Consumers Power
POWERING MICHIGAN'S PROGRESS
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November 1, 1989

Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

DOCKET 50-155 - LICENSE DPR-6 - BIG ROCK POINT PLANT -
RESPONSE TO INSPECTION REPORT 89-015

By letter dated September 19, 1989 the commission provided Consumers Power Company (CPCo) with Inspection Report 89-015 which contained various concerns/questions dealing with the 1989 Integrated Leak Rate Test (ILRT) at Big Rock Point and requested a response by October 19, 1989. On Monday, October 16, 1989 CPCo personnel met with Region III staff to discuss the ILRT issues. By letter dated October 19, 1989 Consumers Power Company requested additional time to respond to Inspection Report 89-015. This letter fulfills Consumers Power Company's commitment to respond to Inspection Report 89-015 by November 1, 1989.

This letter contains responses to Unresolved Item (155/89015-01(DRS)), Unresolved Item (155/89015-02(DRS)), and Open Item (155/89015-03(DRS)), as well as additional information concerning the Big Rock Point Integrated Leak Rate Test (ILRT).

Unresolved Item (155/89015-01(DRS))

NRC Concern

"The regional-based inspectors also reviewed the licensee's Technical Specifications against the requirements of Appendix J. Discrepancies were noted in that (1) no tests were performed to determine the relationship between Ltm at 11.5 psig and Lam at 27 psig. Appendix J requires a CILRT to be run both at the full and reduced pressure during one outage in order to establish the maximum allowable leakage rate, Lt for future tests. Additionally, (2) no exemption, or request for an exemption, from the requirements of Appendix J for the correlating tests was found.

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The licensee was requested to determine if such an exemption had been granted, or otherwise justify how they met the Appendix J requirement.

Additionally the inspectors noted that this test, as well as the majority of the previous tests, was performed at a reduced pressure that was below one-half of Pa. Appendix J requires that reduced pressure testing be performed at a minimum of one-half of Pa. The Technical Specifications, Section 3.7 defines the La (the maximum allowable leakage) as 0.5 wt%/day at the design pressure of 27 psig. Appendix J states that La is to be the maximum leakage at the peak accident pressure, Pa, and that this value is to be documented in the Technical Specifications. Verbal discussions with the licensee indicate that they consider the peak accident pressure to be 23 psig.

The licensee was requested to determine the correct Pa. If Pa is not 27 psig, then the definition of La in the current Technical Specification value is incorrect, and the licensee must determine the correct La as required by Appendix J. If the value of Pa is 27 psig, then the licensee must determine whether an exemption to allow performance of a reduced pressure test at less than one-half Pa has been granted.

The above questions regarding compliance with Appendix J are being tracked as an Unresolved Item. (155/89015-01(DRS))."

Response

The original BRP Final Hazards Summary Report Section 3.2.1 states that the calculated peak pressure in containment is 23 psig, based on the severance of a recirculating pump discharge line, with the reactor in hot standby condition at 1500 psia. This is Pa by 10 CFR 50 Appendix J definition. The value of 27 psig was conservatively chosen in order to accommodate possible increases in reactor volume during course of design and is not Pa. Technical Specification 3.1 also refers to 41.7 psia as design pressure, not accident pressure.

The early Plant Technical Specifications which discussed containment leakage and testing requirements specified the maximum integrated leakage rate as 0.5%/day at the design pressure of 27 psig. We believe this valve was developed prior to the existence of Appendix J and still appears in the current Technical Specifications. The early specifications also required testing at a minimum of 10 psig, however BRP has modified the Technical Specifications to reflect the Appendix J requirement of no less than one-half of Pa which is 11.5 psig.

Amendment 62 to the BRP Technical Specifications dated December 27, 1983 which changed the minimum pressure test from 10 psig to 11.5 psig also documents these facts. The NRC Safety Evaluation associated with this Amendment (attached) states the following:

> "Appendix J to 10 CFR 50 requires that the reduced pressure Type A test be performed at a test pressure, Pt, not less than 0.5 Pa, the calculated peak containment pressure based on the design basis accident. For the Big

Rock Point plant, Pa is 23 psig; therefore, the minimum acceptable value for Pt is 11.5 psig. Since this change will update the compliance of the Technical Specifications for Big Rock Point with Sections III.A.4 and III.A.5 of Appendix J to 10 CFR Part 50, we conclude that the proposed change is acceptable."

The BRP Technical Specifications do not specify a maximum allowable leakage rate at accident pressure (ie., La) but rather utilizes a more conservative limit of leakage at design pressure. This is the leakage limit utilized in accident analysis to determine offsite dose consequences.

Changing Technical Specifications to reflect the acceptance criteria at accident pressure versus design pressure was looked at previously, but was not requested since it would result in an increased allowable leakage that would not comply with containment design analysis. Using the correlation formula from Technical Specification 3.7(g) at Pa versus Pd results in an increased allowable leakage:

3.7(g) All leakage rates determined by a test pressure less than the applicable design pressure (containment design or design basis accident) shall be corrected using the following formula:

$L_t = L_e \, (P_t/P_e)^{\frac{1}{2}}$

L_t = % maximum allowable leakage rate, at test pressure.

L_e = % leakage rate, at extrapolated pressure.

P_t = Test pressure (PSIG).

P_e = Extrapolated pressure (PSIG).

Acceptance criteria on allowable leakage for the ILRT is .75 L_t.

Using P_e = design pressure = 27 psig results in an acceptance leakage of:

$$L_t = .5\%/\text{day} \left(\frac{11.5}{27} \right)^{\frac{1}{2}} = .33\%/\text{day}$$

Using P_e = accident pressure = 23 psig results in an non-conservative acceptance leakage of:

$$L_t = .5\%/\text{day} \left(\frac{11.5}{23} \right)^{\frac{1}{2}} = .35\%/\text{day}$$

The first method which is conservative is used as the acceptance criteria in the Big Rock Point ILRT.

This fact was also reviewed and accepted by the NRC as discussed in the Safety Evaluation associated with Amendment 21 dated October 20, 1978 (attached).

With respect to the following statement from the Inspection Report:

> "Appendix J requires a CILRT to be run both at the full and reduced pressure during one outage in order to establish the maximum allowable leakage rate, Lt for future tests. Additionally, no exemption, or request for an exemption, from the requirements of Appendix J for the correlating tests was found."

Consumers Power re-reviewed the above concerns against 10 CFR 50 Appendix J. The portion of Appendix J which discusses performance of two tests is Section III.A.4. This section only applies to preoperational leakage rate tests. Consumers Power performed the preoperational tests on Big Rock Point prior to this requirement and Appendix J does not require two tests during the subsequent periodic tests. Preoperational Testing occurred at Big Rock Point in 1961. An ILRT was performed at 27 psig which showed leakage was 0.036%/day. The first reduced pressure test was conducted at 10 psig in 1962 which showed leakage at 0.021%/day. This is discussed in Special Report No. SR-9 dated 9/12/66. On the basis that Consumers Power practice as discussed earlier was conservative and that "two tests" are only discussed as "Preoperational Test" requirements we had determined and still conclude that no exemption is needed.

Clarification of Appendix J Requirements

NRC Conern

This section of the Inspection Report transmitted the inspector's clarifications of Appendix J requirements. Review by Consumers Power personnel has resulted in the following comments:

"Periodic Type A, B, and C tests must include as-found results as well as as-left. If Type B and C tests are conducted prior to a Type A, the as-found condition of the containment must be calculated by adding any improvements in leakage rates, which are the result of repairs and adjustments (R/A), to the Type A test results using the "minimum pathway leakage" methodology. This method requires that:

(a) In the case where individual leak rates are assigned to two valves in series (both before and after R/A), the penetration through-leakages would simply be the smaller of the two valves' leakage rates.

(b) In the case where a leak rate is obtained by pressurizing between two isolation valves, and the individual valve's leak rates are not quantified, the as-found and the as-left penetration through-leakage for each valve would be 50% of the measured leak rate, if both valves are repaired.

(c) In the case where a leak rate is obtained by pressurizing between two isolation valves, and only one valve is repaired, the as-found penetration leak rate would conservatively be the final measured leak rate, and the as-left penetration through-leak rate would be zero. (This assumes the repaired valve leaks zero.)

Whenever a valve is replaced, repaired, or repacked during an outage for which Type A, B, and/or C surveillance testing was scheduled, local leak rate testing for the as-found as well as the as-left condition must be performed on that penetration. In the cases of a replaced valve, the as-found test can be waived, except during outages when a Type A test is scheduled, provided that no other containment isolation valve of similar design exists at any nuclear site owned by the same utility."

Response

Consumers Power does utilize the "minimum pathway leakage" methodology as described above in evaluation of as-found and as-left results. Testing Type C penetrations before and after maintenance is performed at or above accident pressure to evaluate corrective actions. Testing is also conducted at one-half accident pressure on penetrations receiving maintenance during the period from initiation of the containment inspection and the performance of the Type A test. This additional test is conducted to comply with Appendix J, Section III.A.5.(b).(1), since Big Rock Point conducts an ILRT at half pressure. Consumers Power also believes that repacking, adjusting, or adding packing to valve does not always affect the leak tightness of a valve. Each valve is reviewed to determine affects of maintenance to determine if pre/post leak testing is required.

