PROPOSED CHANGE RTS-269 TO THE DUANE ARNOLD ENERGY CENTER TECHNICAL SPECIFICATIONS

The holders of license DPR-49 for the Duane Arnold Energy Center propose to amend Appendix A (Technical Specifications) to said license by deleting certain current pages and replacing them with the attached, new pages. The List of Affected Pages is given below.

LIST OF AFFECTED PAGES

iii	3.7-6
iv	3.7-22
3.7-1	3.7-23
3.7-2	3.7-24
3.7-3	3.7-42
3.7-4	6.11-5
3.7-4a	6.11-7
3.7-5	6.12-1 (new page)

SUMMARY OF CHANGES:

The following list of proposed changes is in the order that the changes appear in the Technical Specifications (TS).

Page	Description of Changes
iii, iv	Administrative changes were made to reflect the TS revision.
3.7-1	Current TS (CTS) that duplicate Appendix J, Option A were deleted. Administrative changes (such as renumbering) were also made. A cross reference to the new drywell airlock specification was added. A surveillance requirement (SR) was added to perform visual examinations and leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program. These changes are consistent with the guidance provided by the NRC by letter dated November 2, 1995 from C. Grimes (NRC) to D Modeen (NEI).

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3.7-2 CTS that duplicate Appendix J, Option A were deleted. CTS Sections 4.7.A.1.a(7) and 4.7.A.1.a(8) were also deleted. These sections contained the DAEC-specific values for P_a and L_a. This information is contained in the Bases.

3.7-3 CTS that duplicate Appendix J, Option A were deleted. Section 4.7.A.1.c(3) was renumbered and was revised consistent with the ITS. These changes are consistent with the guidance provided by the NRC by letter dated November 2, 1995 from C. Grimes (NRC) to D. Modeen (NEI).

3.7-4 CTS that duplicate Appendix J, Option A were deleted.

3.7-4a CTS that duplicate Appendix J, Option A were deleted and CTS 4.7.A.1.d.4 was renumbered to 4.7.A.1.c.

3.7-5 CTS that duplicate Appendix J, Option A were deleted.

CTS 4.7.A.1.e contains a requirement to replace the T-ring inflatable seals for the 18 inch purge valves every four years. This provision is not in the ITS as it is a maintenance issue and not a surveillance for operability. It will be relocated to plant procedures. CTS 4.7.A.1.e also contains a requirement to verify (during Type C testing) that the mechanical modification which limits the maximum opening angle for the 18 inch purge valves is intact. The ITS only require this surveillance if the mechanical modification is not permanent. At DAEC, the 18 inch purge valves are permanently blocked to restrict opening to 30°. These CTS 4.7.A.1.e provisions will be relocated to plant procedures. Any change to these requirements will require an evaluation in accordance with 10 CFR 50.59

CTS 4.7.A.1.e also contains a stipulation that the Cycle 6/7 refueling outage establishes the baseline for replacement of the T-ring inflatable seals for the containment purge valves. Since this is a one time provision that has been completed, its deletion is considered an administrative change.

3.7-6 CTS that duplicate Appendix J, Option A were deleted.

Requirements for primary containment air lock operability were added. Primary containment air lock leakage rate requirements were added as supporting surveillances for primary containment air lock operability. These changes are consistent with the guidance provided by the NRC by letter dated November 2, 1995 from C. Grimes (NRC) to D. Modeen (NEI).

3.7-22 The Bases were revised to be consistent with the changes to the TS pages.

3.7-23 The Bases were revised to be consistent with the changes to the TS pages.

3.7-24 The Bases were revised to be consistent with the changes to the TS pages.

3.7-42 The References were revised to reflect the changes to the Bases.

6.11-5 The requirement to report the results of the Reactor Containment Integrated Leakage Rate Test was deleted. The recordkeeping requirements of Option B will be followed.

