mechanism and also displayed evidence of periodic crack growth, i.e. cyclic in nature versus continuous.
- Cracks originated on the outside diameter in the toe region of the fillet weld on the pipe side.

These results were discussed among and concurred with by Iowa Electric and metallurgical experts from Packer and Nutech Engineers. In addition, NRC Region III personnel observed the analysis activities at Packer Engineering.

Subsequently, 13 more CRD lines were metallurgically examined by Packer Engineering. These additional samples were distributed throughout the SW bundle and represented both "wet" and "dry" pipe sections. A summary of the Packer Engineering destructive analyses is shown in Table 1. No results contradictory to the results of the first 4 examinations have been noted.

These additional destructive examinations helped the personnel performing the UT inspections to refine their ability to detect and size indications. The UT inspection results for the SW bundle were then re-analyzed. The final results of the UT inspections of the SW bundle are shown in Figures 3 and 4.

INSPECTION of NW, NE, SE, QUADRANTS

Due to the uncertainty concerning the root cause of the failures in the SW bundle, a decision was made to inspect a sample of the other three bundles of CRD lines (NW, NE, SE quadrants) to determine if these bundles exhibited the same magnitude of cracking as the SW bundle.

An inspection and sampling plan was developed to statistically predict the distribution of cracking that existed in the uninspected piping and to establish a confidence level and probability that a given crack size would not be exceeded in the population.

The sampling plan made the following assumptions:

- The crack size distribution of the SW bundle is a normal distribution.
- The crack size distribution of the remaining (3) bundles is a normal distribution.
- No section of the remaining bundles is more or less susceptible to crack initiation or propagation.
- NDE resolution is accurate enough to categorize indications in 25% increments.
- A maximum of 15 pipes would be inspected.

The acceptance criteria established prior to the inspections was as follows:

- Pipes with identified thru-wall cracks would be replaced.
- With a 95% confidence level, there would be a 95% probability that no crack sizes existed in the remaining bundles such that, with design loads applied, crack size at the end of the next cycle would not exceed code allowable.
- If predicted crack growth exceeded code allowable but was less than thru-wall, further evaluation would be required.

Ultrasonic examinations of an initial 8 lines distributed between the remaining three quadrants yielded no reportable cracks or crack like indications. While these results were good news, it put in doubt one of the assumptions of the sampling plan i.e.,"the crack size distribution of the remaining 3 bundles is a normal distribution". Clearly, if cracks exist in these bundles, they are significantly less severe and if normally distributed, the distribution must be truncated.

None the less, in order to develop a reasonable inspection plan for the 128 tubes in the other bundles, it was conservatively assumed that they would also contain cracking similar to the SW bundle and would be normally distributed with the same general characteristics. This permits development of a sample size selection table which gives estimates of the maximum error on estimation of the mean as shown in Table 2. Using this table and MIL STD 105D for guidance, 13 tubes (a 10% sample) were selected. This sample size is consistent with a 95% confidence bound that the error on the prediction of the mean crack depth in the remaining tubes would be no greater than 15.7% of pipe wall thickness.

Consequently, five additional pipes were selected for inspection. These 5 pipes in the sample had no detectable cracks. This result provides further evidence that the other tube bundles do not exhibit a cracking problem of anywhere near the severity of the SW bundle.

The lines inspected in each of the three remaining quadrants are shown in Figures 5, 6, and 7.

## Repairs to SW Bundle

Based on the results of the UT inspections, metallurgical examinations and continuing lack of a definitive root cause, it was decided that all of the insert and withdraw line drywell penetrations in the SW bundle would be replaced. The replacement piping weld design is different from the original weld design. The original fillet welds attaching the CRD piping to the drywell shell were replaced with a sleeve that is fillet welded to the pipe at both ends. This pipe and sleeve assembly is then welded to the drywell shell with a partial penetration weld with a fillet cap.

This design was chosen after an evaluation of the pipe loading was conducted. Two types of loads which could cause significant stresses in the penetration and could have contributed to the fatigue cracking have been identified. These are thermal loads and mechanical loads. Data on pipe and process temperatures which cause thermal loads is fairly reliable and the stresses to the pipe caused by these temperatures can be predicted with a reasonable degree of confidence. (No thermal loads have as yet been identified which could have caused the observed problem.) Data on the mechanical loads that the penetration may have seen, however, is much less reliable and it is possible that the fatigue cracking could have been caused by mechanical loads of a magnitude and number of cycles greater than those currently identified.

Due to this uncertainly in the mechanical loading, fillet welding the sleeve to the pipe at both ends was chosen because its greater weld area has more capacity to carry potential mechanical loads. This design meets ASME Code requirements for the loadings specified in the design specifications for the CRD system. In addition, this design provides definite improvements over the current design for the following reasons:

- The joint is more flexible so stresses due to thermal loads and residual stresses from weld shrinkage will be less.
- Stresses in the sleeve-to-pipe weld due to mechanical loads will be less because the flexibility of the sleeve allows load from one side to be shared by both welds. The rigidity of the original design forced all the load from one side to be carried by the weld on that side. Further details on the SW bundle repair design are included in Reference 1.

## Root Cause Analysis

Initially, the root cause analysis of the SW bundle focused on possible weld defects and geometrical considerations. The metallurgical results, however, combined with further detailed fatigue analyses led to the conclusion that these defects and considerations alone (fabrication, etc.) were not sufficient to drive the lines to failure.

To enable all parties involved in the root cause determination to better define possible root causes and develop further actions, a Kepner-Tregoe problem analysis session was held on the subject of CRD insert/withdraw line cracking. The meeting included five representatives from Iowa Electric, three Nutech representatives with skills in metallurgical analysis, structural analysis and fracture mechanics and a number of General Electric personnel with metallurgical and structural experience, including GE's CRD System Engineer.

The methodology used was to define the known characteristics of the problem, and then compare these to suggested root causes. The two key characteristics noted were the exclusiveness of the problem to the SW bundle, and the striations seen during metallurgical analysis, which are indicative of high cycle fatigue. Many root cause suggestions were eliminated, or deemed less likely than previously thought, after comparing them to each characteristic of the problem.

The end result of this analysis was the conclusion that there is a forcing function ("driver"), currently unidentified, which is causing (or caused in the past) high cycle fatigue in the SW bundle. Possible sources of this forcing function are:

1. Drywell Shell Vibration
2. Hydrodynamic Loads
3. Thermal Loads
4. Localized Thermal Flexing of the Shell
5. Equipment Vibration on the CRD Pipes

There may also be additional factors involved which make the SW bundle (or potentially all bundles) susceptible (or more susceptible) to the driver's influence. These factors include:

1. Initial Construction Defects
2. Pipe Bending Before Startup
3. Environmentally-Induced Corrosion
4. Shell to Pipe Weld Geometry
5. Pre-Stress Due to Design

Further activities are required to identify the forcing function and clarify the role of the additional factors. Key among these activities is a walkdown of the CRD system to identify unique characteristics of the SW bundle and possible vibration sources, and monitoring of the CRD bundles and drywell shell for vibration and temperature during startup and on-line conditions. A more complete summary of this problem solving session is given in Reference 2.

<u>On-Line Monitoring Program</u>

To assist in the determination of the "driving" force necessary to have caused the SW bundle failures, an on-line monitoring program has been developed which will ultimately perform the following:

1) Monitor and record acceleration, strain and temperature for replacement and existing CRD piping.
2) Utilize the data obtained to perform fracture mechanics analysis to verify that the predicted conditions will not result in unacceptable consequences.
3) Evaluate load/temperature transients against system/plant conditions.

Specifically, strain and temperature sensors have been placed at or near the toe of the sleeve-to-pipe weld (in the drywell air gap) on the outside of the replacement piping for SW bundle lines 26-07I&W, 38-11I&W and 30-07I&W. Strain gauges and temperature sensors will also be affixed to this same piping at the exit point of the pipe from the sleeve in the concrete. This will provide a correlation between the readings which will help in the analysis of the remaining bundles. In addition, the SW bundle will be instrumented with drywell shell vibration monitors and high frequency sensors (acoustic monitoring) which can detect leakage from CRD lines in their vicinity.

The remaining bundles (NW, NE, SE) will each have, as a minimum, one set of CRD piping (i.e., one insert and one withdraw line) instrumented with strain and temperature sensors on the exit point from the sleeve in the concrete. Leak detection sensors will also be placed on the CRD piping outside of the concrete in the reactor building for these bundles. Each of these sensors will provide information to an on-line monitoring system which will store and analyze the data in a real-time domain.

Additional instrumentation may be used as necessary to better characterize the source of any identified load. These could include accelerometers and internal pressure sensor transducers (to monitor potential water hammer transients).

Due to the procurement and calibration requirements of the data acquisition and leak detection systems, a two phased approach will be used for their implementation.

Phase I will utilize a data acquisition system comprised of equipment capable of monitoring 21 channels of data. The monitored parameters will be chosen based on planned plant evolutions and those providing the most valuable information (i.e., temperature, strain, ect). The video probe will be used during startup to inspect several CRD penetrations in each quadrant to ensure no leaks have developed. In addition, the video probe will be available should the monitored data indicate the presence of unusual and significant driving forces. Phase I of the monitoring program will be implemented prior to start up from the Cycle 10/11 Refuel Outage.

Phase II of the monitoring program will consist of a PC-based, high speed, diagnostic and monitoring system capable of monitoring 82 channels of data. Leakage detection will be provided by a separate acoustic emission system capable of "listening" for leaks. Phase II of the monitoring program will be implemented prior to December 31, 1990.

This instrumentation will remain in place during Cycle 11 operation or until a root cause is identified. Use of this equipment beyond Cycle 11 will be based on the results of the on-line monitoring program and the status of our root cause analysis.

## Safety Evaluation

The root cause analysis depends, to a great extent, on information gathered by on-line monitoring of CRD system and vessel parameters. Although CRD failures are highly unlikely, they can not be precluded.

A safety analysis discussing the potential CRD failure mechanisms and their consequences has been prepared by General Electric and Iowa Electric (Reference 3). A detailed summary of this analysis is included in Appendix A.