NRC Concern

Test connections between containment isolation valves must be administratively controlled to ensure their leak tightness or otherwise be subject to Type C testing. One way to ensure their leak tightness is to cap, with a good seal, the test connection after its use. (Note: test connection lines which penetrate containment must have two valves and a cap.) Proper administrative controls should ensure valve closure and cap reinstallation within the local leak rate testing procedure, and with a checklist prior to unit restart.

Response

Consumers Power recognizes and supports the above as current licensing criteria, however, BRP has areas where plant design may not conform to this criteria. This subject was evaluated by Consumers Power and the NRC in the Systematic Evaluation Program Topic VI-4; Containment Isolation System. A copy of SEP Topic VI-4 as described in NUREG-0828, Integrated Plant Safety Assessment for Big Rock Point is attached to provide an evaluation of the containment issues.

Unresolved Item (155/89015-02(DRS))

NRC Concern

Through IE Inspection Report dated 09/19/89, NRC requested additional information to allow regional inspectors to determine the validity of the Containment Integrated Leak Rate Test (ILRT) performed at Big Rock Point during the 1989 refueling outage. Specifically, Consumers Power was requested to provide:

(1) "Detailed information to show why a 20 degrees delta between steel temperature and ambient temperature is expected. This information should provide enough data points in regard to (a) time of day, (b) location on sphere, and (c) local weather conditions (such as cloud cover) so that reasonable extrapolation back to the time of the test is valid.

(2) Justification for the 75% "turbine building factor".

(3) If, as was indicated during the exit, the weather data as supplied by the National Weather Bureau for the area on the day of the test is not applicable, then a log (or other documentation) indicating the weather conditions at the plant during the test shall be provided."

This information was requested to support Consumers Power Company's conclusions that apparent minor air mass increases and decreases during the test were due to diurnal effects principally causing the containment sphere volume to decrease/increase respectively.

Response

To more accurately assess the actual volume changes the Big Rock Point containment sphere experiences due to ambient weather conditions, it is necessary to take actual surface temperatures of the steel surface during the test interval. This has not been done for any previous CILRT satisfactorily performed at Big Rock Point. As stated in the IE Report, Consumers Power Company was verbally requested to estimate the volume change as a result of daily temperature fluctuations. The numerical data informally provided was based upon engineering judgement. We believe that attempting to gather current steel temperature and corresponding ambient conditions in order to extrapolate conditions during the test would produce erroneous results since currently (1) the plant is in power operation and significant internal heat is being generated, (2) the containment is being continuously ventilated, (3) the correlation of ambient weather conditions is questionable, (4) the solar intensity, i.e. angle of incidence, has changed.

The suggested "75% turbine building factor" was based upon engineering judgement and attempted to estimate air mass changes during expansion and contraction of the containment. The actual steel surface temperature which was not recorded can only be estimated. Therefore any refinement of this number is not of significant benefit.

Regarding item (3) above, Attachment 4 is provided for temperatures recorded at the plant site during the CILRT. During the CILRT, weather conditions at the plant site were monitored by two separate methods and at differing locations. Ambient temperatures were recorded by the test data logger at 15 minute intervals from a calibrated RTD (the physical location of this RTD was on the east side of the containment building wherein afternoon shading of this RTD occurred). Plant operations staff also recorded ambient temperature conditions at approximately 30 minute intervals during the CILRT. This temperature sensor is protected from wind and not significantly shaded. Ambient temperatures taken by the Coast Guard Station during the hold portion of the test are also included in Attachment 4. Generally these temperatures were in good agreement. The weather conditions during this period were hazy, hot and humid on July 25, 1989 and July 26, 1989 cloudy with rain the morning of the July 27, 1989, and clearing skies on the afternoon of July 27, 1989 with cooler temperatures than the previous 2 days.

Some of the temperature/weather differences can be attributed to the fact that the Coast Guard Station is located 4 miles away from the plant site on an inland lake while Big Rock Point is directly on Lake Michigan. (See Attachment 5 for map of surrounding area.) Due to Great Lake effects upon weather conditions at the shore, temperatures at Big Rock Point can differ significantly from actual inland temperatures. Outside atmospheric conditions are monitored during the type A test in order to comply with the guidelines imposed by ANSI/ANS 56.8 and N45.4; they have not been intended to be used as data in calculations directly affecting the outcome of the test.

Diurnal Effects

Diurnal effects are cyclical thermal fluctuations originating from temperature changes from daytime to nighttime and vice versa. The diurnal temperature change is generally viewed as the change from minimum temperature to maximum temperature. However, this definition implies this effect is simply due to ambient temperatures, which is not totally correct. Heating of a structure or surface can also take place by solar radiation; this effect enhances the diurnal temperature swing experienced by the Big Rock Point containment structure. This effect is ever-changing and influenced by many factors including atmospheric temperature, cloud cover, wind conditions, precipitation and generally the time of year.

For the Big Rock Point CILRT, the diurnal effect has generally two results: (1) temperature inside containment rises or falls and (2) the spherical steel containment structure expands or contracts as a direct result of the changing metal temperature. While the two net effects are related, they are viewed separately because of the different impact they have on the leak rate calculation.

Because containment mass is calculated using the ideal gas law and the volume is a constant value in the computer program, the computer calculates a decrease in mass due to the increases in temperature and pressure. When the opposite occurs (temperatures and pressures drop) the computer indicates an increase in mass because the volume is assumed to be a constant value. A true volume increase retards the pressure increase that follows rising temperature.

A review of several past Big Rock Point CILRTs indicates diurnal effects were experienced in varying degrees. This is indicated by graphs showing daily swings in containment average temperature and pressure versus time. The degree that the diurnal effect influences test results involves several factors: (1) weather conditions, (2) the magnitude of the actual containment leakage and (3) the time of day that the hold test is initiated.

Evidence to support a small leakage value can be found from local leak rate totals (using maximum pathway leakage). These leak rate totals have been trending downward over the last several years. When the actual containment leakage rate is small, measurement uncertainties compounded by diurnal effects reduce the accuracy of the computer model to quantify an actual leakage rate. Conversely, if the containment leakage rate is large compared to measurement uncertainties and diurnal effects, the model more accurately quantifies the leakage rate. The observed results of the verification test performed with an imposed leak of approximately La clearly indicate the ability of the containment model to measure the required leak rate. While it is not disputed that the diurnal effects may mask some amount of leakage, the amount is minimal and well below the allowable containment leakage of .75La.

In order to gain a perspective on the magnitude of the effects, assume that the diurnal effect masked a leakage rate equal to .75La. The rate of leakage at .75La at a test pressure of 13.5 psig is slightly more than 14 lbm/hr. A 1°F steel temperature change can result in approximately 26 ft^3 containment volume change on a 130 foot diameter ideal sphere. After a test period of 24 hours, the temperature <u>change</u> would have had to be more than 90°F to mask a .75La leakage rate. Attachment 6 shows this calculation in detail. Since the containment is not an ideal sphere in terms of heat transfer, even larger temperature changes would be required. Because no temperature changes of this magnitude occurred either inside or out, the actual leakage rate cannot be greater than .75La.

A second method of establishing whether or not the magnitude of diurnal effects masked a leakage rate equal to .75La is to use the measured variable of containment pressure, assume a leakage rate of .75La and calculate an average RTD value. The calculated RTD value is then compared to the measured RTD value. After 28 hours of the hold test, the calculated average RTD value was slightly more than 80°F compared to the measured average RTD value of 78.3°F, with an RTD accuracy of ±.002°F. The measured average RTD value, being less than that required to artificially maintain pressure with an assumed leakage rate of .75La, indicates that actual 1989 CILRT leakage rate was less than the allowable limit. Attachment 7 is an example of the above calculation. Attachment 8 is a graph of measured RTD versus calculated RTD based on .75La.

The time at which hold test is initiated significantly affects the numerical leakage value obtained at the end of this test. Once a leakage trend is established, the final leakage rate is representative of the amount a containment structure leaks; however, it is not empirically exact. In other words, the final leak rate value can vary numerically depending upon both start time and weather conditions without exceeding allowable leakage values. Appendix J does not allow the luxury of reinitializing hold test start times backward to obtain favorable results in a reasonable amount of time. "Test" cases were run using both stabilization and hold test data with different test periods and start times to analyze the effect this had on the leakage results. While these informational, test results are not significantly different than those observed from the actual CILRT, the leakage values obtained are slightly different, i.e., positive measured leakage. This provides additional evidence that the leakage trend is accurate; although the final negative measured leakage value is misleading due to diurnal effects on containment. Once again, the amount that a containment structure actually leaks is only a function of the mechanical condition of the structure itself and not dependent upon the time a CILRT hold test is started.

Due to the diurnal effect on the containment structure and the small amount of leakage, both the measured and the 95% UCL leak rate were calculated to be negative for most of the hold test. This negative leak rate can be attributed to both cool evening temperatures and the thundershower which resulted in cooler than normal temperatures during the day of the hold test. As a result, the containment mass appeared to increase through a portion of the test when the heating of a normal sunny day should have indicated a decreasing mass in conjunction with a more positive leak rate. The hold test was run for 28 hours to get the 95% UCL above the zero mark. It would have been preferred to have the measured leakage rate positive as well. However, it was concluded after 28 hours that the necessary Appendix J criteria for the CILRT had been met and the hold test was terminated.