6.11-7 The Table of Routine Reports was revised to reflect the elimination of the requirement to report the results of the Reactor Containment Integrated Leakage Rate Test.

6.12-1 The Primary Containment Leakage Rate Testing Program was added to TS Chapter 6. This change is consistent with the guidance provided by the NRC by letter dated November 2, 1995 from C. Grimes (NRC) to D. Modeen (NEI).

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.7 PLANT CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment and Primary Containment Air Lock ~~Integrity~~

1. Primary Containment

~~1.~~ a. PRIMARY CONTAINMENT INTEGRITY shall be maintained at all times when the reactor is critical or when the temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 Mw(t). Compliance with Subsections 3.7.A.2.b, 3.7.A.2.c, 3.7.A.2.d and 3.7.B.2 satisfies the requirement to maintain PRIMARY CONTAINMENT INTEGRITY.

~~2.~~ b. Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 PLANT CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment system integrity.

Objective:

To verify the integrity of the primary and secondary containments.

Specification:

A. Primary Containment and Primary Containment Air Lock ~~Integrity~~

1. PRIMARY CONTAINMENT ~~INTEGRITY~~ ~~shall be demonstrated as follows:~~

~~a. Type A Test~~

~~Primary Reactor Containment Integrated Leakage Rate Test~~

~~1) The interior surfaces of the drywell and torus shall be visually inspected each operating cycle for evidence of deterioration. In addition, the external surfaces of the torus below the water level shall be inspected on a routine basis for evidence of torus corrosion or leakage.~~

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~~Except for the initial Type A test, all Type A tests shall be performed without any preliminary leak detection surveys and leak repairs immediately prior to the test.~~

~~If a Type A test is completed but the acceptance criteria of Specification 4.7.A.1.a.(8) is not satisfied and repairs are necessary, the Type A test need not be repeated provided locally measured leakage reductions, achieved by repairs, reduce the containment's overall measured leakage rate sufficiently to meet the acceptance criteria.~~

~~2) Closure of containment isolation valves for the Type A test shall be accomplished by normal mode of actuation and without any preliminary exercising or adjustments.~~

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a. Perform required visual examinations and leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

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3) ~~The containment test pressure~~ shall be allowed to stabilize for a period of about 4 hours prior to the start of a leakage rate test.

4) The reactor coolant pressure boundary shall be vented to the containment atmosphere prior to the test and remain open during the test.

5) Test methods are to comply with ANSI N45.4-1972.

6) The accuracy of the Type A test shall be verified by a supplemental test. An acceptable method is described in Appendix C of ANSI N45.4-1972.

7) Periodic Leakage Rate Tests

Periodic leakage rate tests shall be performed at or above the peak pressure (Pa) of 43 psig.

8) Acceptance Criteria

The maximum allowable leakage rate (Lam) is 0.75 La, where La is defined as the design basis accident leakage rate of 2.0 weight percent of contained air per 24 hours at 43 psig.

9) Additional Requirements

If any periodic Type A test fails to meet the applicable acceptance criteria the test schedule applicable to subsequent Type A tests will be reviewed and approved by the Commission.

If two consecutive periodic Type A tests fail to meet the acceptance criteria of 4.7.A.1.a.(8) a Type A test shall be performed each operating cycle, or approximately every 18 months, whichever occurs first, until two consecutive Type A tests meet the subject acceptance criteria after which time the retest schedule of 4.7.A.1.d may be resumed.

b. Type B Tests

Type B tests refer to penetrations with gasketed seals, expansion bellows or other type of resilient ~~seals.~~