## Conclusion

The investigation into the root cause of the CRD line failures continues to focus on the identification of a driving mechanism for the high cycle fatigue. To assist in this effort, an on-line monitoring program has been developed which will provide real-time analysis of strain, vibration and temperatures transients to which the pipes are exposed.

All CRD insert and withdraw line drywell penetrations in the SW bundle have been replaced with a two-sided pipe to sleeve weld design specifically engineered to better handle the identified mechanical loading on the CRD piping. One hundred percent of the lines in the remaining three bundles have been visually inspected for leakage; none was found. Ten percent of these lines have undergone ultrasonic testing and no indications of cracking were identified.

All credible failures of the CRD insert or withdraw lines have been reviewed and their consequences analyzed. This safety analysis together with industry experience of the leak-before-break phenomenon indicates that if leaks occur, they will be minor in nature and be of minimal safety significance. Control room personnel have been briefed on the operational anomalies that could result from leaking CRD insert and/or withdraw lines. If leaks are identified, the nature of the leakage will be evaluated and limitations on reactor operation considered as appropriate to the nature and amount of the leakage.

## References


| | |
|---|---|
| Reference 1 | DCP 1501, "CRD Line Repair" |
| Reference 2 | Summary of Kepner-Tregoe Study of CRD Insert/ Withdraw Line Indication Problem, DAEC-90-0611, August 13, 1990. |
| Reference 3 | General Electric Startup/Operational Assessment, "Degraded Control Rod Drive Insert and Withdraw Lines," August 22, 1990. |

**DESTRUCTIVE ANALYSIS FOR SW QUADRANT**

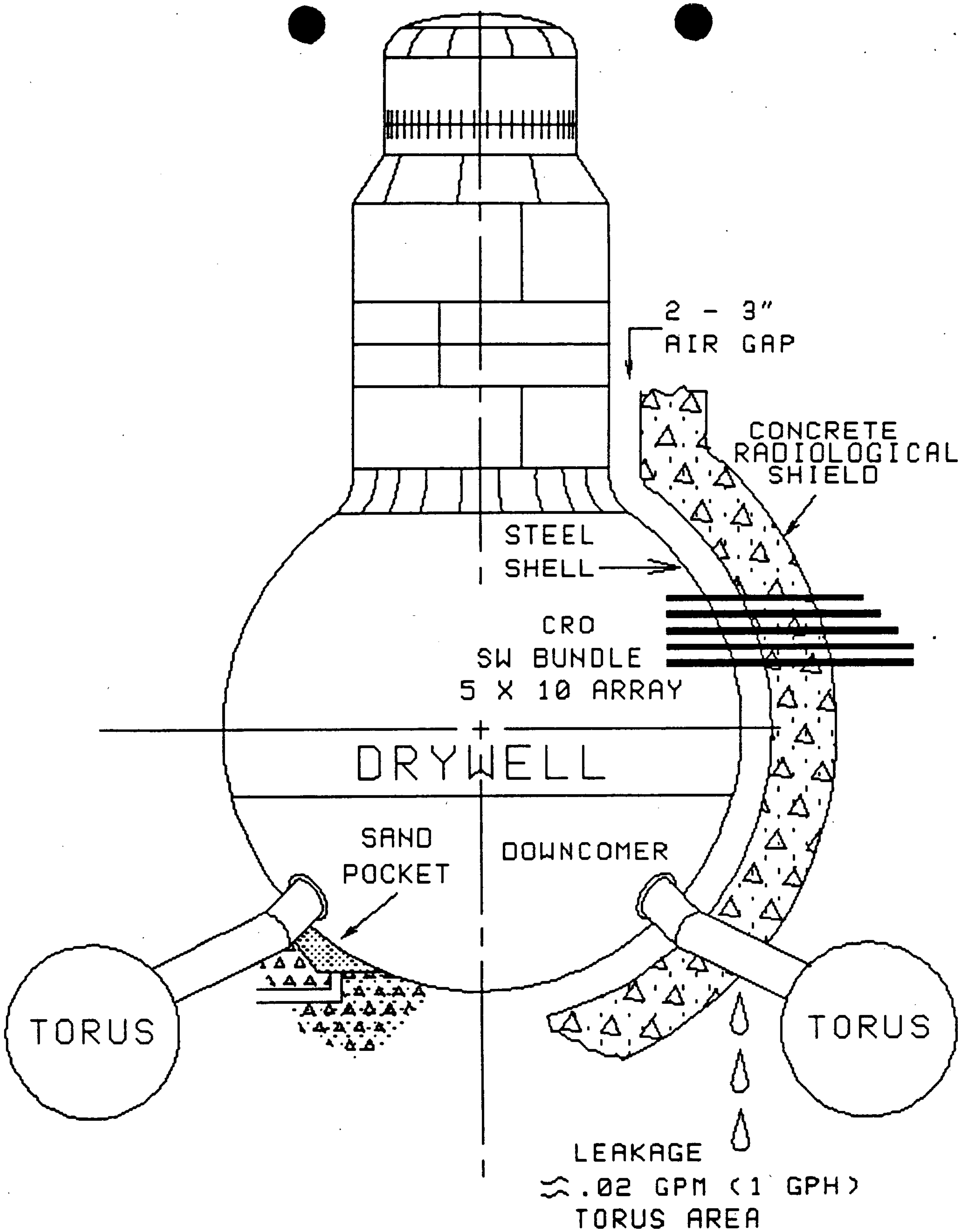
| Sample | Indications | Measured Depth |
|---|---|---|
| 30-07W | 2:00-7:30 | T.W |
| 30-07I | 2:30-5:00 | 40% |
| 26-07W | 1:00-7:00 | T.W |
| 26-07I | 12:00-6:00 | T.W |
| 34-07W | 2:00-4:00 | 72% |
| 34-07I | 3:00-4:00 | 10% |
| 22-03W | 2:30-3:30 | 24% |
| 22-11W | 2:00-4:00 | 40% |
| 26-15W | 2:00-3:30 | 20% |
| 34-15I | - | - |
| 26-19W | - | - |
| 22-19I | - | - |
| 34-19I | 2:00-4:00 | 31% |
| 38-19I | 3:30-4:30 | 7% |
| 26-23W | 2:00-4:00 | 15% |
| 38-23W | 2:30-3:30 | 12% |
| 34-23I | - | - |

TABLE 1

**SAMPLE SIZE SELECTION TABLE**

| Confidence | Z | No. of Samples 6 | 13 | 19 |
|---|---|---|---|---|
| 99% | 2.58 | 33.1 | 22.5 | 18.6 |
| 95% | 1.96 | 23.2 | 15.7 | 13.0 |
| 90% | 1.645 | 19.5 | 13.2 | 10.9 |

Table 2




DAEC CONTAINMENT

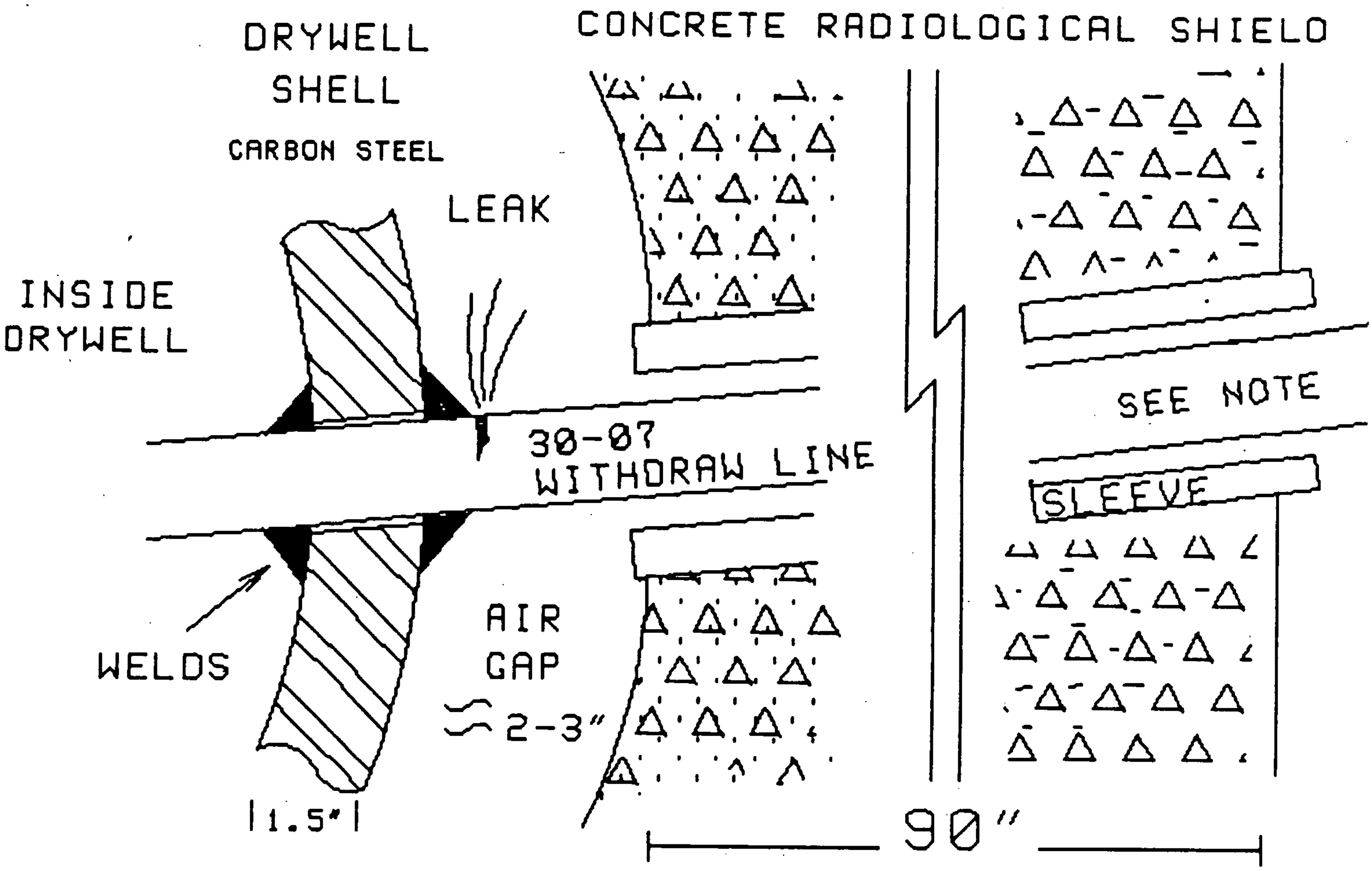
(NOT TO SCALE)