Open Item (155/89015-03(DRS))

NRC Concern

"The licensee was to submit revisions to the calculated leak rate due to the (1) changes in sump level, or justification why these changes are negligible, and (2) corrections to the CRD accumulator penalty, based on correct application of R/S data. The licensee also needed to revise their CILRT procedure in order to ensure that the inconsistencies mentioned above were eliminated. These will be tracked as Open Item (155/89015-03 (DRS))."

Response

Calculations for (1) changes in sump level and (2) CRD accumulator readability and sensitivity have been performed in addition to the correction for the LPS header. The net effect of the sump level changes in an increase of 13.45 lbs/24 hour period. This is a .01059 %/day increase in the leak rate. Incorporating readability and sensitivity into the penalty calculation for the CRD accumulators and LPS header resulted in a total penalty of 6.9128 lbs/24 hr or a leak rate correction of .00544 %/day. (Attachments 9 & 10 show these penalty calculations.)

Revisions to the CILRT procedure to accommodate water level and pressurized container corrections will be made prior to the next ILRT.

FUTURE ILRT MODIFICATION

Consumers Power recognizes that the Big Rock Point containment is affected by unstable weather conditions. This can at times cause increased uncertainty in the data taken during the ILRT. Dealing with diurnal effects has been recognized since the first tests have been performed and Consumers continues to implement changes to improve the tests. Early tests were performed using the reference volume method and hand calculations. Diurnal effects during these years resulted in long delays, up to a week, to gain acceptable results. In the late seventies, the reference volume method was replaced by a computer based testing system with new sensors followed by modeling and program changes. These efforts have improved leakage quantification, however, diurnal effects still can cause difficulties. Prior to the 1989 ILRT, Consumers Power and the NRC amended the Big Rock Point Technical Specifications to permit use of the "Bechtel" method for Type A testing. This method was preferred if a stable time period during the evening, (minimizing the impact of any diurnal effects on leakage measurement) could be obtained. During the 1989 ILRT, coordination and preparation difficulties caused a delayed start resulting in not meeting the acceptance criteria for the "Bechtel" test. A subsequent default into the 24 hour mass-point method then allowed the diurnal effects to cause the results noted during the test.

To improve test performance (i.e., leakage measurement) during the next Integrated Leak Rate Test the following actions will be taken:

- Coordination and preparation activities will be improved to assure an ideal start time for the "Bechtel" type test which should reduce the impact of the diurnal effects.

- Advances in modeling will be examined for improvements which would reduce the impact of atmospheric changes on leakage data.

° Mass variations due to diurnal effects are primarily related to containment volume changes due to expansion and contraction of the containment shell. Evaluation of various methods will be conducted during the next Refueling Outage to determine if quantification of the volume change is feasible. Prior to utilizing volume corrections during the next ILRT, the proposed refinements will be submitted for NRC review and comment.

J Daniel Eddy

J Daniel Eddy
Plant Licensing Engineer

CC Administrator, Region III, USNRC
NRC Resident Inspector - Big Rock Point

Attachments

ATTACHMENT 1

Consumers Power Company
Big Rock Point Plant
Docket 50-155

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 62 TO
FACILITY OPERATING LICENSE NO. DPR-6

November 1, 1989

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OC1089-0017-NL02

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 62 TO FACILITY OPERATING LICENSE NO. DPR-6

CONSUMERS POWER COMPANY

BIG ROCK POINT PLANT

DOCKET NO. 50-155

1.0 INTRODUCTION

By letter dated January 28, 1983, Consumers Power Company (CPC) (the licensee) requested changes to the Technical Specifications (TS) appended to Facility Operating License No. DPR-6 for the Big Rock Point Plant. The changes would increase the containment vessel reduced test pressure from 10 psig to 11.5 psig. The changes approved by this amendment involve slight revisions over the changes proposed by Consumers Power Company. These revisions were discussed and agreed to by the NRC staff and Consumers Power Company.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on August 23, 1983 (48 FR 38398). No request for hearing was received and no comments were received.

This amendment also corrects a typographical error made in Amendment No. 61. The changes made in Amendment No. 61 were addressed by a Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action which was published in the Federal Register on October 12, 1983 (48 FR 46457). No request for hearing was received and no comments were received. The correction addressed by this amendment is supported in the SER attached to Amendment No. 61 and is within the scope of change addressed by the Notice.

2.0 EVALUATION

The proposed change was recommended by the NRC staff in a Technical Evaluation transmitted by letter to the licensee on November 23, 1982. Appendix J to 10 CFR 50 requires that the reduced pressure Type A test be performed at a test pressure, Pt, not less than 0.5 Pa, the calculated peak containment pressure based on the design basis accident. For the Big Rock Point plant, PA is 23 psig; therefore, the minimum acceptable value for Pt is 11.5 psig. Since this change will update the compliance of the Technical Specifications for Big Rock Point with Sections III.A.4 and III.A.5 of Appendix J to 10 CFR Part 50, we conclude that the proposed change is acceptable.

Amendment No. 61 transmitted by letter dated November 14, 1983 contained a typographical error in revised Table 2 on page 5-9b. The highest value of Planar Average Exposure should be 41,400 MWD/STM as was indicated in the body of the supporting SE attached to Amendment 61.

3.0 ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or the health and safety of the public.

5.0 ACKNOWLEDGEMENT

This evaluation was prepared by J.R. Hall and R. Emch.

Date: December 27, 1983

ATTACHMENT 2

Consumers Power Company
Big Rock Point Plant
Docket 50-155

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 21 TO LICENSE NO. DPR-6

November 1, 1989

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ATTACHMENT 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 21 TO LICENSE NO. DPR-6

CONSUMERS POWER COMPANY

BIG ROCK POINT POWER PLANT

DOCKET NO. 50-155

Introduction

By letter dated May 17, 1978, Consumers Power Company (CPCo) submitted an application for an amendment to the Technical Specifications appended to Facility Operating License No. DPR-6 for the Big Rock Point Plant. This amendment changes the Technical Specifications by incorporating the requirements of Appendix J to 10 CFR 50 for the periodic test schedule and the formula for reduced pressure leak rate.

Evaluation

The proposed amendment would change current Specifications 3.7(f) and 3.7(g). Specification 3.7(f) specifies when the tests need to be repeated if the integrated leak rate test (ILRT) show the containment does not meet leakage acceptance criteria. The regulations require that the Commission review and approve the test schedule when the leakage rates exceed the acceptance criteria during an ILRT. In addition, the regulations require that whenever the leakage acceptance criteria is not satisfied in two consecutive ILRT's, then an ILRT shall be performed at each refueling or every 18 months, whichever occurs first, until two consecutive ILRT's give acceptable results.

CPCo proposes to adopt the wording directly from 111.A.6(b) of 10 CFR 50 Appendix J for the case where two consecutive ILRT's result in unacceptable leakage rates. This proposed specification replaces the current specification which addresses the action required with one ILRT with unacceptable leakage rates. Since the Big Rock Point containment does not require special considerations or more limiting specifications than the current regulations, we find this change acceptable.

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Specification 3.7(g) provides the acceptance criteria for the periodic ILRT's performed at pressure less than the design pressure. CPCo proposes to adopt the formula given in III.A.4.iii of 10 CFR 50 Appendix J to determine the maximum allowable leakage at reduced test pressure. The acceptance criteria for the ILRT is also taken directly from the Regulations. The acceptance criteria for type B and type C tests are more restrictive than for ILRT and therefore, the wording proposed by CPCo was changed to limit the use of the acceptance criteria to the ILRT.

Since CPCo uses the design basis accident pressure to determine acceptable leakage for some tests and the design pressure for other tests, they propose wording that allows either pressure to be used in determining acceptable leakage rates. The design basis accident pressure is lower at Big Rock Point than the design pressure, therefore, use of the design pressure is more conservative than required by regulation and is acceptable for use with the leakage formula.

Since the proposed change in 3.7(g) is consistent with, and in some cases more conservative than, the regulations and will not reduce the accuracy of leakage testing of the containment, we find this change to be acceptable.

Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusions

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in

(f) If two consecutive integrated leak rate tests fail to meet the specifications contained in this section, then an ILRT shall be performed at each plant shutdown for refueling or approximately 18 months, whichever occurs first, until two consecutive ILRTs meet the acceptance criteria. After the above special retest requirement is satisfied, then the testing schedule outlined in 3.7.E may be resumed from the date of the last special test (i.e., 3-1/3 years after completion of the second consecutive satisfactory special test).

(g) All leakage rates determined by a test pressure less than the applicable design pressure (containment design or design basis accident) shall be corrected using the following formula:

$L_t = L_e\,(P_t/P_e)^{1/2}$

L_t = % maximum allowable leakage rate, at test pressure.

L_e = % leakage rate, at extrapolated pressure.

P_t = Test pressure (PSIG).

P_e = Extrapolated pressure (PSIG).

Acceptance criteria on allowable leakage for the ILRT is .75 L_t.

ATTACHMENT 3

Consumers Power Company
Big Rock Point Plant
Docket 50-155

BIG ROCK POINT SYSTEMATIC EVALUATION PROGRAM TOPIC VI-4

November 1, 1989

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at the onset of an accident, there is no assurance that the water to be used for emergency core cooling and containment spray will be maintained within chemistry conditions during recirculation to minimize the probability for chloride-induced stress-corrosion cracking of austenitic stainless steel components and to minimize chemically induced hydrogen generation (i.e., corrosion induced).