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	1) Test Pressure All Type B tests shall be performed by local pneumatic pressurization of the containment penetrations, either individually or in groups, at a pressure not less than Pa. 2) Acceptance Criteria The combined leakage rate of all penetrations subject to Type B and C tests shall be less than 0.60 La. c. Type C Tests 1) Type C tests shall be performed on containment isolation valves. Each valve to be tested shall be closed by normal operation and without any preliminary exercising or adjustments.
INSERT B	2) Acceptance criteria - The combined leakage rate for all penetrations subject to Type B and C tests shall be less than 0.60 La. 3) The leakage from any one main steam isolation valve shall not exceed 100 scf/hr at a test pressure of 24 psig.* The combined maximum pathway leakage rate for all four main steam lines shall not exceed 200 scf/hr at a test pressure of 24 psig. 4) The leakage rate from any containment isolation valve whose seating surface remains water covered post-LOCA, and which is hydrostatically Type C tested, shall be included in the Type C test total. * If a main steam isolation valve exceeds 100 scf/hr, it will be restored to ≤ 11.5 scf/hr.

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INSERT B

b. Verify leakage rate through each MSIV is ≤ 100 scfh when tested at ≥ 24 psig and that the combined maximum pathway leakage rate for all four main steam lines is ≤ 200 scfh when tested at ≥ 24 psig in accordance with the Primary Containment Leakage Rate Testing Program.*

*If the leakage rate through an individual MSIV exceeds 100 scfh, the leakage rate will be restored to ≤ 11.5 scfh.

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d. Periodic Retest Schedule

1) Type A Test

After the preoperational leakage rate tests, a set of three Type A tests shall be performed, at approximately equal intervals during each 10-year service period. (These intervals may be extended up to eight months if necessary to coincide with refueling outages.) The third test of each set shall be conducted when the plant is shut down for the 10-year plant in-service inspections.

The performance of Type A tests shall be limited to periods when the plant facility is nonoperational and secured in the shutdown condition under administrative control and in accordance with the plant safety procedures.

2) Type B Tests

a) Penetrations and seals of this type (except air locks) shall be leak tested at greater than or equal to 43 psig (P_a) during each reactor shutdown for major refueling or other convenient interval but in no case at intervals greater than two years.

b) The personnel airlock shall be pressurized to greater than or equal to 43 psig (P_a) and leak tested at least once every six (6) months. This test interval may be extended to the next refueling outage (up to a maximum interval between P_a tests of 24 months) provided there have been no airlock openings since the last successful test at P_a.

c) Within three (3) days after securing the airlock when containment integrity is required, the airlock gaskets shall be leak tested at a pressure of P_a.

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	~~3) Type C Tests~~ ~~Type C tests shall be performed during each reactor shutdown for major refueling or other convenient interval but in no case at intervals greater than two years.~~ C.4. Additional Periodic Tests Additional purge system isolation valve leakage integrity testing shall be performed at least once every three months in order to detect excessive leakage of the purge isolation valve resilient seats. The purge system isolation valves will be tested in three groups, by penetration: drywell purge exhaust group (CV-4302 and CV-4303), torus purge exhaust group (CV-4300 and CV-4301), and drywell/torus purge supply group (CV-4307, CV-4308 and CV-4306).

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	e. Seal Replacement and Mechanical Limiter The T-ring inflatable seals for purge isolation valves CV-4300, CV-4301, CV-4302, CV-4303, CV-4306, CV-4307 and CV-4308 shall be replaced at intervals not to exceed four years. During Type C testing, it shall be verified that the mechanical modification which limits the maximum opening angle for purge isolation valves CV-4300, CV-4301, CV-4302, CV-4303, CV-4306, CV-4307 and CV-4308 is intact. The baseline for this requirement shall be established during the Cycle 6/7 refuel outage. f. Containment Modification Any major modification, replacement of a component which is part of the primary reactor containment boundary, or resealing a seal-welded door, performed after the preoperational leakage rate test shall be followed by either a Type A, Type B, or Type C test, as applicable, for the area affected by the modification. The measured leakage from this test shall be included in this test report. The acceptance criteria as appropriate, shall be met. Minor modifications, replacements, or resealing of seal-welded doors, performed directly prior to the conduct of a scheduled Type A test do not require a separate test. g. Reporting Periodic tests shall be the subject of a summary technical report submitted to the Commission approximately 3 months after the conduct of each test. The report will be titled "Reactor Containment Integrated Leakage Rate Test." The results of the periodic testing performed to satisfy the requirements of 4.7.A.1.d.(4) shall be reported with the summary technical report prepared to provide the results of the testing performed in accordance with Section 4.7.A.1.d.(3).