FIGURE 1

# 30-07 WITHDRAW LINE

## LEAKAGE OBSERVED AT POWER




THRU-WALL
INDICATION
NOT VISIBLE
UPON REMOVAL

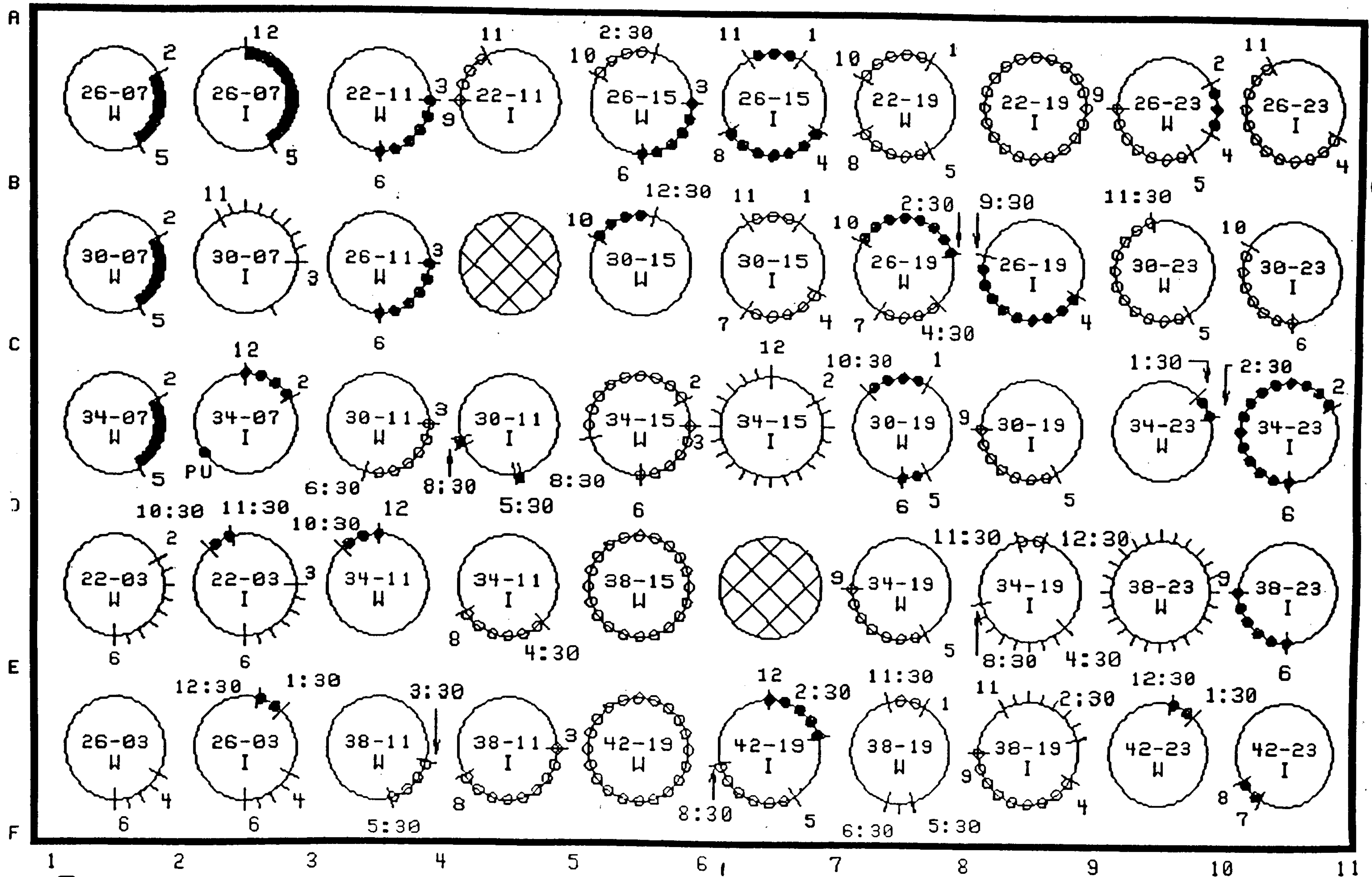
NOTE
1" NPS SCH 160
304 SS
.250 NOM. WALL