In a letter dated June 17, 1982, the licensee maintained that 20 years of operating experience and the ongoing ISI program have demonstrated the adequacy of the existing limits and Technical Specifications in view of the actual salinity of Lake Michigan. Recent operating experience with false initiation of the emergency core cooling system has shown that such events are manageable. As noted under SEP Topic V-12.A (Section 4.18), the staff does not consider the differences between the plant Technical Specification limits and the requirements for new plants to be significant.

Offsite doses for these events are evaluated under Topic XV-18 (Section 4.28), as part of the Systematic Evaluation Program. Hydrogen generation from chemical reactions between metals inside containment and the containment and core spray water will be evaluated under the TMI Task Action Plan (Task II.B.7 in NUREG-0660) and Unresolved Safety Issue A-48 in NUREG-0705 generically in the future. In the interim, hydrogen generation does not pose a serious threat for Big Rock Point because of the large containment volume in relation to the core size and because containment failure as a result of hydrogen explosions was not a dominant contributor in the PRA accident sequences. The low probability of a core-degrading accident, coupled with the reduced temperatures that would exist after an accident, significantly reduces the potential for chloride-induced stress corrosion cracking. In addition, even if such corrosion were to occur, it would occur over a relatively long period of time and only in random locations, so that the staff would not expect it to affect the consequences of the accident or the ability to maintain the plant in a safe condition following an accident. Therefore, the staff concludes that the existing chemistry limits and inspections are adequate.

4.20 Topic VI-4 Containment Isolation System

10 CFR 50 (GDC 54, 55, 56, and 57), as implemented by SRP Section 6.2.4 and Regulatory Guides 1.11 and 1.141, requires isolation provisions for the lines penetrating the primary containment to maintain an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment. The topic evaluation of the containment penetrations at Big Rock Point has identified several areas that do not conform to current licensing criteria for containment isolation. The staff's limited PRA for Big Rock Point rates the reduction in containment leakage probability as a result of improving the isolation of electrical faults as being of low risk significance because of the high likelihood of containment valve leakage (0.1/demand compared with a contribution of 1 x 10-4/year from the specified penetrations) as a failure mode. The dominant contributor to containment leakage (0.1) is a failure of an operator to close valves VPI-1 and VPI-2 or VPI-3 in penetrations H-28 and H-29 if a leak develops. However, the design of these lines was found to conform to current licensing criteria in the topic evaluation.

4.20.1 Administrative Controls

The isolation valving arrangements for the following test, vent, and drain lines, associated with containment penetrations, differ from that required by current licensing criteria:

Penetration	Valve
H-11	VFW-138 and VFW-171
H-17	Undesignated vent valve on Drawing M-108
H-27	VFP-170
H-29	VPI-101
H-36	VFP-167, VFP-168, and VFP-169

The licensee has committed to administratively control these valves, except for valves VFW-171 and VPI-101. Valve VFW-171 is on a feedwater sampling line which must be open to provide continuous sample flow. The sample line is outside containment and the boundary formed by the redundant containment isolation valves and the test line containing valve VFW-138. Because the test line containing valve VFW-138 will be administratively closed and by applying the single-failure criterion to the containment isolation valves, the staff concludes that valve VFW-171 need not serve as a containment boundary. Valve VPI-101 is in a drain line for the core spray system pump return addressed in Sections 4.20 and 4.20.3.

The staff finds the licensee's proposal to administratively control these valves with locks or seal closures acceptable, provided that each of these lines is also equipped with either a pipe cap (in accordance with the ASME Code) or a redundant isolation valve. By letter dated September 13, 1983, the licensee provided suitable controls for all of the valves.

* 4.20.2 Instrument Lines

The isolation provisions for the following instrument lines, associated with containment penetrations, differ from that recommended by Regulatory Guide 1.11:

Penetration	Instrument/valve
H-10	Main steam/turbine control system (VTO-1A, PT-151, PT-175, PT-176, VFW-165, VFW-166)
H-27	VPI-137, VPI-157
H-36	VPI-136, VPI-156
H-89	RP-12.3
H-90	RP-12.4
H-96	VCI-15
H-98	RP-12.2
H-99	RP-12.1

The instrument lines associated with penetration H-10 are a part of the turbine control system. The licensee has determined that the radiation levels following an accident are low enough to permit manual isolation of these lines (note: the pressure instruments have root valves), and the licensee has committed to develop appropriate procedures to identify the conditions under which these lines should be isolated. This work is scheduled to be completed by July 1984.

The instrument lines associated with penetrations H-36 and H-27 are spares. The licensee has committed to seal-close the valves on these lines. The staff finds this proposal acceptable, provided the valves are included in the administrative check list to periodically verify the isolation of these lines. The remaining penetrations (H-89, -90, -96, -98, and -99) are sensing lines for containment pressure. The pressure instruments provide signals for engineered safety features and postaccident monitoring. Modifying these lines to provide automatic isolation would jeopardize that function. The integrity of the lines and instruments is verified during each containment integrated leakage rate test. In addition, the limited PRA concluded that leakage from such small lines does not significantly increase overall risk. On the basis of these considerations, the staff concludes that no further action is necessary.

4.20.3 Local Manual Valves on Safety Systems

The isolation provisions for the following containment penetrations differ from the explicit requirements of GDC 55 and 56, in that manual rather than automatic isolation valves are used:

Penetration	Valve
H-27	VFP-30
H-28	VPI-1, VPI-3
H-29	VPI-2, VPI-3, VPI-9
H-36	VFP-29
H-112	VPI-108
H-113	VPI-4

All of these lines are associated with the core spray, post-incident cooling, and fire water systems, which serve safety-related functions to mitigate the consequences of accidents.

VPI-1, -2, and -9 are located inside the containment and would not be accessible following a significant accident. VPI-9 is currently locked-closed and under administrative control. VFP-29 and -30 are closed from the control room as part of the procedure to switch from injection to recirculation cooling following an accident. VPI-108 is a locked-open vent valve in the core spray system, in a line that returns to the containment floor drains; a check valve inside the containment isolates this line in the event of a break in the line outside containment. VPI-3 is a locked-open isolation valve in the common core spray suction line outside containment.

The licensee has concluded that most of these valves should be locked-open to ensure the safety function following an accident. In addition, the licensee concluded that procedures for remote isolation of these lines is not warranted because isolation at the wrong time by human error might exacerbate the conditions of the accident. However, if any of these systems had to be taken out of service after an accident, the operator would want to close these valves to minimize leakage outside containment. This is an example of the procedures to be developed in Section 5.3.3.3.

The staff concludes that automatic isolation for these penetrations is not warranted because of the safety functions provided by the associated systems and the low likelihood of a passive failure in these systems following an accident. However, because most of the locked-open isolation valves could be used to mitigate the effects of pipe breaks in the associated systems, the licensee has committed to develop appropriate procedures to describe the conditions under which these valves should or should not be closed and identify the indicators available to the operator to verify those conditions. This project is scheduled to be completed by July 1984.

4.20.4 Local Valves on Nonsafety Systems

The isolation provisions for the following containment penetrations differ from the explicit requirements of GDC 55 and 56, in that manual rather than automatic isolation valves are used:

Penetration	Valve
H-10	VTG-101 and VFW-ST-01
H-11	CV-4000 and CV-4012
H-17	VRW-52
H-18	CV-4105
H-20	VA-14
H-23	VCU-13
H-25	VA-7

The line associated with penetration H-10 is the main steam line drain. This issue is addressed in the context of the isolation provisions for the main steam line itself in Section 4.20.5.

For the remaining penetrations, except H-18, the licensee has concluded that the valves identified do not serve a containment isolation function because existing, redundant isolation provisions already exist, as follows:

Penetration	Isolation barriers
H-11	VFW-9, VFW-304, and VFW-305
H-17	CV-4049, VRW-313
H-20 and H-25	Closed system inside containment with check valve
H-23	CV-4091, CV-4092, and CV-4093

These isolation barriers are all inside containment, rather than one inside and one outside as required by GDC 55 and 56. However, the limited PRA for Big Rock Point and other plants has found that the valve location does not significantly affect the penetration failure probability; that is, the probability of a break between the outermost valve and the containment is small compared with the probability of failure of all isolation valves. In addition, many of these valves are normally closed. The closed systems associated with penetrations H-20 and H-25 (service air and instrument air) normally operate at a pressure higher than the peak containment pressure, providing a constant leakage check, and these systems would have to passively fail upstream of the check valve to create a leakage path outside containment.

In a letter dated June 22, 1983(c), the licensee evaluated the reliability of the instrument and service air systems. Because of the potential for air inleakage to the containment as well as failure of the check valve to restrict leakage when the compressors are inoperable, the licensee concluded that implementing a leakage test program for these systems would be worthwhile. The licensee will begin this testing program during the 1984 refueling outage and monitor the results until sufficient data have been developed to draw a definitive conclusion.

