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NOV 22 1994

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SERIAL: BSEP 94-0460
10 CFR 50.12

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

BRUNSWICK NUCLEAR PLANT, UNIT NO. 1
DOCKET NO. 50-325/LICENSE NO. DPR-71
REQUEST FOR EXEMPTION TO 10 CFR 50, APPENDIX J
CONTAINMENT INTEGRATED LEAKAGE RATE TESTING

Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Parts 50.12, Carolina Power & Light Company hereby requests a one-time schedular exemption from the requirements of 10 CFR Part 50, Appendix J, Section III.D.1.(a) for the Brunswick Steam Electric Plant (BSEP), Unit No. 1. The exemption would allow the third Type A overall containment integrated leakage rate test to be performed during the last refueling outage of the current 10-year service period by extending the current Type A test interval from 50 months to 67 months. This would extend the current Type A interval until the scheduled start of the Reload 10 outage (i.e., September 1996). The NRC has previously approved similar requests for schedular exemptions of up to 15 months for Type A testing in accordance with Appendix J to 10 CFR Part 50. Enclosure 1 provides a detailed description and basis for the requested schedular exemption.

Approval of this schedular exemption is needed by March 1, 1995 in order to support planning activities for the upcoming Unit 1 refueling outage, which is currently scheduled to begin on April 1, 1995. Carolina Power & Light Company considers this request to be a cost-beneficial licensing action for the Brunswick Plant, with anticipated savings in excess of \$750,000 for the plant. CP&L therefore requests that the proposed schedular exemption be given appropriate review priority.

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Please refer any questions regarding this submittal to Mr. R. P. Lopriore at
(910) 457-2212.

Sincerely,



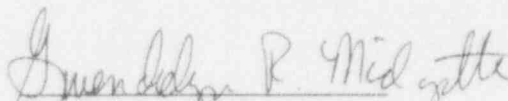
Roy A. Anderson

WRM/wrm (94-0460.wp)

Enclosures:

1. Basis for Scheduling Exemption Request
2. Unit 1 Integrated Leak Rate Test Timeline

Roy A. Anderson, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, and agents of Carolina Power & Light Company.


Notary (Seal)

My commission expires: August 12, 1996

pc: Mr. S. D. Ebner, Regional Administrator, Region II
Mr. P. D. Milano, NRR Senior Project Manager - Brunswick Units 1 and 2
Mr. C. A. Patterson, Brunswick NRC Senior Resident Inspector
The Honorable H. Wells, Chairman - North Carolina Utilities Commission

ENCLOSURE 1

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1
NRC DOCKET NO. 50-325
OPERATING LICENSE NO. DPR-71
REQUEST FOR EXEMPTION TO 10 CFR 50, APPENDIX J
CONTAINMENT INTEGRATED LEAKAGE RATE TESTING

BASIS FOR SCHEDULAR EXEMPTION REQUEST

Introduction

10 CFR 50.54(o) requires that primary reactor containments for water cooled power reactors be subject to the requirements set forth in 10 CFR Part 50, Appendix J. 10 CFR 50, Appendix J, section III.D.1.(a) states:

After the preoperational leakage rate tests, a set of three Type A tests shall be performed, at approximately equal intervals during each ten year service period. The third test of each set shall be conducted when the plant is shutdown for the 10-year plant inservice inspection.

Due to the operating history of Unit 1, adherence with 10 CFR Part 50, Appendix J, section III.D.1.(a) will require the performance of 4 Type A tests during the current 10-year service period. The purpose of this submittal is to request an exemption from the schedular requirements of Appendix J to 10 CFR Part 50. This exemption would extend the period between the second and third Type A test by approximately 18 months. This exemption obviates the need for an unnecessary fourth Type A test during the second 10-year service period.

Discussion of Circumstances and Need

The second 10-year service period for the Brunswick Nuclear Plant (BNP) began July 10, 1986 and ends July 10, 1997. Brunswick Nuclear Plant, Unit 1 experienced an extended shutdown during the period between April 1992 and February 1994. By letter dated August 5, 1994 (Reference 1), Carolina Power & Light Company notified the NRC that the original 10-year service period end date (July 10, 1996) was being extended by one year due to this extended outage. This one-year period extension is allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI, IWA-2400(c).