FIGURE 2

# CRD SOUTHWEST BUNDLE

## AIR GAP WELDS



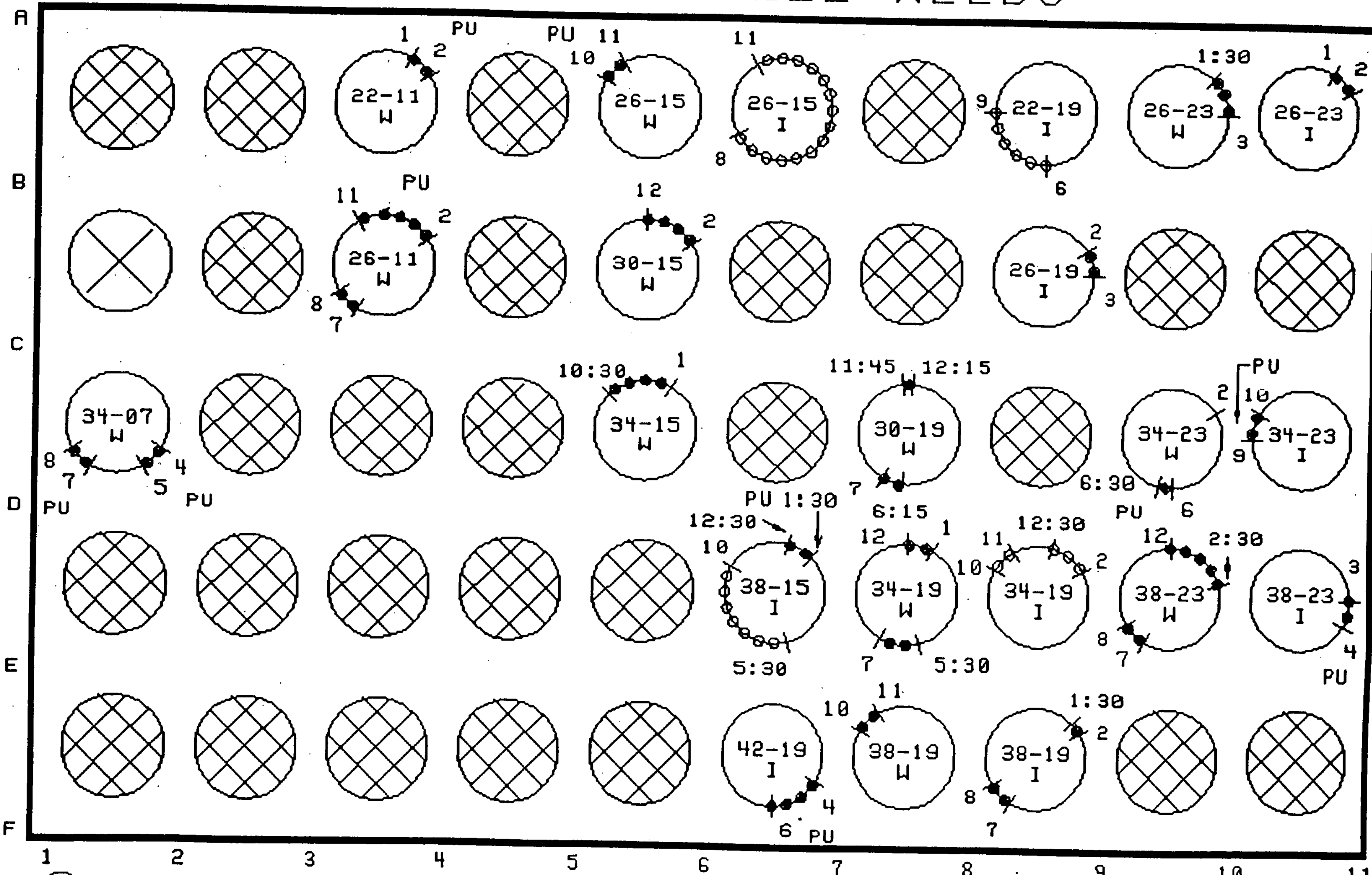

NO CRACKS , ●●● <1/3 WALL , ○○○ 1/3 TO 2/3 WALL , |||| >2/3 WALL . ▬ THRU-WALL

-Page 15-

FIGURE 3

# CRD SOUTHWEST BUNDLE

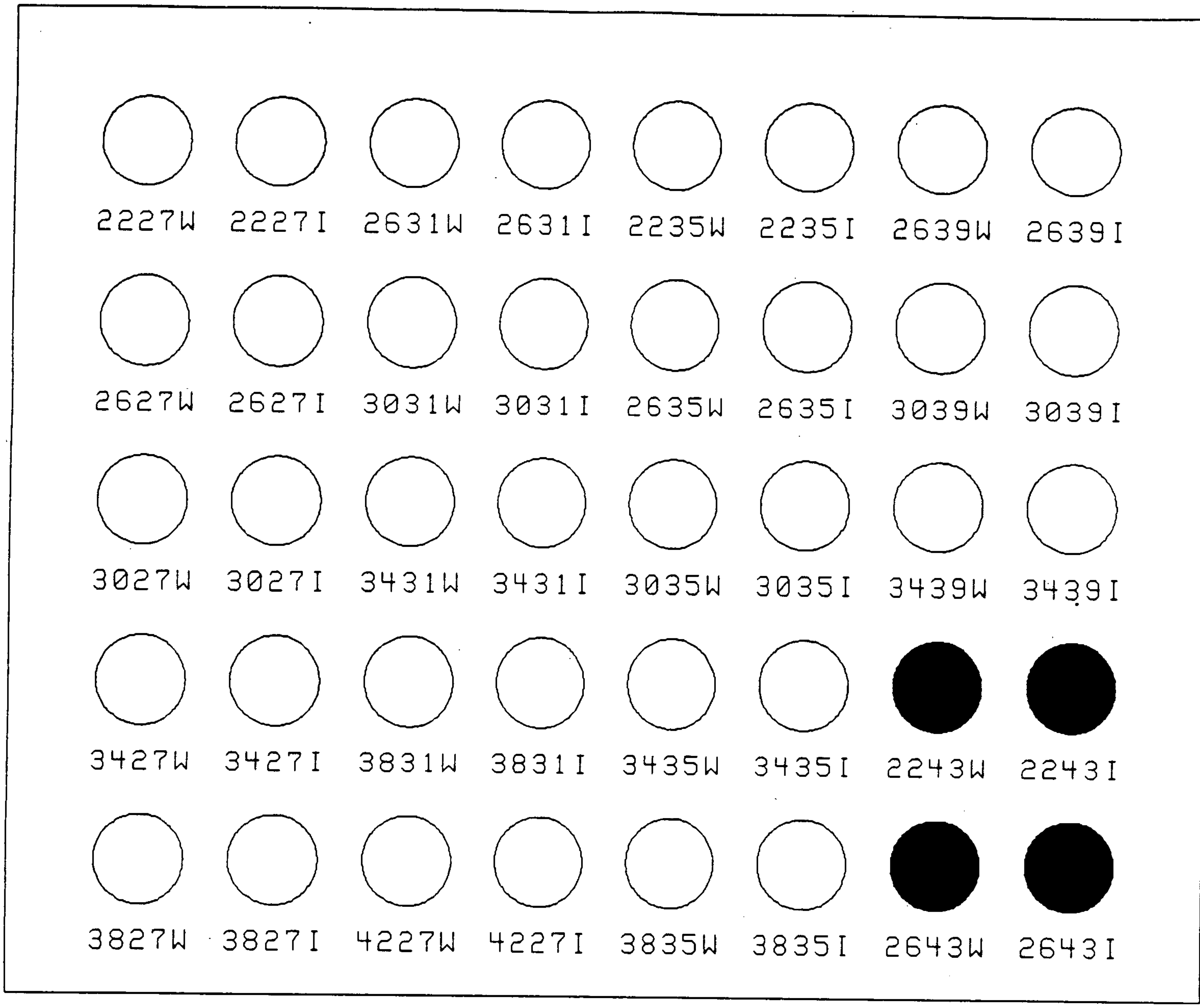
## INSIDE DRYWELL WELDS




NO CRACKS , ●●● <1/3 WALL , OOO 1/3 TO 2/3 WALL , |||| >2/3 WALL , ▬ THRU-WALL

PU = POSSIBLE UNDERCUT   X = NOT EXAMINED   NOTE: 1 CLOCK HOUR ...

FIGURE 4

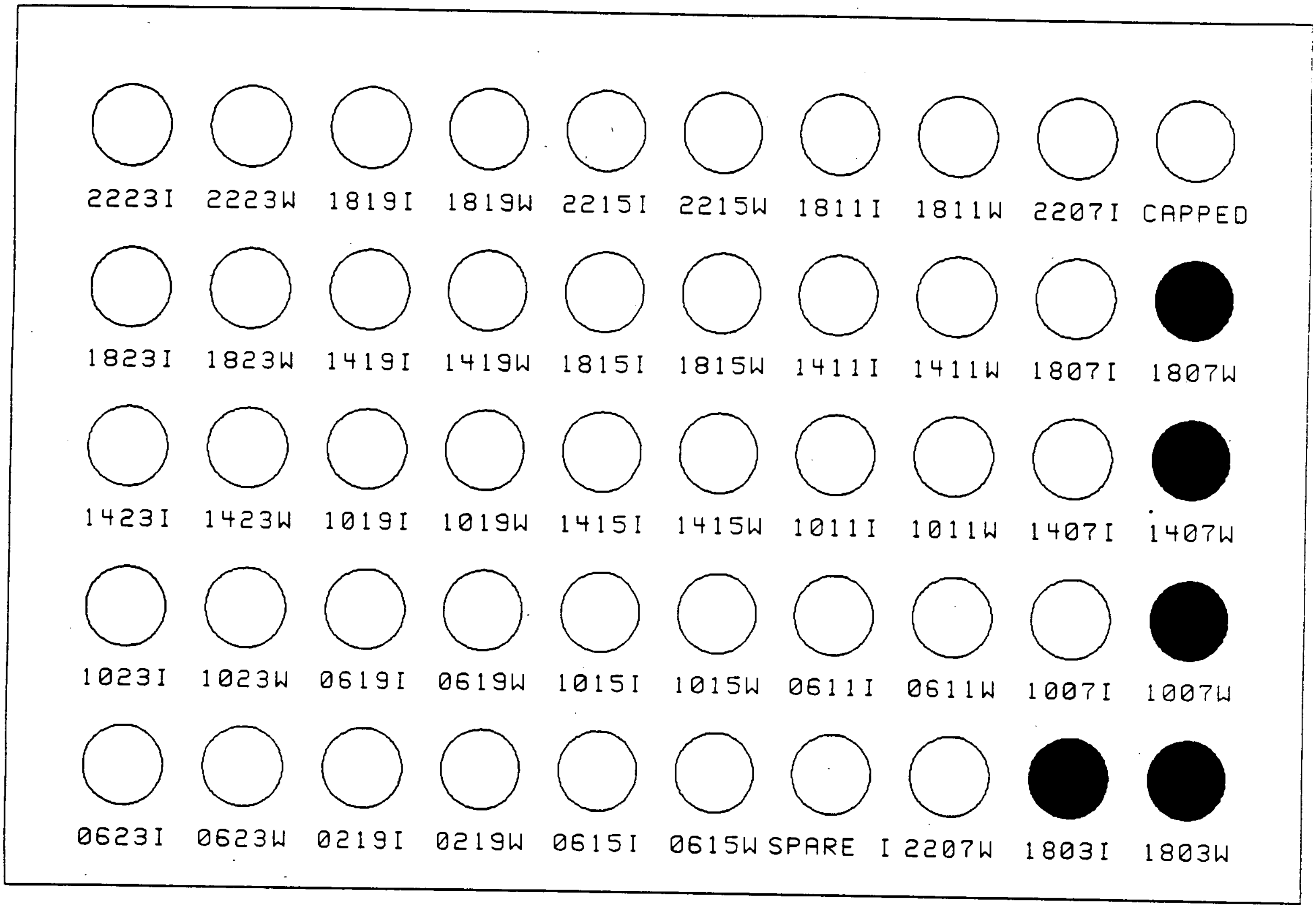



SOUTHEAST QUADRANT

● SELECTED FOR INSPECTION



FIGURE 5

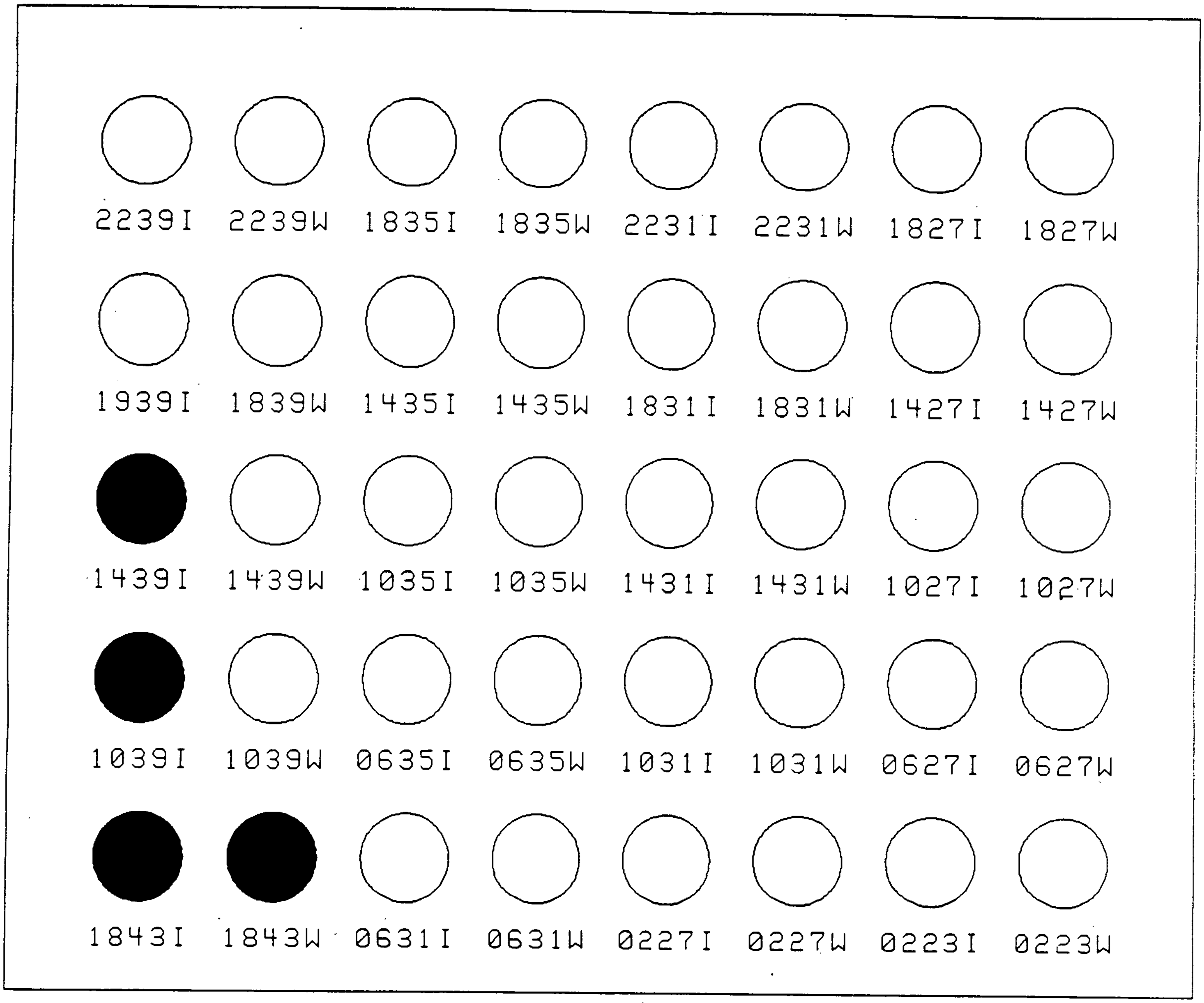



NORTHEAST QUADRANT

● SELECTED FOR INSPECTION



FIGURE 6




NORTHEAST QUADRANT

● SELECTED FOR INSPECTION



FIGURE 7

## SAFETY ANALYSIS

While the root cause of the CRD line failures is aggressively being pursued, no definitive load or cyclic duty events which could have caused the cracking have been identified. Therefore, while highly unlikely, it is impossible to preclude additional CRD line leaks.