In a letter dated December 22, 1983, the licensee concluded that valves VFW-9 and -304 in the feedwater system do not serve a containment isolation function, even though leakage through them has contributed to integrated (Type A) test failures, because the system would likely be in operation following an accident. However, for an accident caused by a break in the feedwater line, these valves would serve an isolation function. Nevertheless, on the basis of the risk perspective and the typical procedures for such accidents, the staff concludes that the existing isolation provisions are adequate.

In a letter dated February 2, 1984, the licensee committed to install an automatic operator for valve CV-4049 during the 1984 refueling outage.

For penetration H-18 (demineralized water), the licensee has determined that the remote manual control valve CV-4105 can be isolated by a hand switch in the control room. The licensee has committed to review the existing procedures to confirm that the operator has adequate instructions to determine when to close this valve.

On the basis of these considerations, the staff concludes that these isolation provisions are adequate and no additional actions are necessary.

4.20.5 Main Steam Line Isolation Valve

The main steam line is equipped with only a single isolation valve (MO-7050, with valve MO-7065 on the upstream drain), rather than redundant isolation valves as required by GDC 55. In the topic evaluation, the staff recommended that the licensee qualify downstream valves in the main steam system as containment isolation valves. However, this action would require automatic closure with a diverse isolation signal and leak testing for these valves.

The licensee evaluated various leak testing programs using PRA to develop cost-benefit estimates (see Appendix H, Issue 10). The results of this evaluation were presented in a letter dated June 22, 1983(c). The licensee concluded that a program for periodic stroke testing of the main steam line isolation valve (MSIV), to improve valve reliability, should be pursued. The licensee has estimated that the cost of adding a second isolation valve, to conform to current criteria, would be approximately $150,000. The corresponding reduction in exposure was estimated to be 33.8 person-rem/ reactor-year. Conversely, the licensee estimated that a testing program to improve the reliability of the existing isolation valve would be approximately $4,000 with an exposure reduction of 20.2 person-rem. The action recommended in the topic evaluation would fall somewhere between these two estimates.

The staff has reviewed the licensee's evaluation and, although several of the assumptions are questionable, agrees that the cost of adding a second isolation valve is not warranted. This conclusion is based, in part, on the conservative assumptions in the offsite dose evaluations performed in conjunction with SEP Topic XV-19.

Currently, the containment integrated leakage rate test is the means of determining the leakage integrity of the MSIV. The periodic testing proposed by the licensee is directed at determining the ability of the valve to shut, as opposed to the ability of the valve to restrict leakage. The staff believes that both functions are important. Consequently, the staff concludes that the licensee's proposal to develop a periodic testing program is acceptable, provided that the evaluation include a study of the feasibility of conducting periodic leakage integrity tests against some baseline condition. The licensee's operability testing program development is scheduled to begin in 1985, and the data collection and analysis to prove desired reliability is scheduled to be completed by March 1989. The licensee is continuing the evaluation of the staff's proposal to provide automatic closure of the downstream valves. In the interim, the licensee will monitor the results to determine whether any trends require a more immediate action.

4.20.6 Closed Systems

The following containment penetrations are associated with closed systems inside containment that have no containment isolation valves and so differ from the explicit requirements of GDC 57:

Penetration	System
H-9	Emergency condenser vent
H-12	Service water return
H-13	Service water supply
H-14	Heating steam
H-19	Heating condensate

The emergency condenser (penetration H-9) is being reviewed in conjunction with Topic III-5.A (Section 4.10) with regard to the ability to detect leakage and take corrective action. For the heating and service water systems, the licensee evaluated the cost-benefit of installing containment isolation valves in his June 22, 1983 submittal referenced earlier. The licensee has concluded that the estimated exposure reduction (3.2 person-rem/reactor-year) does not justify the cost ($150,000).

The staff agrees that the cost of adding isolation valves is not warranted, provided the system integrity is periodically verified to qualify the system as an extension of the containment. The licensee's evaluation did not consider the cost-benefit associated with periodic testing to verify the system integrity. Therefore, the staff recommended that the licensee develop a periodic inspection procedure to identify and correct significant system leakage.

The licensee has concluded that the existing roving patrols inside the containment provide adequate surveillance to identify significant degradation in these systems. In addition, the leakage detection system (see Section 4.16) is capable of detecting leaks as small as 0.02 gpm. The licensee has estimated that the probability of a breach in these systems is more than two orders of magnitude below the probability of the dominant containment failure modes;

even then, the systems would likely be at a pressure higher than the containment pressure so that any leakage would be into the containment.

On this basis, the staff concludes that the existing surveillance conditions are sufficient and, therefore, no further action is warranted.

4.20.7 Appendix J Leak Test Requirements

On November 23, 1982(a), a number of exemptions to the containment leak test requirements of Appendix J to 10 CFR 50 were granted to Big Rock Point. The forwarding letter for those exemptions and the safety evaluation that was attached indicated that several issues in the Appendix J review were being deferred to the integrated assessment in the SEP. The following sections describe the resolution of those items.

4.20.7.1 Containment Airlock Testing Frequency

Currently, the containment airlocks (equipment, personnel, and emergency) are leak tested every 6 months. Appendix J to 10 CFR 50 requires that airlocks be leak tested within 72 hours after each use or every 72 hours if the airlocks are used daily. Therefore, the explicit requirements of Appendix J to 10 CFR 50 are not met. The Appendix J safety evaluation proposed reduced pressure leak tests within 72 hours after each use or every 72 hours during frequent use in addition to the 6-month tests as an acceptable airlock leak test schedule.

The licensee has concluded that frequent use of the personnel airlock is necessary for the safe operation of the plant; the personnel airlock is used many times a day. Airlock testing is time consuming (requiring at least 4 hours to obtain statistically significant data), even for a reduced pressure test, because the entire airlock must be pressurized. The airlocks are all of the single seal design, not the double seal design which allows testing by pressurizing between the seals. During testing of the personnel airlock, entry to containment is curtailed because the only available entrance is the emergency airlock. The emergency airlock is opened daily as a personnel safety measure to ensure operability. The equipment airlock is used a couple of times a month. Each of the airlocks is tested every 6 months, and each airlock is covered by a preventive maintenance program, including seal inspection and cleaning. Moreover, the as-found leakage observed during the 6-month tests has been quite low. The leak rates have averaged 3% to 5% (closer to 3% since 1974) of the maximum Technical Specifications leakage limit. The requiremant of additional tests, even reduced pressure tests, would (1) place a burden on plant operations and (2) provide no increase in safety based on the record of the 6-month leakage tests. Installation of doors with testable seals (double-seal design) would be expensive.

On this basis, and on the basis of information from the limited PRA for Big Rock Point, the staff concludes that the present airlock leak test frequency is acceptable, provided the seals are periodically replaced in accordance with manufacturer's recommendations. In a letter dated February 2, 1984, the licensee committed to inspect these seals in accordance with the manufacturer's recommendations, which the staff understands include replacement as necessary. NRC action on this exemption request will be completed following issuance of the Final Integrated Plant Safety Assessment Report.

4.20.7.2 Testing of Main Steam and Main Steam Line Drain Isolation Valves

Currently, the Appendix J Type C leak tests of the main steam isolation valve and the main steam line drain valve are performed using water as the testing medium. Because these valves are not normally pressurized with fluid from a seal system, Appendix J requires that they be tested with air or nitrogen. The licensee has concluded that testing of the MSIV and the drain valve with air or nitrogen is not feasible. Because these valves are single valves, not a pair of valves in series, the common testing method of pressurizing the piping between the two valves in series cannot be done.

An air test of the MSIV and drain valve would require pressurizing a very large volume of piping with many other valves being used as isolation valves; this would be an impractical test. These valves are tested with air as part of the integrated containment leak rate test every 40 months. They are also tested with water during hydrostatic testing of the primary system at each refueling. Leakage during the hydrostatic tests is measured as drops of water per second.

In a letter dated February 2, 1984, the licensee committed to develop and implement a procedure, including any necessary modifications, to permit pneumatic testing of the MSIV beginning in the 1985 refueling outage. This procedure would not include the main steam line drain, because of the system configuration; however, that valve is normally closed, the line is small, and the leakage integrity is verified during both the system hydrostatic test and the containment integrated leakage test. In discussions with the licensee, the licensee has committed to develop a suitable test for the drain valve or to cut and cap the line downstream of the valve. Therefore, the staff finds the licensee's proposed action acceptable.

4.20.7.3 Testing of Isolation Devices for Closed Systems Inside Containment

The leak rate testing of isolation boundaries for the following systems, which are closed systems inside containment and which penetrate containment, was deferred to the integrated assessment because Topic VI-4 initially identified the possible need for additional isolation valves in some of these systems:

(1) service air
(2) service water
(3) heating and cooling
(4) instrument air
(5) integrated leakage rate test (ILRT) reference volume
(6) shutdown flushing

The licensee has concluded that lines associated with these systems would not rupture or leak significantly because they contain no high-energy fluids and have no openings to the containment atmosphere that provide a path to the environment. These lines are subject to the same environment as the containment shell and are provided the same surveillance for leakage during the ILRT. As further protection against leakage, the service water, service air, and instrument air systems normally operate at pressures greater than the maximum pressure during loss-coolant-accident (LOCA) conditions. The instrument air and service air systems are addressed in Section 4.20.4. These two systems have check valves inside containment and gate valves outside containment.