During the first 10-year service period, the inservice inspection periods for Unit 1 and Unit 2 were not aligned. To accomplish this realignment, the end of the Unit 1 10-year service period was moved backward to coincide with the end of the first 10-year service period for Unit 2 (References 2, 3, and 4). As a result of these realignments, the first Unit 1 Type A test of the second 10-year service period was performed only 11 months into the interval (i.e., in May 1987). The "early" performance of the initial Type A test for the second 10-year interval was due to the realignment of the Unit 1 and Unit 2 inspection intervals. The second Type A test of the second 10-year service period was performed within the stipulated "approximately equal" 40 ± 10 month interval (i.e., in

February 1991). The results of these tests were as follows:

- First Type A Test: The "as-left" leakage was satisfactorily measured as 0.215 weight-percent per day. The "as found" leakage was accounted as being greater than $0.75L_a$ due to Type C leakage test failures on the feedwater "A" loop and the containment atmosphere monitor discharge line (CAC-AT-1262). The identified local leak rate failures were repaired during the refueling outage. The results of this Type A test were submitted to the NRC in a letter dated July 29, 1987 (Reference 5).
- Second Type A Test: The "as-left" leakage was satisfactorily measured as 0.340 weight-percent per day. The "as found" leakage was accounted as being greater than $0.75L_a$ due to Type C leakage test failures on the feedwater "B" loop the reactor water clean-up system suction line, and the reactor core isolation cooling turbine steam supply line. The identified local leak rate failures were repaired during the refueling outage. The results of this Type A test were submitted to the NRC in a letter dated May 3, 1991 (Reference 6). NRC inspection activities associated with this Type A test were documented in NRC Inspection Report No. 50-325/93-12 and 50-324/93-12 (Reference 7).

Brunswick Unit 1 experienced an extended shutdown during the period between April 1992 and February 1994. The duration of this out-of-service period was approximately twenty-two (22) months. Because of this extended shutdown, the remaining two Unit 1 refueling outages (Reload 9 and Reload 10) in the second in-service inspection period were rescheduled. The Reload 9 outage is now rescheduled to begin April 1, 1995; the Reload 10 outage is now rescheduled to begin September 5, 1996.

Appendix J of 10 CFR Part 50 requires that Type A tests be conducted at "approximately equal intervals during each 10-year service period." Based on the extended outage that occurred from April 1992 until February 1994 and the performance date of the last Type A test (February 1991), the next Type A test must be performed during the Reload 9 outage (April 1995) unless this schedular exemption request is granted. In addition, based on the requirement of Appendix J to 10 CFR Part 50 that the final Type A test "...be conducted when the plant is shutdown for the 10-year plant inservice inspection," a fourth primary containment integrated leakage rate test will be needed during the final outage of the current 10-year service period (i.e., during the Reload 10 outage that currently is scheduled to begin in September 1996).

Discussion of Basis

The purpose of performing Type A tests (i.e., Containment Integrated Leakage Rate Tests (CILRT)) is to demonstrate that the total leakage from the primary containment does not exceed the maximum allowable leakage rate (designated as L_a) specified in the Technical Specifications, the Updated Final Safety Analysis Report, and Appendix J to 10 CFR Part 50. The maximum allowable primary containment leakage rate for the Brunswick units is 0.5 percent by weight per 24 hours at the peak accident pressure of 49 psig. The Technical Specifications and 10 CFR 50, Appendix J require the measured Type A test to be less than or equal to 75 percent of L_a (0.375 percent by weight per 24 hours) to allow for deterioration of leakage paths during the month intervals between Type A tests. The

containment maximum allowable leakage rate, L_a , provides an input assumption to the calculation for ensuring the maximum allowable offsite dose during a design basis accident does not exceed that specified in 10 CFR 100.

The proposed exemption introduces the possibility that primary containment leakage in excess of the allowable value would remain undetected during the proposed 18 month extension of the interval between performance of the second and third Type A test. The primary containment structure is a passive component. During normal operating conditions, there are no environmental or operational stresses that could contribute to the degradation of the primary containment. The Unit 1 Technical Specifications impose requirements on containment temperature and pressure that must be maintained; therefore, the primary containment is not exposed to elevated temperatures or pressures during normal operating conditions. Likewise, during out-of-service periods, the primary containment is not exposed to elevated temperatures or pressures.