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SURVEILLANCE REQUIREMENTS

~~The report shall include a schematic arrangement or description of the leakage rate measurement system, the instrumentation used, the supplemental test method, the test program selected, and all subsequent periodic tests. The report shall contain an analysis and interpretation of the leakage rate test data for the Type A test results to the extent necessary to demonstrate the acceptability of the containment's leakage rate in meeting the acceptance criteria.~~

~~For each periodic test, leakage test results from Type A, B, and C tests shall be reported. The report shall contain an analysis and interpretation of the Type A test results and a summary analysis of periodic Type B and Type C tests that were performed since the last Type A test. Leakage test results from Type A, B, and C tests that failed to meet the acceptance criteria shall be reported in a separate accompanying summary report. The Type A test summary report shall include an analysis and interpretation of the test data, the least-squares fit analysis of the test data, the instrumentation error analysis, and the structural conditions of the containment or components, if any, which contributed to the failure in meeting the acceptance criteria. Results and analyses of the supplemental verification test employed to demonstrate the validity of the leakage rate test measurements shall also be included.~~

~~The Type B and C tests summary report shall include an analysis and interpretation of the data and the condition of the components which contributed to any failure in meeting the acceptance criteria.~~

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2. Primary Containment Air Lock

a. When in RUN, STARTUP, or HOT SHUTDOWN MODE, the primary containment air lock shall be OPERABLE.

b. With one primary containment air lock door inoperable, verify the OPERABLE door is closed within 1 hour; lock the OPERABLE door closed within the following 23 hours; and verify the OPERABLE door is locked closed once per 31 days.[1, 2, 3, 4]

c. With the primary containment air lock interlock mechanism inoperable, verify an OPERABLE door is closed within 1 hour; lock an OPERABLE door closed within the following 23 hours; and verify an OPERABLE door is locked closed once per 31 days.[1, 2, 4, 5]

d. With the primary containment air lock inoperable for reasons other than 3.7.A.2.b or c above, immediately initiate action to evaluate primary containment overall leakage rate per 3.7.A.1, using current air lock test results; verify a door is closed within 1 hour; and restore air lock to OPERABLE status within the following 23 hours.[1, 2]

e. With Specifications 3.7.A.2.b, 3.7.A.2.c or 3.7.A.2.d not met, be in HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.[1, 2]

Note 1: Entry and exit is permissible to perform repairs of the air lock components.

Note 2: Take actions per Specification 3.7.A.1, "Primary Containment," when air lock leakage results in exceeding overall containment leakage rate acceptance criteria.

Note 3. Entry and exit is permissible for 7 days under administrative controls.

Note 4: Air lock doors in high radiation areas or areas with limited access due to inerting may be verified locked closed by administrative means.

Note 5: Entry into and exit from containment is permissible under the control of a dedicated individual.

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2. Primary Containment Air Lock

a. Perform required primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.[6, 7]

b. Once per 184 days, verify only one door in the primary containment air lock can be opened at a time.[8]

Note 6: An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

Note 7: Results shall be evaluated against acceptance criteria applicable to SR 4.7.A.1.a.

Note 8: Only required to be performed prior to startup following entry into primary containment when the primary containment is de-inerted.

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3.7.A & 4.7.A BASES:

Primary Containment ~~Integrity~~ and Primary Containment Air Lock

The integrity of the primary containment and operation of the core standby cooling system in combination, limit the offsite doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time, thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep offsite doses well below 10 CFR 100 limits.