ATTACHMENT 4

Consumers Power Company
Big Rock Point Plant
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AMBIENT TEMPERATURE DATA - DURING CILRT TEST

November 1, 1989

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ATTACHMENT 4

AMBIENT TEMPERATURE DATA - DURING CILRT TEST

	Date	Time	Temp °F ILRT Outside RTD	Operations Log	U.S. Coast Guard
Stabilization	07/25/89	1600	83	82	
		1800	82	83	
		2000	78	80	
		2200	75	75	
	07/26/89	000	73	74	
		200	74	74	
		400	73	75	
		600	73	76	
		800	78	82	
		1000	86	87	
		1200	86	87	
		1400	85	88	
Hold Test		1600	85	85	
		1800	79	80	
		2000	78	78	
		2200	76	76	76
	07/27/89	000	77	77	77
		200	76	77	76
		400	73	80	76
		600	72	74	71
		800	74	72	71
		1000	82	80	77
		1200	72	73	70
		1400	78	83	79
		1600	77	82	76
		1800	76	82	74
		2000	73	79	73
Verification		2200	70	75	69
	07/28/89	000	66	68	66
		200	65	67	65
		400	65	66	64
		600	63	64	64
		800	63	62	59
		1000	67	69	-

ATTACHMENT 5

Consumers Power Company
Big Rock Point Plant
Docket 50-155

BIG ROCK POINT AREA MAP

November 1, 1989

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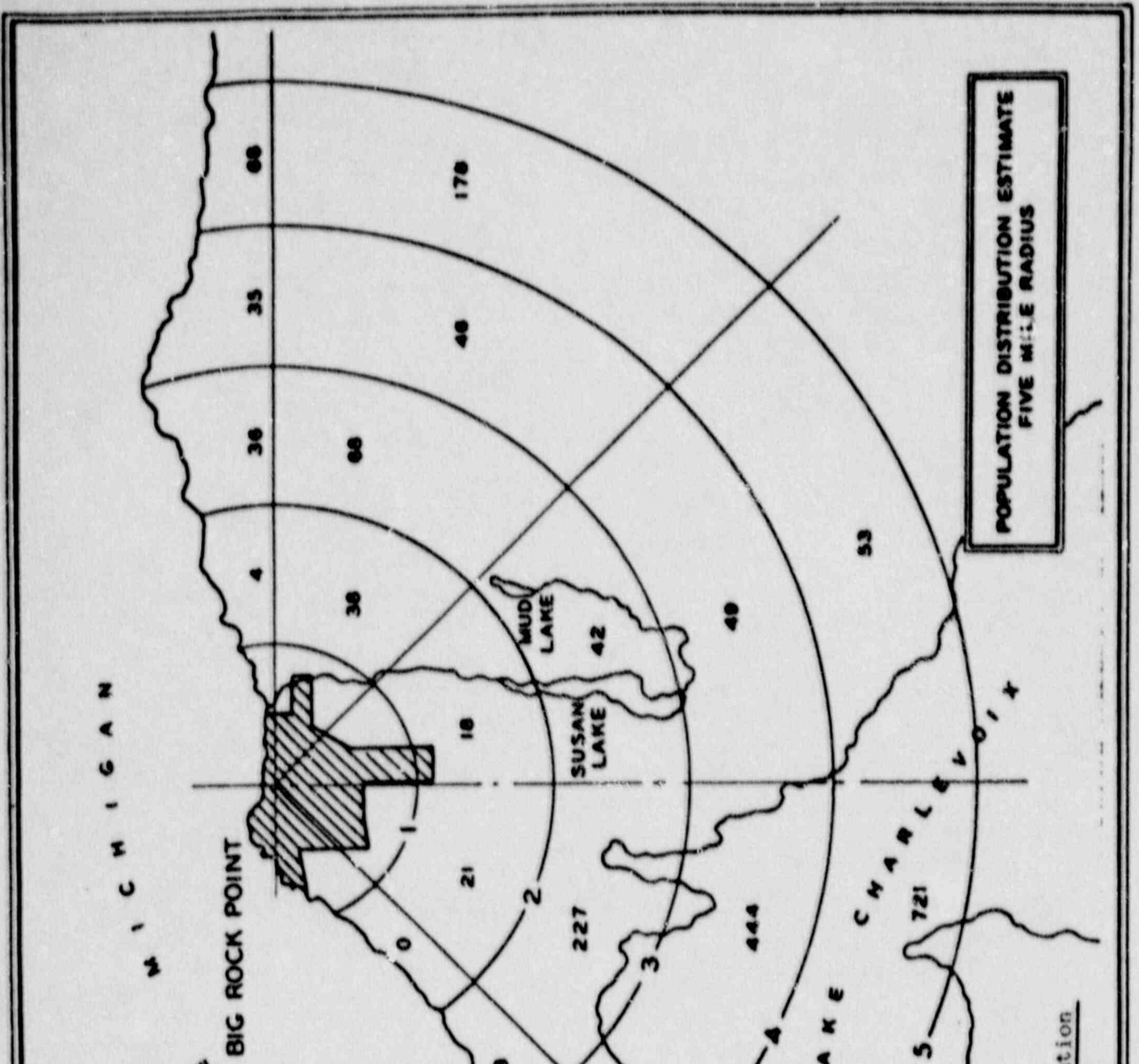
ATTACHMENT 5



ATTACHMENT 6

Consumers Power Company
Big Rock Point Plant
Docket 50-155

CONTAINMENT TEMPERATURE CHANGE CALCULATION

November 1, 1989

2 Pages

ATTACHMENT 6

To calculate the temperature change required to mask a leak rate of .75La, the change of volume per °F must first be determined. From this, the change in mass per °F can be derived.

Volume of an ideal sphere $= \frac{4}{3}\pi R^3$

Also, the circumference of a sphere is $2\pi R$ or πD.

For Big Rock Point, circum. $= \pi \times 130 \text{ ft} \times \frac{12 \text{ in}}{1 \text{ ft}} = 4900.8845$ inches

The incremental expansion of the circumference is given by:

$$\Delta L = L\alpha\Delta T$$

Where L = Length (or the circumference), inches = 4900.8845

α = Coef. of thermal expansion $= 7.7 \times 10^{-6} \frac{\text{in}}{\text{in. °F}}$

ΔT = Change of temperature, °F

$$\Delta L = (4900.8845 \text{ in.})\ 7.7 \times 10^{-6} \frac{\text{in}}{\text{in. °F}} \times \Delta T \quad \text{or}$$

$$\Delta L = .037737 \frac{\text{in}}{\text{°F}} \Delta T$$

The change of R is then:

$$\Delta R = \frac{\Delta L}{2\pi} = .037737 \frac{\text{in}}{\text{°F}} \times \frac{\Delta T}{2\pi}$$

$$\Delta R = 6.006 \times 10^{-3} \frac{\text{in}}{\text{°F}} \times \Delta T$$

The change of volume per °F is:

$$\Delta Vol = \frac{4}{3}\pi (R + \Delta R)^3$$

$$= 4.1888\ [65 \text{ ft} \times \frac{12 \text{ in}}{1 \text{ ft}} + 6.006 \times 10^{-3} \frac{\text{in}}{\text{°F}} \Delta T]$$

For a 1°F temperature change:

$$\text{Vol } 1 = 4.1888\ [780 \text{ in} + 6.006 \times 10^{-3} \frac{\text{in}}{°\text{F}} \times 1°\text{F}]^3$$

$$= 4.1888\ [780.006006 \text{ in}]^3$$

$$= 4.1888\ [4.74563 \times 10^{+8}] \text{ in}^3$$

$$= 1.987849 \times 10^9 \text{ in}^3 \times \frac{1 \text{ ft}^3}{1728 \text{ in}^3} = 1{,}150{,}376 \text{ ft}^3$$

For a 2°F temperature change:

$$\text{Vol } 2 = 4.1888\ [780 \text{ in} + 6.006 \times 10^{-3} \times 2]^3$$

$$= 4.1888\ [780 \text{ in} + .012012 \text{ in}]^3$$

$$= 4.1888\ [4.74574 \times 10^{+8} \text{ in}^3] = 1.98789 \times 10^9 \text{ in}^3$$

$$= 1.98789 \times 10^9 \text{ in}^3 \times \frac{1 \text{ ft}^3}{1728 \text{ in}^3} = 1{,}150{,}402 \text{ ft}^3$$

The volume change per °F is then:

$$\frac{V_2 - V_1}{1°\text{F}} = \frac{1{,}150{,}402 \text{ ft}^3 - 1{,}150{,}376 \text{ ft}^3}{1°\text{F}} = 26 \frac{\text{ft}^3}{°\text{F}}$$

The associated mass change per °F is then:

$$\frac{\Delta \text{Vol}}{°\text{F}} \times \rho_{\text{air @ 13.5 psig}}$$

Where: $\rho_{\text{air @ 13.5 psig}} = .1417 \frac{\text{lbm}}{\text{ft}^3}$

Therefore:

$$\frac{\Delta \text{Mass}}{°\text{F}} = 26 \frac{\text{ft}^3}{°\text{F}} \times \frac{.1417 \text{ lbm}}{\text{ft}^3} = 3.7 \frac{\text{lbm}}{°\text{F}}$$

From this relationship, the temperature change required to mask the .75La leak rate can be derived by dividing the theoretical decayed mass at any given time by 3.7 lbm/°F. For BRP the .75La leak rate is 14.095 lbm/hr. A straightforward conversion to temperature indicates the magnitude of temperature change necessary to mask the limiting .75La leak rate.