The types of mechanisms that could cause the degradation of the containment can be categorized into two types: degradation due to work which is performed as part of a modification or maintenance activity (i.e., activity based) or degradation resulting from a time-based failure mechanism. Based on a review of activities conducted since the second Type A test of the interval, CP&L has concluded that there have not been any alterations or challenges to the Unit 1 primary containment that have adversely affected Unit 1 primary containment leakage rates. A review of previous containment integrated leakage rate tests was performed. The only failure mechanisms which have been detected during the past Type A tests are activity-based; no time-based mechanisms were identified that would not have been detected during the performance of Type B and Type C testing. The majority of the leakage detected during the two Type A tests of the second 10-year service period is attributable to containment penetrations and not the containment barrier itself.

Based on a review of activities planned for the upcoming Reload 9 outage and for Cycle 10 operation, CP&L has concluded that there will not be any outage or future maintenance activities during the proposed test interval extension that will adversely affect the Unit 1 primary containment leakage rates without administrative controls requiring performance of an individual local leak rate test. Furthermore, any future maintenance activities during the proposed interval extension that have the potential to affect primary containment leakage rates will also include administrative controls that require the performance of an individual local leak rate test. Type B and C testing will identify most of the primary containment leakage. As noted earlier, the Brunswick Type B and Type C testing program will continue to be performed throughout the proposed interval extension. Therefore, CP&L has concluded that the risk of a non-detectable increase of Unit 1 primary containment leakage is negligible. CP&L also plans to continue reviewing any future modifications that may be initiated during the requested extension period to ensure that these new modifications do not have the potential to adversely affect the Unit 1 containment leakage rate.

The containment barrier is inspected as part of routine maintenance activities conducted during the performance of each Type A test. These inspections are implemented through plant procedure PT-20.5.1. The purpose of the procedure is to provide guidelines for the performance of a visual inspection of the accessible interior and exterior surfaces of the primary containment. Significant structural deterioration or apparent changes in appearance of the primary containment surfaces that may affect the integrity of the

containment are recorded during performance of these inspections. Any abnormal degradation would be reported to the Nuclear Regulatory Commission. The inspection procedure was successfully performed during the Reload 7 outage and no abnormal degradation of the containment was identified. In the event this schedular exemption request is approved, CP&L will perform an inspection of the containment barrier during the Reload 9 outage using plant procedure PT-20.5.1 even though the Type A test would not be conducted.

Type B and Type C testing (i.e., local leak rate testing for containment isolation valves, penetrations, and air locks) is conducted more frequently than the Type A tests (i.e., every 24 months). Performance of these tests provides further assurance that the integrity of the primary containment boundary is being maintained. The test intervals and acceptance criteria for Type B and Type C tests are not affected by this requested schedular exemption.

The failure effects that are potentially created by the requested schedular exemption have been considered. The risk impact of containment structural leakage is to create an unacceptable release pathway for radionuclides in the event the containment is challenged in a severe accident. The BNP Individual Plant Examination (IPE) examined containment response during severe accidents and provides estimates of consequences. The IPE is used as the basis for qualitatively evaluating the impact of extending the Type A test interval. One significant insight gained from the IPE is that practically all containment failures are due to benign and self-regulating leak-before-break failures due to drywell head leakage, and that gross containment failures are very unlikely. This is due to the unique design of the Brunswick containment, which is a reinforced concrete containment with a steel liner. All other BWR Mark I containments are a steel shell, and the typical dominating failure is a shell rupture. The Brunswick drywell head flange seals are tested in accordance with the Type B test requirements of 10 CFR Part 50, Appendix J. Therefore, the proposed change will not have any impact on this type of failure mechanism.

The post-accident environment within containment may be severe and contribute to failure of its function. Such environments were considered as part of the IPE. The IPE found, however, that postulated containment failure under severe accident conditions is dominated by phenomenological effects (i.e., core debris interacting with the containment structure) associated with severe accidents. These containment failure mechanisms would not be impacted by the requested schedular exemption.