The purpose of this safety analysis is to evaluate the consequences of CRD line leakage on 1) CRD system operation, 2) Containment Integrity, 3) Reactor Vessel Make-Up, 4) Reactivity Control and 5) Radiological Analyses.

It should be noted that postulated scenarios in this discussion are based on complete ruptures or gross leakage from the insert and withdraw lines. None of the following consequences could have occurred at the identified leak rate (1-2 gph).

### 1. CRD SYSTEM OPERATION

Per UFSAR Section 4.6.1.2, the CRD system controls gross changes in core reactivity by incrementally positioning neutron-absorbing control rods within the reactor core in response to manual control siqnals. It is also designed to quickly shutdown the reactor in emergency situations by rapidly inserting control rods into the core in response to a manual or automatic scram. As discussed in UFSAR Section 3.9.4, the safety basis of the CRD system is to insert the control rods with sufficient speed to limit/prevent fuel barrier damage due to any abnormal transient. Given the (limited) potential for leaking insert and/or withdraw lines, the ability of the CRD system to perform as designed is evaluated in the following paragraphs.

The consequences of a leaking CRD insert or withdraw line on CRD system operation are dependent upon crack size and reactor pressure conditions at the time of the failure. The operational anomalies that could occur are slow scram times, inadvertent scrams, insert/withdraw function difficulties, control rod drift into the core, and high CRD temperature.

Several breaks and combinations of breaks of the withdraw and insert lines at reactor power have been postulated and previously analyzed as documented in USFAR Section 4.6.2. This section states that there are three types of possible ruptures of CRD hydraulic lines - (1)insert (pressure-under) line, (2) withdraw (pressure-over) line, (3) coincident breakage of both.

If the insert line were to fail with the collet latched, the CRD could not be withdrawn since there would be no pressure differential across the collet piston and therefore no tendency to unlatch the collet. If the CRD is being withdrawn when the insert line breaks, the hydraulic force would be insufficient to hold the collet open and the normal force of the collet springs would cause the collet to latch and stop rod withdrawal. The ball check valve is designed to seal off a broken insert line by using reactor pressure to shift the ball to its upper seat. However, the cooling water to that drive would be lost which will shorten the lifetime of the seals. The CRD would still be inserted by reactor pressure on a scram signal, however, the drive could be damaged due to exposure to high temperatures. If the ball were prevented from seating, reactor water would leak to the atmosphere in addition to the loss of cooling water to the drive. Annunciation of high drive temperature is provided in the Control Room. It should be noted that actual operating experience, both at the DAEC and at other BWR's, demonstrate that the 7/8" ball check valve is highly reliable.

For the case of a withdraw line break, pressure over the drive piston would drop from reactor pressure to atmospheric pressure. Any significant reactor pressure (or CRD pressure) would act on the bottom of the drive piston and fully insert the CRD. The drive would be inserted at a speed dependent on the reactor pressure. Leakage would eventually occur past the seals. This leak rate is estimated to be 1 to 3 gpm, but will not exceed 10 gpm. Indication of drift alarm and light, fully inserted drive and high drive temperature are provided in the Control Room.

For the case of coincident insert and withdraw line breakage, pressure above the drive piston would drop to atmospheric, and reactor pressure will seat the ball check valve (isolating the broken insert line) and pressurize the area under the drive piston. The drive would be inserted at a speed dependent on the reactor pressure.

A more detailed discussion of the consequences of CRD line failures is contained below.

<u>Scram</u>

Failure of the withdraw line has no effect on the ability to insert the rod during scram, as the withdraw line acts to direct the scram exhaust water to the scram discharge volume. A leaking withdraw line will still allow the exhaust function to succeed during a scram, and the motive force provided by reactor pressure or the Hydraulic Control Unit (HCU) scram accumulator will scram the rod.

Failure of the insert line has no adverse affect on the ability to scram the control rod during operation at high reactor pressure (>500 psig) because the CRD internal ball check valve feature allows the CRD to scram utilizing reactor pressure only. However at low reactor pressure (<450 psig), a scram using reactor vessel pressure as the only motive force, i.e. no credit for the HCU accumulator, may not occur. A minimum reactor pressure of approximately 500 psig is required to provide the motive force to reliably scram the rod to position 04 within the maximum Technical Specification scram time of 7 seconds (reference Technical Specification Section 3.3.C.3). Control rod insertion can be achieved with reactor pressure between 450 psig and 500 psig, but scram times may be significantly greater than 7 seconds.

## Rod Drift

A leaking withdraw line may depressurize the area above the drive piston. Hence a leaking or broken withdraw line may cause partially or fully withdrawn control rods to inadvertently drift into the core if reactor pressure is above 450 psig. With the collet latched, calculations indicate that a minimum leakage of 3.3 gpm through the leaking withdraw line is required to initiate a control rod drift. In the event of a rod drift, the rod drift alarm will annunciate in the Control Room. This provides indication to the operators that there is a potential problem with an individual CRD, and action can be taken to determine if the rod drift is a result of a leaking withdraw line. Rod drift will not occur for a leaking insert line due to the presence of the collet mechanism.

## Rod Insertion and Withdrawal

CRD single-notch withdrawal difficulties could be exhibited with excessive leakage through the insert and/or withdraw lines. Withdrawal line leakage in excess of 3 gpm could result in failure of the CRD to single-notch withdraw because of losses in the drive-down pressure required to unlock the CRD collet. However, leak-before-break would indicate that new instances of thru-wall cracks will begin as small leaks. (The leakage from 30-07W was approximated at 1 to 2 gallons per <u>hour</u>.) Therefore, as our experience indicates, leakage would be identified well below the 3 gpm necessary to cause operational problems with a CRD.

Insert line leakage in excess of 0.8 gpm could result in double-notching during a single-notch withdrawal. Insert line leakage greater than 1.5 gpm would result in failure of the CRD to single-notch insert or single notch withdraw. Single notch errors, however, are not considered to be significant. The reasoning provided above for withdraw line leakage, is also valid for insert leakage. We should be able to identify leakage well below the calculated values which could cause notching concerns. It should also be noted that with the leakage rates discussed above, CRD continuous rod insertion in the normal or emergency mode will not be impaired.

In the extreme case of a ruptured insert line, the CRD ball check valve will seal off the broken line by using reactor pressure to shift the ball to its upper seat. If the collet were latched (CRD at rest), no control rod withdrawal would occur. If the insert line were to fail while the control rod is being withdrawn, CRD withdrawal velocity would increase (because of the reduction in pressure under the piston), with a corresponding drop in drive down pressure. However, the hydraulic force would not be sufficient to hold the collet open and the spring force would cause the collet to latch and stop rod withdrawal. If the reactor were initially at 1000 psig, and the collet were to remain open (highly unlikely), calculations indicate that the steady state control rod withdrawal velocity would be approximately 2 ft/sec (less than CRDA assumptions).

In addition, the Rod Worth Minimizer (RWM) System supplements procedural requirements to aid in the enforcement of control rod withdrawal sequences. As described in USFAR Section 7.7.4.7.1, the RWM ensures that established startup, shutdown and low power level control rod procedures are followed.

The RWM microcomputer prevents the operator from establishing control rod patterns that are not consistent with those used in the Control Rod Drop Accident analysis by initiating appropriate rod withdrawal block and rod insert block interlock signals to the Reactor Manual Control System rod block circuitry. Per Technical Specifications Section 3.3.B.3.b, if the reactor is in the Startup or Run modes below 30% rated power, the RWM shall be operable or a second Reactor Operator shall verify that the Reactor Operator at the reactor console is following the control rod pattern. Availability of the RWM system (or second licensed operator) provides positive assurance that notch errors are promptly identified, even those that may be due to rod drift or double notching caused by CRD line leakage.

Also, control rod position information is obtained from reed switches in the control rod drive that open or close during rod movement. This is part of the Control Rod Information Display described in UFSAR Section 7.7.3.8. Both a rod selected for movement and the rods not selected for movement are monitored for drift. A drifting rod is indicated by an alarm in the control room. This provides another mechanism for monitoring for single-notch errors. The operators will be instructed that if notching problems do occur during rod manipulations, as indicated by the RWM system or the reed switch alarms, this may be an indication of a leaking CRD line.

CRD Temperature

Excessive leakage through insert and/or withdrawal lines could cause higher than normal CRD temperatures. Annunciation of high CRD temperature (>250° F alarm setpoint) is provided in the Control Room and would be an indication of possible CRD insert and/or withdraw line leakage. Leakage through the withdraw line would continuously draw reactor water through the CRD. An estimated withdraw line leakage of 0.2-0.4 gpm would increase the CRD temperature above the 250° F setpoint.

Failure of or excessive leakage in the insert line would prevent cooling water from entering the CRD, and thus would increase CRD temperature. An insert line leakage of 7 gpm would increase the CRD temperature above the 250° F alarm setpoint. A total system leakage of 20 gpm from several broken/leaking insert lines would decrease the cooling header pressure and reduce cooling flow to all CRDs. In this situation, many CRDs may exhibit temperatures above 250° F. Prolonged CRD exposure to temperatures at or near reactor temperature could lead to deterioration of the graphitar seals. However, the CRD safety function (scram) would not be impaired by the higher than normal operating temperatures and it is highly improbable that a total system leakage greater of 20 gpm or greater could occur.

2. CONTAINMENT INTEGRITY

The effects of a release of high temperature CRD water from a leaking line on the integrity of the drywell shell are evaluated below.