For example, after 24 hours of hold test the required ΔT (skin temperature) is:

$$\Delta T = [14.095 \frac{\text{lbm}}{\text{hr}}]\ (24 \text{ hr})\ (\frac{1}{3.7} \frac{°\text{F}}{\text{lbm}}) = 91.4\ °\text{F}$$

ATTACHMENT 7

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RTD CALCULATION

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ATTACHMENT 7

The argument that the diurnal effects did not mask a leak equal to .75La can be verified mathematically. Using the ideal gas law, and assumed leakage of .75La and the measured values for containment pressure, a new average RTD value can be calculated and compared with the actual measured value.

The new average RTD value is calculated from the following equation:

$$T = \frac{(P_{ave} - P_{vap})\ V}{R\ M}$$

Where:

- T - calculated average RTD corresponding to a leak of .75La, (°R)
- P_{ave} - measured average total pressure, psia
- P_{vap} - measured vapor pressure, psia
- V - containment free volume (assumed constant), ft^3
- R - ideal gas constant, ft-lbf/lbm - °R
- M - containment mass corresponding to .75La, lbm

After 28 hours of the hold test, the measured containment internal conditions were:

RTD_{ave} = 78.32°F P_{ave} = 28.090 psia P_{vap} = 0.3552 psia

The quantity of mass allowed to leak (corresponding to .75La) after 28 hours is:

(28 hrs) x (14.095 lbm/hr*) = 394.7 (lbm)

Therefore:

$$T = \frac{(28.090 - 0.3552)\ (912891.4)\ (144\ in^2/ft^2)}{(53.35)\ (127003.6^{**} - 394.7)}$$

$$= 539.77\ °R - 459.67$$

$$= 80.1°F$$

Since the calculated average RTD value of 80.1°F is larger than the measured average RTD value of 78.32°F (note: RTD accuracy ±0.002), a leakage of .75La was not masked by diurnal effects.

A graph of measured average RTD versus calculated average RTD for the duration of the hold test is included on the following page.

*The value of 14.095 lbm/hr is the slope of the .75La line.
**The value of 127003.6 lbm is the initial mass calculated at the start of the hold test.

ATTACHMENT 8

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GRAPH OF RTD vs TEMPERATURE

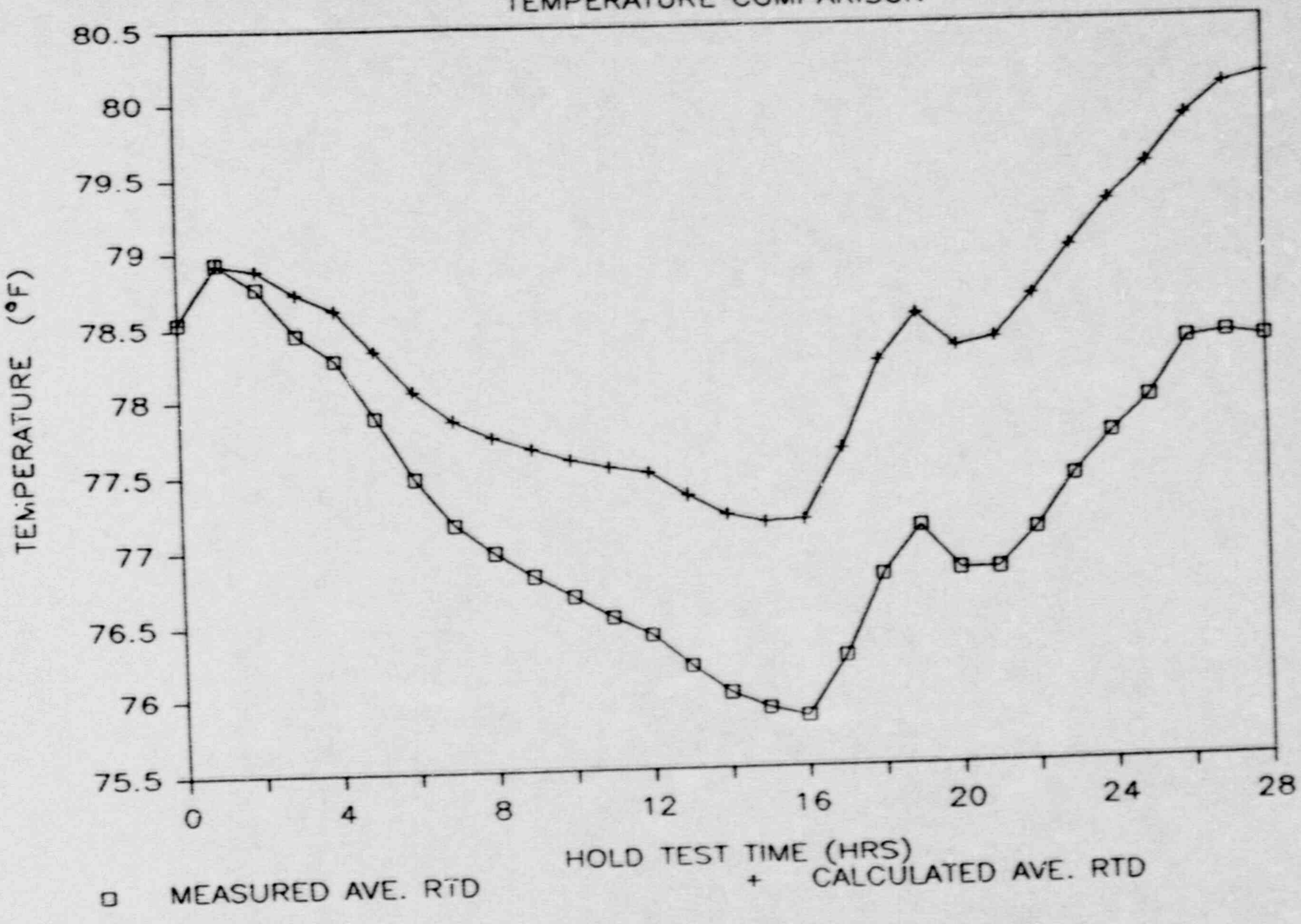
November 1, 1989

1 Page

.75La LIMITING LEAKAGE

TEMPERATURE COMPARISON

ATTACHMENT 8



ATTACHMENT 9

Consumers Power Company
Big Rock Point Plant
Docket 50-155

WATER INVENTORY BALANCE

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2 Pages

OC1089-0017-NL02

ATTACHMENT 9

WATER INVENTORY BALANCE

1) Level Decreases

a) Steam drum level from C/L - 1.5" to C/L - 3.5"

Diameter = 77-15/16"
Length = 36'-10"

$$\text{Volume} = \left(\frac{77.94 \text{ in}}{12 \text{ in/ft}} \right) (36.83 \text{ ft}) \left(\frac{2 \text{ in}}{12 \text{ in/ft}} \right)$$

$$= 39.9 \text{ ft}^3$$

b) Rod drive sump from 4.6" to 4.4"

From Tech Data book

4.6"	=	109.36 gal
4.4"	=	104.04 gal
Total	=	5.32 gal

$$\frac{5.32 \text{ gal}}{7.48 \text{ gal/ft}^3} = .7 \text{ ft}^3$$

c) RCW tank from 7'-4" to 7'-0"

(100 gal/in) x (4 in) = 400 gal

$$\frac{400 \text{ gal}}{7.48 \text{ gal/ft}^3} = 53.5 \text{ ft}^3$$

a + b + c = total decreases

$39.9 \text{ ft}^3 + 0.7 \text{ ft}^3 + 53.5 \text{ ft}^3 = 94.1 \text{ ft}^3$

2) Level Increases

d) Enclosure clean sump from 2.9" to 9.0"

From Tech Data book

9.0"	=	132.3 gal
2.9"	=	38.5 gal
Increase	=	93.8 gal

$$\frac{93.8 \text{ gal}}{7.48 \text{ gal/ft}^3} = 12.5 \text{ ft}^3$$

e) Enclosure dirty sump from 3.9" to 34.0"
From Tech Data Book

A 32" manometer reading corresponds to top of sump. The 34" reading is 2" above top of sump.

Therefore there was 2" of water on the floor of the Recirc Pump Room and the CRD Access Room. This corresponds to 565.5 gallons of water in addition to the water contained in the sump pit.

32.0"	= 615.7 gal
3.9"	= 51.2 gal
Increase	= 564.5 gal

Total for dirty sump = 564.5 + 565.5 = 1130.0 gal.

$$\frac{1130 \text{ gal}}{7.48 \text{ gal/ft}^3} = 151.1 \text{ ft}^3$$

f) Surge tank from 36% to 40%

From system description volume = 4750 gal

(.04) (4750 gal) = 190.0 gal

$$\frac{190.0 \text{ gal}}{7.48 \text{ gal/ft}^3} = 25.4 \text{ ft}^3$$

d + e + f = total increases

$12.5 \text{ ft}^3 + 151.1 \text{ ft}^3 + 25.4 \text{ ft}^3 = 189.0 \text{ ft}^3$

Unaccounted water volume = (d + e + f) - (a + b + c)

$= 189.0 \text{ ft}^3 - 94.1 \text{ ft}^3 = 94.9 \text{ ft}^3$

Equivalent air mass $= (.1417 \text{ lbm/ft}^3)\ (94.9 \text{ ft}^3)$

= 13.45 lbm of air

g) Calculation of %/day for penalty

$$\%/\text{day} = \frac{13.45 \text{ lbm}}{127049.1 \text{ lbm}^*} \times 100 = .01059\ \%/\text{day}$$

*The value of 127049.1 lbm is the Y-intercept of the .75La line.