The containment isolation failure probability is dominated by active component failures, which are not affected by the proposed change to the Type A test schedule. The IPE found that containment isolation failure was not a significant failure mechanism. This probability is dominated by active failures of containment isolation valves to close on demand. The requested schedular exemption does not impact the probabilities for such demand failures.

NUREG/CR-4330 Volume 2, "Review of Light Water Reactor Regulatory Requirements" examined the risk impacts associated with increasing the allowable containment leakage using two different methods. The first method used existing Probabilistic Safety Assessments to calculate the incremental risk due to increasing the allowable containment leakage rate. The study concluded that overall plant risk is not very sensitive to changes in containment leakage rates. The study further concluded that accident risk for BWRs is

relatively insensitive to the containment leakage rate because the risk is dominated by accident sequences that result in failure or bypass of containment. This conclusion is consistent with the Brunswick IPE findings.

The second approach in NUREG/CR-4330 examined selected accident sequences and considered several additional measures, including individual radiation exposure and early health effects. This approach analyzed two specific BWR scenarios. The conclusions show that the two BWR scenarios would not exceed the 10 CFR 100 siting guidelines even at an assumed 50 percent weight per day leak rate.

Additional Information

On January 11, 1994, the NRC issued Amendment 167 to the Unit 1 Facility Operating License and an exemption to Appendix J of 10 CFR Part 50. The amendment and exemption authorized Unit 1 to return to a normal (i.e., the "approximately equal" intervals) Type A testing frequency instead of an accelerated (i.e., every plant refueling outage) test schedule. The exemption was requested due to discrepancies in interpreting the "as found" Type A test acceptance criteria. The exemption evaluated and approved the use of L_a as the "as found" Type A acceptance criteria.

On November 16, 1994 (Reference 8), CP&L submitted a license amendment request to adopt wording for primary containment integrated leak rate testing that is consistent with the requirements of the BWR-4 Improved Standard Technical Specifications (NUREG-1433). The proposed revision to Technical Specification 4.6.1.2 will remove the requirement that Type A tests be performed at 40 ± 10 month intervals and the specific acceptance criteria for these tests. Instead, the Technical Specifications will reference the containment integrated leakage testing requirements of Appendix J to 10 CFR Part 50. Issuance of the November 16, 1994 license amendment request is needed either prior to or in conjunction with approval of this requested schedular exemption; otherwise, another issue-specific license amendment will be required in order for this schedular exemption to be implemented upon NRC approval.

10 CFR 50.12 Analysis

Carolina Power & Light Company is requesting a one-time, temporary, schedular exemption from 10 CFR 50, Appendix J, section III.D.1.(a) which establishes the periodic test schedule for Type A (containment integrated leakage rate) tests, specifically the 10 CFR 50, Appendix J requirement that three Type A tests be performed at approximately equal intervals during the 10-year service period. Based on the technical justification discussed in this submittal, CP&L requests a one-time exemption of the requirements of 10 CFR 50, Appendix J, section III.D.1.(a) in accordance with three of the criteria of 10 CFR 50.12.

The NRC may, upon application, grant exemptions from the requirements of 10 CFR 50 where special circumstances are present. 10 CFR 50.12(a)(2)(ii) defines such a circumstance as "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule...." The underlying purpose of 10 CFR 50, Appendix J,

section III.D.1(a) is to establish and maintain a level of confidence that any primary containment leakage, during a hypothetical design basis accident, will remain less than or equal to the maximum allowable value, L_a , established by 10 CFR 50, Appendix J by performing periodic Type A testing. Based on the current Brunswick Unit 1 circumstances, Appendix J of 10 CFR Part 50 requires that the third Type A test and a fourth Type A test be performed during two consecutive Unit 1 refueling outages that are scheduled only about 18 months apart; which is not necessary to achieve the underlying purpose of the rule. The technical justification provided herein supports the conclusion that the requested schedular exemption to extend the third Unit 1 Type A test of the 10-year service period by 18 months will maintain the same level of confidence that any Unit 1 primary containment leakage will remain less than or equal to the maximum allowable leakage rate value, L_a , during the extension interval.