The total cooling flow is approximately 40 gpm or an average of 0.45 gpm per CRD. As previously mentioned, the CRD ball check valve would seal off a broken insert line (i.e. prevent reactor water discharge flow). Hence, a leaking or broken insert line would discharge cooling water flow at the temperature of the CRD system. However, since there is no ball check valve in the CRD withdrawal port, the amount of leakage from a ruptured withdraw line would depend on line crack size and the condition of the CRD internal seals. Assuming a worst-case withdraw line failure at 1000 psi reactor pressure, the leakage rate would be approximately 1 to 5.6 gpm. This flow rate is similar to the post scram flow into the SDV. However, with severely degraded graphitar stop piston seals, the leakage rate could be as high as 10 gpm. The discharge leakage temperature could range from CRD system temperature up to reactor vessel temperature depending upon the severity of the leakage and the condition of the seals. Normally, discharge temperature through a completely broken withdraw line would be about 280° F, which is similar to the post scram condition. Discharge leakage less than cooling water flow (0.2-0.4 gpm) would yield leakage temperatures of approximately CRD system temperature. However, temperatures greater than 280° F are possible for pressurized withdraw line conditions with leakage flow rates greater than cooling flow.

Primary Containment System design parameters are defined in UFSAR Table 6.2-1. The design temperature of the drywell is 281° F. As described above, the temperature of the CRD water leaking in the air could be as high as 500°F. However, the drywell shell will only be exposed to this temperature in a localized area of the CRD piping bundles. Given the thickness and large surface area of the drywell shell and the high thermal conductivity of the carbon steel shell material, the heat transfer would be sufficient to prevent the shell temperature from reaching the design temperature of 281° F. Therefore, no threat exists to containment shell integrity due to exposure of the shell to water from any of the CRD withdraw and/or insert lines.

## 3. REACTOR VESSEL MAKEUP

Loss of water inventory through leaking CRD lines must also be considered. The CRD System cooling water flow rate of 40 gpm and the normal RPV level control (feedwater system) would provide make-up for the maximum leakage thru a number of leaking or broken CRD withdraw lines. In addition, the Reactor Core Isolation Cooling (RCIC) System at DAEC has a 400 gpm capacity and could thus handle the maximum postulated leakage from up to 40 CRD withdraw lines.

In the most extreme case of complete failure of all CRD withdraw lines, a maximum leakage flow of 890 gpm could exist. Leakage of this magnitude could easily be made up by either the normal excess capability of the feedwater system or by operation of the High Pressure Coolant Injection (HPCI) System. The HPCI System at DAEC has a 3000 gpm capacity which substantially exceeds the leakage rate postulated above. In the event that neither of these systems remains available, the reactor vessel could be depressurized via the Automatic Depressurization System (ADS) and adequate makeup provided by any of the low pressure Emergency Core Cooling System (ECCS) pumps.

Therefore, even with catastrophic failure of all the CRD lines, and it should be noted that this event is not postulated to occur, makeup capacity is available through normal and emergency system flowpaths.

## 4. REACTIVITY CONTROL

General Electric has evaluated CRD operational anomalies relative to their effect on reactivity control and fuel design limits. Their evaluation was performed using available generic analyses that are applicable to the DAEC.

The evaluations are separated into the Startup Range (low reactor pressure and power) and the Power Operation Range (high pressure and power) because of the potential difference in system response under these conditions.

The transients and accidents that could potentially be affected in the startup range are:

- Continuous Control Rod Withdrawal Error During Startup (RWE)
- Control Rod Drop Accident (CRDA).

Each of these situations has been evaluated relative to the potential failure modes of the CRD system during startup which includes: failure to scram, rod drift (insertion direction), fast control rod withdrawal (approximately 2 fps) and multiple notching (during insertion or withdrawal). These failure modes have been evaluated relative to their impact on reactivity control, fuel limits during startup and the capability to bring the plant to cold shutdown.

### Rod Withdrawal Error (RWE) During Startup

During startup, the reactor operator follows prescribed control rod withdrawal sequences designed to minimize the control rod worth. The Rod Worth Minimizer (RWM) monitors control rod movement and will provide control rod blocks should the operator deviate from the prescribed sequence. The Rod Sequence Control System (RSCS) performs the same basic function as the RWM and will also provide a control rod block. (We have submitted our application to the NRC to remove the RSCS. Consequently, we have not taken credit for the RSCS in this discussion.)

If a control rod is withdrawn out-of-sequence, the potential exists to have a reactivity excursion. The Intermediate Range Monitors (IRM) will initiate a control rod block and then a reactor scram on high neutron flux and the subsequent control rod insertion will terminate the event. Analyses have shown that if the control rod that is the "withdraw error" fails to scram, the consequences can potentially exceed the 170 cal/gm limit that is applicable to this event during startup (UFSAR Section 15.4.7.5.). Failure of a single control rod to scram other than the error rod has not been evaluated in the past. However, if the control rod that fails to scram is separated from the error rod by at least two four bundle control cells the effect on the event is expected to be minimal. Given the margin to fuel thermal limits for this event, it is therefore concluded that if the control rod that fails to scram is sufficiently separated from the error rod, the consequences will be within limits. Should multiple control rods fail to scram during the RWE during startup, not only would the scram reactivity be degraded but the probability that one of these failed rods is adjacent to the error rod is also increased. It is expected that multiple scram failures could also lead to a potential violation of the 170 cal/gm limit. However, the probability that several insert lines would completely rupture simultaneously during a startup is extremely remote. As stated previously, our experience indicates that we will detect any leakage long before it reaches a rate where the control rod scram function could be precluded.

A control rod drift in the insert direction will not effect the RWE event during startup since the insertion of the control rod will be in the direction of reducing core reactivity. The effect on flux distribution from a single control rod drifting in is also negligible based on evaluations of inserted inoperable control rods during the Control Rod Drop Accident (CRDA).

If a complete insert line failure occurs and the collet fails to latch in the control rod that is being withdrawn, which results in an increased withdrawal speed, the consequences of a RWE event during startup (assuming the RWM and IRM control rod blocks do not block further withdrawal) would be more severe. Sensitivity studies have shown, however, that increased speeds up to 1 foot per second (fps) will not result in violating the 170 cal/gm peak fuel enthalpy limit. If the withdrawal speed were to increase further, the consequences of the event could potentially exceed the 170 cal/gm limit but would be bounded by the consequences of the CRDA (i.e. less than the 280 cal/gm limit) up to a velocity of 3.11 fps. Withdrawal velocities greater than 3.11 fps are not considered credible due to the design of the control rod velocity limiter. In addition, the probability of these simultaneous events occurring is remote.

Multiple notching results in a control rod being withdrawn more notches than planned by a reactor operator. This condition is bounded by the consequences of the RWE during startup which assumes the complete withdrawal of an out-of-sequence control rod. As noted earlier, notching problems are not an operational concern.

In summary, a CRD line rupture coincident with a RWE during startup has the potential to cause a reactivity excursion event. However, the probability of a RWE and CRD line rupture(s) is extremely remote. In addition, each of the CRD bundles will be monitored for stresses and inspected with the video probe during startup.

Also, the RWM will be available for use during startup. The RWM restricts withdrawals and insertions of control rods to prespecified sequences. Use of this system will help to prevent RWE or will be another indication available to the operator of the potential for leakage from the CRD lines.

Control Rod Drop Accident (CRDA) During Startup

As described in UFSAR Section 15.4.7, the CRDA is the Design Basis Accident for the consequences of a reactivity excursion. The CRDA occurs when a control rod becomes decoupled from its drive, becomes stuck in the core and then drops to the position of its withdrawn drive. The ensuing reactivity excursion is terminated initially by Doppler reactivity feedback and then by scram and void reactivity feedback. The scram is initiated by high neutron flux sensed by the IRMs or APRMs. The dropped control rod is assumed to not scram.

The failure of an additional single control rod to scram (due to an insert line failure) during a CRDA during startup will have negligible impact on the CRDA consequences since the scram only affects the last portion of the event and, therefore, is only important in terminating the final 30% of the energy generated and deposited in the fuel by delayed neutrons. The failure of a single control rod to scram will not significantly affect this analysis. However, the failure of multiple control rods to scram could result in an increase in the consequences of the CRDA and potentially result in exceeding the 280 cal/gm limit. However, given the very low probability of a CRDA coincident with simultaneous ruptures of multiple insert lines, this scenario presents an insignificant risk.

Control rod drifting in the insert direction can potentially result in control rods not in compliance with the required control rod withdrawal sequences assumed in the CRDA analysis. A single control rod drift in the insert direction will have a negligible effect on the consequences of a CRDA. Up to eight control rods can drift in the insert direction and still remain bounded by generic CRDA analysis provided the control rods are separated by at least two cells in each direction.

Multiple notching during control rod withdrawals could result in control rods being positioned beyond their allowed limits in the sequences designed to mitigate the consequences of the CRDA. However, analyses have demonstrated that the effects of a "double notch" (i.e., the control rod withdraws an extra notch) are bounded by generic CRDA analysis. Additional notch errors could result in control rod patterns that are not covered by the generic CRDA analysis and would be promptly corrected. The impact of multiple notching in the insert direction is bounded by the consequences discussed for rod drifts in the insert direction.

Shutdown Margin

The capability to achieve cold shutdown is dependent on the number and distribution of control rods which can be inserted. The failure of a single control rod to scram will still result in reaching subcritical conditions by design. However, the additional single failure of an adjacent control rod (and in some cases a diagonally adjacent control rod) will in general result in insufficient shutdown margin to satisfy Technical Specification requirements except when the control rods in question are located on the periphery of the core. However, as noted previously, scram will still occur at pressures greater than 450 psig even with a completely ruptured insert line. The probability of several insert lines rupturing at less than 450 psig is unlikely.

Reactivity Control During Power Operation

The transient that could potentially be affected in the power operation range is the Rod Withdrawal Error at power (RWE). Additionally, the impact on thermal limits during inadvertent control rod motion has been evaluated. Each of these situations has been reviewed relative to potential failure modes of the CRD system at power. These are rod drift (insert direction), fast control rod withdrawal (approximately 2 fps) and multiple notching (during insertion or withdrawal).