ATTACHMENT 10

Consumers Power Company
Big Rock Point Plant
Docket 50-155

CALCULATION OF CRD ACCUMULATOR AND
LPS CHANGING HEADER PRESSURE ADDITIONS

November 1, 1989

4 Pages

OC1089-0017-NL02

Purpose:

To determine additions to be added to the containment leak rate due to changes in pressure in the Control Rod Drive (CRD) accumulators and the Liquid Poison System (LPS) charging header.

Assumptions:

1. Post test pressure increases in bottle pressures are caused by temperature effects. These bottles are assumed to have zero leakage. The change in mass calculated is assigned to all other bottles of lesser final pressure (ΔP).

2. Pressures are corrected 10 psi for readability and gage accuracy of 1% full scale, corresponding to CRD = 20 psi & LPS = 50 psi

References:

1. Engineering Thermodynamics, J.B. Jones & G.A. Hawkins, John Wiley & Sons, 1960

2. Handbook of Operation and Service Instructions, Greer Hydraulics Inc., January, 1962

A. Pressure and Temperature Corrections for CRD Accumulators

   1. Time: All corrections will be based on a 24 hr period to coincide with corrected period for test data. All initial data was recorded on 7-24-89; all final data taken after 2400 hrs on 7-29-89. Therefore, 96 hours is used for the time period between readings.

   2. Mass: For conservatism all mass calculations will be based on the lowest recorded temperature and highest recorded pressure during the test period. This will produce the largest mass based on Pv = RT (ideal gas equation).

   3. Compressibility: Per Reference 1 page 89, the compressibility of N_2 at <2100 psi is negligible for the small ΔP experienced (30 psi). Therefore, compressability concerns are considered negligible.

   4. N_2 vs. Air: Because air and N_2 have nearly identical gas constants (Reference 1 page 152), the masses can be added together to form a reasonably accurate total leakage from containment.

5. <u>Sample calculation for Accumulator A-3</u> See Table 1

a. P_i = 650 psig (Corrected for gage accuracy and readability.)
P_f = 590 psig (Corrected for gage accuracy and readability.)

b. Temperature = 66°F (lowest temperature recorded)

c. Volume = 1160 in^3 = .6713 ft^3 (Reference 2)

d. Calculation of specific volume:

$Pv = RT$

Where: P = Absolute Press (psia)

v = Specific Volume (ft^3/lbm)

V = Volume of Accumulator Bottles, ft^3

R = Gas Constant = 55.16 $\frac{ft/lbf}{lbm/°R}$ for Nitrogen

T = Temperature (°F)

i = Initial

f = Final

$$v_i = RT_i/P_i = \frac{55.16\ (460 + 66)}{(650 + 14.7)\ 144\ in^2/ft^2} = .3031\ ft^3/lbm$$

$$v_f = v_i\ (P_i/P_f) = .3031\ \frac{(650 + 14.7)}{(590 + 14.7)} = .3332\ ft^3/lbm$$

e. Mass release due to ΔP

$$Mass = V/v_i - V/v_f = \frac{.6713\ ft^3}{.3031\ ft^3/lbm} - \frac{.6713\ ft^3}{.3332\ ft^3/lbm} = .2001\ lbm$$

Corrections for time:

Mass = .2001 x 24 hrs/96 hrs = .0500 lbm/24 hrs

6. Mass release due to temperature changes:

The maximum pressure increase was 20 psig. This increase is assumed to result in temperature increases as mass was not added to the system. The mass correction due to the temperature rise is applied to all accumulators that displayed a pressure increase less than 20 psig.

Following the calculation format in section 5 above with

P_i = 640 psig

P_f = 660 psig

The mass release corrected for temperature is:

.0167 lbm/24 hrs based on ΔP = +20 psi

7. The total additional mass from all CRD accumulations is 2.0753 lbm/24 hr from Table 1.

B. LPS Charging Header N_2 Release Based on a Measured ΔP

Temperature: T = 66°F

Pressure: P_i = 2120 psig (corrected for gauge accuracy and readability.)

P_f = 1980 psig (corrected for gauge accuracy and readability.)

Volume: (16 cyl @ 1.73 ft^3 ea. + 50 ft of 1" dia. pipe.)
27.68 ft^3 + .273 ft^3 = 27.953 ft^3

Specific volume calculation:

$$v_i = RT/P_i$$

$$= \frac{55.16\ (460° + 66°)}{(2120 + 14.7)\ (144)} = .0944\ ft^3/lbm$$

$$v_f = v_i\ P_i/P_f$$

$$= .0944\ \frac{(2120 + 14.7)}{(1980 + 14.7)} = .1010\ ft^3/lbm$$

$$\text{Mass release} = V/v_i - V/v_f$$

$$= \frac{27.953}{.0944} - \frac{27.953}{.1010} = 19.350\ lbm \times 24\ hrs/96\ hrs = 4.8375\ lbm/24\ hrs$$

C. Calculation of %/Day for Penalty

CRD Accumulators = 2.0753 lbm/24 hr

LPS Header = 4.8375 lbm/24 hr

Total additional correction = 6.9128 lbm/24 hr

$$\%/\text{day} = \frac{6.9128}{127049.1^{*}} \times (100) = .00544\ \%/\text{day}$$

*The value of 127049.1 lbm is the Y-intercept of the .75La line.

TABLE 1

Consumers Power Company
Big Rock Point Plant
Docket 50-155

CRD ACCUMULATOR PENALTIES

November 1, 1989

2 Pages

CRD ACCUMULATOR PENALTIES

Table 1

Gage #		Pressure Reading (psig)	Corrected Readings* (psig)	Temperature Correction (#/24 hr)	ΔP Correction (#/24 hr)	Total Leakage (#/24 hr)
A-2	Before	620	650			
A-2	After	620	590	.0167	.0500	.0667
A-3	Before	620	650			
A-3	After	620	590	.0167	.0500	.0667
A-4	Before	580	610			
A-4	After	580	550	.0167	.0500	.0667
A-5	Before	610	640			
A-5	After	600	570	.0167	.0583	.0750
B-1	Before	600	630			
B-1	After	620	590	0	.0333	.0333
B-2	Before	600	630			
B-2	After	620	590	0	.0333	.0333
B-3	Before	620	650			
B-3	After	620	590	.0167	.0500	.0667
B-4	Before	620	650			
B-4	After	620	590	.0167	.0500	.0667
B-5	Before	640	670			
B-5	After	640	610	.0167	.0500	.0667
B-6	Before	590	620			
B-6	After	600	570	.0084	.0416	.0500
C-1	Before	620	650			
C-1	After	620	590	.0167	.0500	.0667
C-2	Before	640	670			
C-2	After	660	630	0	.0333	.0333
C-3	Before	620	650			
C-3	After	620	590	.0167	.0500	.0667
C-4	Before	600	630			
C-4	After	620	590	0	.0333	.0333
C-5	Before	610	640			
C-5	After	620	590	.0084	.0416	.0500
C-6	Before	610	640			
C-6	After	580	550	.0167	.0750	.0917
D-1	Before	620	650			
D-1	After	620	590	.0167	.0500	.0667
D-2	Before	620	650			
D-2	After	620	590	.0167	.0500	.0667
D-3	Before	630	660			
D-3	After	620	590	.0167	.0583	.075
D-4	Before	610	640			
D-4	After	620	590	.0084	.0416	.0500
D-5	Before	620	650			
D-5	After	620	590	.0167	.0500	.0667
D-6	Before	620	650			
D-6	After	620	590	.0167	.0500	.0667

Table 1 (Cont'd)

Gage #		Pressure Reading (psig)	Corrected Readings* (psig)	Temperature Correction (#/24 hr)	ΔP Correction (#/24 hr)	Total Leakage (#/24 hr)
D-6	Before	620	650			
D-6	After	620	590	.0167	.0500	.0667
E-1	Before	620	650			
E-1	After	600	570	.0167	.0666	.0833
E-2	Before	600	630			
E-2	After	600	570	.0167	.0500	.0667
E-3	Before	620	650			
E-3	After	600	570	.0167	.0666	.0833
E-4	Before	600	630			
E-4	After	600	570	.0167	.0500	.0667
E-5	Before	620	650			
E-5	After	600	570	.0167	.0666	.0833
E-6	Before	620	650			
E-6	After	600	570	.0167	.0666	.0833
F-2	Before	620	650			
F-2	After	620	590	.0167	.0500	.0667
F-3	Before	620	650			
F-3	After	620	590	.0167	.0500	.0667
F-4	Before	620	650			
F-4	After	600	570	.0167	.0666	.0833
F-5	Before	620	650			
F-5	After	600	590	.0167	.0500	.0667
Total						2.0753

*30 psi added to before reading and 30 psi subtracted from after reading for gauge accuracy and readibility.