10 CFR 50.12(a)(2)(iii) states that the NRC may grant exemptions from requirements of 10 CFR 50 where "Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated...." The current Brunswick Unit 1 Type A test schedule established by 10 CFR 50, Appendix J, section III.D.1(a) will require that the Type A test be performed during two consecutive Unit 1 refueling outages. This current test schedule will result in additional personnel radiation exposure in order to perform the test (estimated to be approximately 1.5 REM) and costs associated with an increase in the refueling outage length of 3 days.

Furthermore, 10 CFR 50.12(a)(2)(v) states that the NRC may grant exemptions in special circumstances such as when "the exemption would provide only temporary relief from the applicable regulation and the licensee or applicant has made good faith efforts to comply with the regulation...." As discussed previously, performance of the Type A test is not necessary to provide assurance that the primary containment is maintained with a margin for deterioration of leak-tightness between tests. Thus, performance of the Type A test during the Unit 1 Reload 9 outage is not necessary to achieve the underlying purpose of Appendix J to 10 CFR Part 50. The requested exemption would provide only temporary relief pertaining only to the third Unit 1 Type A test interval and only the second 10-year service period. Approval of this one-time exemption would align the start of the third 10-year service period with the start of the third 10-year inservice inspection period. Nevertheless, provisions have been made to perform the third Unit 1 Type A test during the Reload 9 outage.

Environmental Assessment Information

The proposed action is to grant an exemption from 10 CFR 50, Appendix J, section III.D.1.(a), which requires a set of three Type A tests to be performed at approximately equal intervals during each 10-year service period and that the third test of each set be conducted when the plant is shut down for the performance of the 10-year inservice inspection (ISI). This one-time exemption would allow the third Unit 1 Type A test to be performed during the Reload 10 outage scheduled to begin in September 1996, approximately 18 months after the last Unit 1 test, thereby coinciding with the 10-year plant ISI refueling outage.

The proposed action will not significantly increase the probability of exceeding the maximum allowable value of expected primary containment leakage (i.e., L_a , as established by Appendix J of 10 CFR 50) during a hypothetical design basis accident. There is the potential that containment degradation could remain undetected during the proposed 18 month surveillance interval extension and result in the containment leakage exceeding the allowable value assumed in the safety analysis. However, the potential primary containment degradation mechanisms, including both activity-based and time-based causes, have been reviewed. This review concluded that there has not been any alteration or challenge to primary containment since the last Type A test, nor will there be any future maintenance activity during the proposed test interval extension, that will adversely affect primary containment leakage rate without administrative controls requiring the performance of local leak rate testing. Scheduled modifications during the requested extension period have also been reviewed, and CP&L has concluded that these modifications do not have the potential to adversely affect the integrity of the primary containment boundary. Therefore, these reviews concluded that the risk of a non-detectable increase of primary containment leakage is considered to be negligible because Type B and Type C testing required by Appendix J of 10 CFR Part 50 will identify the majority of the containment leakage.

The post-accident radiological releases associated with a design basis accident will not be greater than previously determined; therefore, the consequences of an accident will not be increased. The requested schedular exemption will not affect plant radiological effluents; therefore, there are no significant radiological environmental impacts associated with the requested exemption. Furthermore, the requested exemption does not affect non-radiological effluents and has no other environmental impact.

Since CP&L has concluded that there is no significant environmental impacts associated with the requested exemption, any alternatives would have either no or greater environmental impact. The principal alternative would be to deny the requested exemption; this would require the performance of a 10 CFR Part 50, Appendix J Type A test during two consecutive Unit 1 refueling outages. This would not reduce the environmental impact attributable to the facility as compared to the impact of granting the requested schedular exemption.

Summary

Carolina Power & Light Company has concluded that the requested schedular exemption will not result in a non-detectable Unit 1 primary containment leakage rate in excess of the maximum allowable value (0.5 percent weight by day) established in the Unit 1 Technical Specifications. This conclusion is based on the following:

- No modifications or maintenance activities that may have adversely affected the containment leakage rate have occurred since the last performance of the Type A test.
- Modifications and maintenance activities that will affect the containment leakage rates during the next refueling outage will include administrative controls requiring the performance of Type B or Type C tests, as appropriate.