At power conditions (>25% of rated thermal power),the RWE assumes that an operator continuously withdraws a control rod. As local power increases, the Rod Block Monitor (RBM) will provide an alarm and a control rod withdrawal block if local power increases beyond the specified setpoints. The setpoints are chosen to ensure that if the event starts from the Technical Specification power distribution limits for Minimum Critical Power Ratio (MCPR) and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR), the subsequent RWE will not violate the fuel design or safety limits.

A control rod drift in the insert direction will not affect the RWE at power since the insertion of the control rod will be in the direction of reducing core thermal power. The affect on power distribution from a single control rod drifting in is also negligible based on evaluations of single control rod scrams.

If an insert line rupture occurs in a control rod that is being withdrawn, which results in an increased withdrawal speed, the consequences of a RWE at power may be more severe, depending on the speed of the withdrawal. The potential impact is that the control rod could travel further than analyzed following initiation of the control rod block from the RBM. This can increase the change in MCPR and MAPLHGR during the event. If the control rod withdrawal speed increases such that only a single additional notch of withdrawal is possible, the consequences of the RWE at power are bounded by the generic analysis which provides appropriate levels of conservatism to account for the potential additional notch withdrawal. If the failure in the CRD system is such that the control rod can be withdrawn several notches beyond the blocked position, it is possible that the Safety Limit MCPR could be violated. However, this is considered highly unlikely given the conservatism in the RBM setpoint determination, the margin to thermal limits during normal operation and the extremely low probability of a complete failure of an insert line on the rod being withdrawn. As the speed of the withdrawn control rod increases (in the range of feet per second), the event is bounded by the CRDA at power. Generic analyses have demonstrated that a CRDA at power levels above 10% of rated thermal power will not violate the 280 cal/gm limit.

Power Distribution

The reactor is normally operated with symmetrical control rod patterns and maintained within the power distribution limits specified in the Technical Specifications. The inadvertent drift of a single control rod in the insert direction can potentially result in a reduction in operating margin to these power distribution limits. As the control rod drifts into the core, the power distribution may be shifted such that increased peaking occurs near the tip of the control rod. Depending on the location of the control rod, total core thermal power may be reduced during the control rod insertion. These two effects tend to compensate, and the resultant impact on power distribution limits is very small. The reduction is not sufficient to violate the fuel design or safety limits.

For multiple, simultaneous control rod drifts, the effects on power distribution and total core thermal power will be increased, but with the same compensating effects. Therefore, even the simultaneous drifting of multiple control rods in the insert direction would not be expected to violate fuel design or safety limits.

A variety of monitoring methods exist which will provide information to the operator on rod position and status. This includes drive temperature annunciation in the Control Room providing indication of potential CRD withdraw and insert line leakage. Also, if complete failure of either line would occur, the rod would either be inserted automatically or could be inserted on a scram signal. Therefore, there is no adverse effect on the ability of the individual rods or all rods to safely scram.

## 5. RADIOLOGICAL ASSESSMENT

The radiological implications of CRD withdraw and insert lines leaking into the drywell air gap must be reviewed and addressed. GE performed an analysis of the potential off-site consequences of airborne releases due to assumed leakage from a CRD line during both normal operation and accident scenarios. They assumed that the airborne source term would be dominated by iodine. And for that reason, only thyroid dose evaluations were performed. Inhalation or ingestion of radioiodine will result in ultimate deposition of the material in the thyroid gland because of natural metabolic processes. This, of course, presumes that such exposure pathways actually exist. Ingestion, rather than inhalation, is the controlling path for thyroid exposure from radioiodine because of concentration effects unique to the air-vegetation-cow-milk-infant food chain.

The failures of concern are complete breaks of the CRD withdraw lines outside the containment. This scenario results in the maximum unisolable leakage.

In the event of a complete withdraw line failure, the estimated leak rate at 1000 psig reactor pressure is 1 to 3 gpm per UFSAR Section 4.6.2.2. This 1 to 3 gpm leakage would be past the stop piston seals. The leak rate could reach as high as 10 gpm with the stop piston seals severely degraded. Per UFSAR Section 11.1.1, the normal expected leak rate of primary coolant into the Reactor Building is stated as 1 gpm for use in the source term calculation for normal effluent releases. The leakage from the CRD lines would be in addition to the normal rate. Leakage from the CRD lines would be outside of primary containment and would eventually be collected by the Reactor Building floor drain sumps and then processed by the plant radwaste system.

Since it is Iowa Electric's policy not to release processed radwaste liquid, even if within release limits, no liquid release to the environment would occur due to a CRD line failure. Any gaseous release resulting from a line failure would be exhausted from the Reactor Building or Radwaste Building through the Reactor Building exhaust. The Reactor Building exhaust is continually monitored for radioactivity by an effluent radiation monitoring system installed in the Reactor Building vent shaft. The following provides GE's evaluation of potential normal releases outside of containment due to leaking CRD lines.

### Release During Normal Operation

An annual thyroid dose objective of 15 mrem to any individual is specified to meet the goal of 10CFR50 Appendix I (Section 3.15.D of Technical Specifications). In Section 11A.3.3.5 of the UFSAR, the objective annual, average, airborne I-131 concentration at the limiting offsite location is 1.43 E-14 μCi/cc for thyroid dose via the milk food chain. The limiting offsite iodine concentration occurs at a dairy farm in the WNW sector, about 2500 meters from the building exhaust. The upper bound impact of a leaking CRD line at the limiting dairy farm was calculated using the following assumptions:

1) A CRD line is leaking at the maximum 10 gpm.
2) The reactor water iodine concentration is at the Technical Specification limit of 1.2 μCi/gm dose-equivalent I-131. (Technical Specification Section 3.6.B.1)
3) The leaking water is at reactor water temperature. The fraction of leaking water which flashes to steam and becomes airborne is 40%
4) Half of the airborne iodine is removed by plateout, condensation, etc.
5) During normal operation, the Reactor Building ventilation effluent is discharged to the environment unfiltered. Therefore, the iodine releases to the environment for this event are also assumed to be unfiltered. Based on UFSAR Section 11A.3.3.5, for annual average meteorology, the ratio of concentration at the dairy farm to iodine release rate is (1.43 E-14μCi/cc)/(0.044 μCi/sec) = 3.25 E-13 μCi/cc per μCi/sec.

Based on the stated assumptions, the calculated airborne iodine release rate is 151 μCi/sec dose-equivalent I-131. The concentration at the limiting dairy farm would be 4.9 E-11 μCi/cc. Exposure of approximately 2 1/2 hours at this concentration would be sufficient to exceed the design objective annual average dose of 15 mrem per 10 CFR50 Appendix I.

10CFR20 Thyroid Dose Rate Limit

Technical Specification 3.15.B.2 implements the requirements of 10CFR20 by imposing a limit of 1500 mrem/year on the thyroid dose rate in the unrestricted area. The Offsite Dose Assessment Manual (ODAM) specifies calculation of the dose rate for the worst sector on the site boundary using a reference X/Q value of 4.3 E-6 sec/m(3). Use of that X/Q value and assumptions (1) through (4) above results in a calculated thyroid dose rate of 7100 mrem/year. This is approximately five times the limiting Technical Specification dose rate of 1500 mrem/year.

The relationship between iodine concentration in the coolant, the leakage flow rate and the time to reach Appendix I dose criteria is linear. Therefore, the off-site dose varies linearly with concentration, leak rate, and duration of the release.

In GE's analysis above, the Technical Specification limit of I-131, which is 1.2 μCi/gm, was used. However, the surveillance requirement in Technical Specification Section 4.6.B.1.C, states that during startup, if the I-131 is greater than or equal to 0.012 μCi/gm of dose equivalent I-131, then a coolant chemistry sample will be taken and analyzed. In addition, Section 4.6.B.1.d of the Technical Specifications states that when the equilibrium iodine value is greater than or equal to 0.012 μCi/gm of dose equivalent I-131 and the gaseous waste monitor located prior to holdup indicates an increase of greater than 50% in the steady state fission gas release, after factoring out increases due to power changes, a sample of reactor coolant shall be taken and a specific determination shall be made for I-131 dose equivalent for the iodine mixture.

Revising the reactor water iodine concentration used in the above GE analysis to the surveillance limit of 0.012 μCi/gm and still assuming a worst-case 10 gpm leak, the time to reach the 10CFR50 Appendix I limits would be increased by a factor of 100, or 250 hours (10.4 days).

Per Technical Specification Section 4.15.A.1, determination of effluent I-131 concentration is performed weekly. Since I-131 analyses is performed every 7 days for the effluent from the reactor building, a maximum of 0.012 μCi/gm reactor water iodine concentration is an acceptable limit at which to maintain coolant chemistry iodine concentration. This is because with an assumed 10 gpm of CRD water leaking in the air gap region with a concentration of 0.012 μCi/gm, it will take 10.4 days before the Appendix I annual limit of 15 mrem of I-131 to any organ is exceeded. Since the Reactor Building effluent is analyzed for iodine concentration on a 7-day basis, high iodine levels in the effluent would be identified prior to exceeding Appendix I limits and corrective action could be taken. (It should be noted that this radioactivity evaluation takes no credit for discovering leakage early.)

For the 10CFR Part 20 limits, reducing the coolant iodine concentration by a factor of 100 will result in a reduction in thyroid dose rate of 100, due to the linear relationship. This results in a thyroid dose rate of 71 mrem/year, which is well below the Technical Specifications 3.15.B.2 limit of 1500 mrem/year.

<u>Accident Consequences</u>

In this analysis, GE considers potential off-site radiological consequences from assumed accident scenarios. A Loss-Of-Coolant Accident (LOCA) was assumed to occur at a time when there is existing leakage from a CRD line outside primary containment. The offsite dose increment due to the additional containment leakage path was evaluated for the following assumptions:

1) The CRD line leaks at 3 gpm after the vessel is depressurized by the LOCA.
2) The Regulatory Guide 1.7 iodine source term (50% of the core iodine) is mixed in the combined reactor and suppression pool water inventory.
3) The fraction of iodine in the leaking coolant which becomes airborne is 10% in accordance with Standard Review Plan (SRP) 15.6.5 Appendix B.
4) The release to the environment is via the 99% iodine efficient SGTS filters and the 100m stack.
5) Regulatory Guide 1.3 meteorology with fumigation for the first 30 minutes is used.