- CP&L's commitment to perform an inspection of the containment barrier during the Reload 9 outage using plant procedure PT-20.5.1, even if the requested scheduler exemption is approved and the Type A test is not conducted.
- The Type B and Type C testing programs will effectively determine containment leakage caused by degradation of containment penetrations.
- Passive failure of the containment structure is extremely unlikely.
- Accident sequences with the potential to challenge the containment have calculated frequencies of occurrence that are very low.
- The containment failure mechanisms that are dominant in the IPE are associated with severe accident phenomena which cannot be detected by Type A testing.

References

1. Letter from R. P. Lopriore to the NRC Document Control Desk dated August 5, 1994, Serial Number BSEP 94-0302, "Second Inservice Inspection Interval Extension."
2. Letter from A. B. Cutter (CP&L) to D. B. Vassallo (USNRC) dated May 20, 1985, Serial Number NLS-85-034, "Inservice Inspection Intervals."
3. Letter from D. B. Vassallo (USNRC) to E. E. Utley (CP&L) dated July 1, 1985, "Inservice Inspection Intervals."
4. Letter from S. R. Zimmerman (CP&L) to Dan Mueller (USNRC) dated July 16, 1986, Serial Number NLS-86-248, "Inservice Inspection/Testing Intervals."
5. Letter from S. R. Zimmerman to the NRC Document Control Desk dated July 29, 1987, Serial Number NLS-87-147, "Integrated Leak Rate Test Results."
6. Letter from S. D. Floyd to the NRC Document Control Desk dated May 3, 1991, Serial Number NLS-91-127, "Integrated Leak Rate Test Results."
7. NRC Inspection Report Number 50-325/93-12 and 50-324/93-12 dated March 30, 1993.
8. Letter from R. A. Anderson to the NRC Document Control Desk dated November 16, 1994, Serial Number BSEP 94-0401, "Request for License Amendment, Containment Integrated Leakage Rate Testing."

ENCLOSURE 2

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1
NRC DOCKET NO. 50-325
OPERATING LICENSE NO. DPR-71
REQUEST FOR EXEMPTION TO 10 CFR 50, APPENDIX J
CONTAINMENT INTEGRATED LEAKAGE RATE TESTING

UNIT 1 INTEGRATED LEAK RATE TEST TIMELINE

Unit 1 Integrated Leak Rate Test Time Line

Unit 1 In-service Inspection Ten Year Period July 10, 1986 to July 10, 1997

1st INSPECTION INTERVAL
July 10, 1986 to November 10, 199

ILRT Completed May 1987
"As Left" leakage was 0.215 Wt%/Day
"As Found" leakage was greater than 0.75La
(0.375 Wt%/Day) because of Type C failures
on Feedwater A loop and Containment
Atmosphere Monitor (CAC-A7-1262) discharge
line.

2nd INSPECTION INTERVAL
November 11, 1989 to March 10, 1994

ILRT Completed February 1991
"As Left" leakage was 0.340 Wt%/Day
"As Found" leakage was greater than 0.75La
(0.375 Wt%/Day) because of Type C failures
on Feedwater B loop, RWCU suction line, and
RCIC turbine steam supply line.

Unit 1 Forced Outage
Started -- April 1992
Returned To Service -- February 1994
Total Duration -- 22 Months

3rd INSPECTION INTERVAL
March 11, 1994 to July 10, 1997

Original Period Completion was July 1996.
The inspection period was extended by one
year to coincide with plant outages because
of the Unit 1 and Unit 2 being out-of-service
by greater than one year.

Refuel Outage #9
Scheduled Start -- April 1995
Scheduled Finish -- June 1995

Refuel Outage #10
Scheduled Start -- September 1996
Scheduled Finish -- November 1996

Tech. Spec. ILRT Test Frequency is
40 + 10 Months.
Normal Test Frequency -- ILRT will be
past due in April 1995.
Addition of 22 Months (out-of-service time)
to Normal Test Frequency -- ILRT would
be past due in February 1997.