The calculated thyroid inhalation dose from this pathway only is 0.9 rem at the Exclusion Area Boundary (EAB) (2 hour dose) and 1.0 rem at the Low Population Zone (LPZ) (30 day dose). Table 15.10-1 of the UFSAR shows doses of 29 rem and 24 rem, respectively for doses at the EAB and LPZ with Reg. Guide 1.3 LOCA assumptions. Consequently, the resultant off-site dose is not significantly increased above that in the UFSAR. In fact, the 10CFR100 thyroid dose limit of 300 rem would not be exceeded even if all 89 CRD withdrawal lines failed.

Small Line Break Outside Containment

In this scenario, failure of a CRD line was considered to be similar to an Instrument Line Break outside primary containment and therefore was evaluated under SRP 15.6.2 against an offsite dose criterion of 10% of 10CFR100 (30 rem thyroid). The assumptions were:

1) The failed CRD line leaks at the maximum 10 gpm while the reactor is at full pressure.
2) Reactor depressurization occurs in a six hour time period, with the temperature decreasing linearly from 550 degrees F to 350 degrees F. The leak flow rate is assumed to vary linearly from 10 gpm to 3 gpm over a six hour period.
3) The reactor water iodine concentration is at the Technical Specification limit of 1.2 $\mu$Ci/gm prior to depressurization.
4) Spiking activity released to the coolant during depressurization is based on GE assumptions for a 95th percentile spiking event. The release of activity to the coolant is 2.1, 3.2, 5.0, 5.4, and 4.8 curies per fuel bundle for I-131, I-132, I-133, I-134, and I-135 respectively.
5) The fraction of fluid which flashes to steam varies with

the reactor water temperature and pressure conditions during the transient. The minimum allowable fraction was assumed to be 10%, consistent with SRP 15.6.5 Appendix B.

6) Regulatory Guide 1.3 meteorology is applied for releases from the 100m stack. Regulatory Guide 1.3 ground level release meteorology, as modified by Regulatory Guide 1.145 is used for roof vent releases.

The results of the analysis indicate that if SGTS operation is initiated prior to or coincident with the reactor depressurization transient, the offsite dose consequences for the period during and after depressurization, are on the order of 1 to 2 mrem. If SGTS initiation is assumed to be delayed for 30 minutes, the calculated two hour EAB dose increased to approximately 200 mrem and the 30 day LPZ dose was approximately 30 mrem. With a two hour delay in SGTS initiation, the calculated doses increase to about 1200 mrem at the EAB and 200 mrem at the LPZ. These are still relatively low compared to the 30,000 mrem acceptance criterion.

If there is a time delay between occurrence of the failure and detection of the leakage and subsequent startup of the SGTS, activity releases would occur via the roof vent and would be treated as unfiltered releases at ground level. With this assumption and iodine concentration at the Technical Specification limit of 1.2 $\mu$Ci/gm, application of Regulatory Guide 1.145 meteorology resulted in a calculated thyroid inhalation dose rate of 75 mrem/hr at the EAB. Thus, if the monitoring capability is such that the delay before SGTS initiation is no more than two hours, the EAB dose increment for this release period would be no more than 150 mrem. All lines could fail without exceeding 10% of 10CFR100 provided that the SGTS is in operation during and after reactor depressurization.

As noted earlier, the resultant offsite doses are linear with the source term. Therefore, assuming the iodine concentration is at the surveillance limit of 0.012 $\mu$Ci/gm, the offsite doses would remain within the allowable limits provided the SGTS is started within 200 hours after the occurrence of the failure. This is well within our weekly surveillance interval for our effluent monitoring system. Consequently, we would be able to identify the leakage prior to exceeding the offsite dose limit.

## Conclusion

While the above discussion details the worst case consequences of completely severed insert and withdraw lines, our experience with 30-07W indicates that the leak-before-break phenomenon is a more likely occurrence. (Note: Although classic leak-before-break theory is not strictly applicable to this situation, the fundamental premise that pipes should show signs of minor leakage prior to catastrophic failure is applicable.)

Type 304 stainless steel is a very ductile material. Any leaks that occur due to propagation of cracks will be detected long before a complete line break occurs. Even in the unlikely event of a Design Basis Earthquake (DBE), the resulting stresses would not result in the complete shear of a CRD insert or withdraw pipe, even in the cases of thru-wall cracking of up to 180° circumferentially. This is substantiated by actual experiences of piping systems during earthquakes with much higher loads than the DBE for the DAEC, as documented in various EPRI and Seismic Qualification Utility Group (SQUG) reports.

In addition, there are numerous CRD system operational anomalies that would warn the operator of a seriously leaking or broken insert and/or withdrawal line. The operating anomalies that provide the earliest warning are notch withdrawal problems (insert line leakage of >0.8-1.5 gpm) and CRD high temperatures above 250° F (withdraw line leakage of >0.2-0.4 gpm). However, our experiences with CRD leakage monitoring systems together with the leak-before-break phenomenon indicates that we should have early indications of leaking lines if they occur at all. Appropriate actions can than be taken to ensure that the design function of the CRD system and the transient and accident analyses are not adversely affected.

Several questions and concerns have been raised in the course of discussions on the CRD problem, pertaining to the history of the CRD drywell penetrations, test methods, etc. Due to the current status of the project, some of the issues are less relevant. However, in order to provide a complete review of the CRD concerns, several of these questions are addressed below.

Original Construction and Installation of the CRD Insert/Withdraw Line Penetrations

All four penetrations were manufactured by Chicago Bridge and Iron in a shop environment, using the manual Shielded Metal Arc Welding (SMAW) process and an E309 electrode. No documentation regarding the Stop/Start locations for the weld were located. Liquid penetrant, and baroscope inspection were performed on each pipe and its weld, and each bundle was subjected to a hydrostatic test. The shop-fabricated assemblies (pipes and plate) were shipped as completed assemblies to the site.

Magnetic particle examinations of the assembly perimeter welds occurred as follows: Southeast on 4/30/71, Northwest and Northeast on 5/12/71 and Southwest on 5/19/71. It is inferred from these records that the assemblies were installed in the drywell shell in that order. Fabrication records indicate the assemblies were defect free. Records indicate some of the pipes in the four bundles had to be reconfigured in August, 1971, to meet the specified slope requirement. No record of the technique used for the reconfiguration exists. In November of 1973, prior to plant startup, one Northwest bundle CRD pipe was noted as leaking at its drywell penetration. It was plugged, and the line re-routed through a spare penetration.

1990 Sampling and Removal

During the 1990 refueling outage, ultrasonic testing of the CRD lines with an internal probe was used to determine the extent of the problem in the other bundles. This method was chosen in lieu of less intrusive methods (such as eddy current testing) because of its demonstrated ability to detect fine indications, and the availability of equipment and skilled personnel. Removal of pipes for metallurgical examination was accomplished by coring out the lines at the steel drywell shell so that the portion of the shell around the pipe was removed. This ensured that all welds to the shell were not destroyed, so that further analysis could be performed.

## Metallurgical Analysis Detail

A total of 17 CRD lines from the Southwest bundle were sent offsite to two independent laboratories. For the four lines in the top lefthand corner of the Southwest bundle, the analysis of the weld and base metal by Packer Engineering found them to be within the specifications for SA312 type 304 stainless steel, and E309 welding electrode. These are the respective materials called out in the design drawings. This group of four pipes contained three of the four thru-wall indications, including the visible pinhole leaker, 30-07W. The 1990 metallurgical exams found no readily apparent fabrication defects. No consistent correlation between indication location and evidence of Stop/Start locations were noted. Hardness of the weld metal, heat-affected zones, and the base metal were obtained for the aforementioned four samples. No excessively high values of hardness, which might suggest improper welding procedures, were noted. Final results are still pending for the remaining lines, but there has been no conflicting data reported for the additional examinations performed thus far.

## Ball Check Valves

The safety analysis developed for the CRD insert/withdraw indication problem relies on the existence of a ball check valve in the insert line of a CRD. This check valve is a Haynes Stellite three-fourths inch diameter ball. The ball check is inspected at least once every ten years during CRD mechanism overhaul. There has been a minimal history of replacement of these ball check valves. The CRD mechanism must pass a leakage test following reassembly. There have not been difficulties with these ball check valves failing this test.

## Root Cause

The location and orientation of the indications suggest that a "bending" motion, from an unknown source, resulted in propagation of the indications. As previously noted, the existence of high cycle fatigue in the indications narrowed the scope of possible primary forcing functions which could cause bending. The discovery that the problem was unique to the southwest bundle also eliminated a number of root cause possibilities. One possibility now considered unlikely as the primary forcing function is the relative thermal expansions of the steel drywell shell, the concrete radiological shield, and the CRD piping. This was neither unique to the Southwest bundle, nor would it result in high cycle fatigue. Specifically, thermal displacements are as follows:

| Locations/Condition | Vertical | Horizontal |
|---|---|---|
| CRD Housing Flange | 0.25 inches down | none |
| Drywell Wall - Normal | 0.192 inches up | 0.239 in. radially out |
| Drywell Wall - Accident | 0.53 inches up | 0.656 in. radially out |

Original design drawings specified a minimum gap between the top of the CRD lines and their sleeve to ensure room for expansion. The lead wool packed into the space between the sleeve and the lines would not, under normal circumstances, impede differential motion. Although not responsible for the high cycle fatigue seen, line stress due to thermal displacement, pipe position in the sleeves, and the relative tightness of the wool packing has not been ruled out as a factor which could make one bundle more susceptible to high cycle fatigue. This may require further study depending what additional information is obtained about the primary forcing function. CRD system measurements internal to the drywell have been taken. More will be performed on the outside of the drywell following startup. Measurements on the CRD line positions will not reflect the position of the lead wool, which was removed earlier on all lines in order to insert the fiber-optic probe. During initial walkdowns which have taken place in the area, no noticeable distinction between bundles has been noted regarding CRD pipe position relative to the sleeves.