In the event primary containment is inoperable, primary containment must be restored within 1 hour. The 1 hour time provides a period of time commensurate with the importance of maintaining primary containment and also ensures that the probability of an accident requiring primary containment during this time period is minimal.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response corresponding to the design basis loss-of-coolant accident. The peak drywell pressure would be about 43 psig which would rapidly reduce to 27 psig within 30 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises

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to about 25 psig within 30 seconds, equalizes with drywell pressure shortly thereafter and then rapidly decays with the drywell pressure decay, (Reference 1).* The primary containment is designed with a maximum allowable leakage rate (La) of 2.0% by weight of the containment air per 24 hours at the calculated maximum peak containment pressure (Pa) of 43 psig.

The design pressure of the drywell and suppression chamber is 56 psig, (Reference 2). ~~The design basis accident leakage rate is 2.0%/day at a pressure of 42 psig.~~ As pointed out above, the drywell and suppression chamber pressure following an accident would equalize fairly rapidly. Based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The design basis loss-of-coolant accident was evaluated by the AEC staff incorporating the primary containment design basis accident leak rate of 2.0%/day, (Ref. 3). The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90% for halogens, 90% for particulate iodine, and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about 2 rem and the maximum thyroid dose is about 32 rem at the site boundary over an exposure duration of two hours. The resultant thyroid dose that would occur over the course of the accident is 98 rem at the boundary of the low population zone (LPZ). Thus, these doses are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment, resulting in a direct release of fission products from the primary containment through the filters and stack to the environs.

*NOTE: The initial leak rate testing performed during plant startup was conducted at a pressure of 54 psig in accordance with the original FSAR analysis of peak containment pressure (Pa).

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Therefore, the specified primary containment leak rate is conservative and provides additional margin between expected offsite doses and 10 CFR 100 guidelines.

~~The design basis accident leak rate (L_a) at the peak accident pressure of 43 psig (P_a) is 2.0 weight percent per day. To allow a margin for possible leakage deterioration during the interval between Type A tests, the maximum allowable containment operational leak rate (L_{am}), is 0.75 L_a.~~

~~Type B and Type C tests are performed on testable penetrations and isolation valves during the interim period between Type A tests. This provides assurance that components most likely to undergo degradation between Type A tests maintain leaktight integrity. A controlled list of the testable penetrations and isolation valves subject to Type B and Type C testing is located in the plant Administrative Control Procedures.~~

~~The containment leakage testing program is based on NRC guidelines for development of leak rate testing and surveillance schedules for reactor containment vessels, (Reference 4).~~

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3.7.B and 4.7.B Bases

Primary Containment Power Operated Isolation Valves

Automatic isolation valves are provided on process piping which penetrates the containment and communicates with the containment atmosphere. The maximum closure times for these valves are selected in consideration of the design intent to contain released fission products following pipe breaks inside containment. Several of the automatic isolation valves serve a dual role as both reactor coolant pressure boundary isolation valves and containment isolation valves. The function of such valves on reactor coolant pressure boundary process piping which penetrates containment (except for those lines which are required to operate to mitigate the consequences of a loss-of-coolant accident) is to provide closure at a rate which will prevent

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Primary containment OPERABILITY is maintained by limiting leakage to less than or equal to 1.0 L_a, except prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met.

Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. Failure to meet air lock leakage testing, purge valve leakage testing, or main steam isolation valve leakage does not necessarily result in a failure of surveillance requirement 4.7.A.1.a. The impact of the failure to meet these SRs must be evaluated against the Type A, B, and C acceptance criteria of the Primary Containment Leakage Rate Testing Program.

One double door primary containment air lock has been built into the primary containment to provide personnel access to the drywell and to provide primary containment isolation during the process of personnel entering and exiting the drywell. The air lock is designed to withstand the same loads, temperatures, and peak design internal and external pressures as the primary containment. As part of the primary containment, the air lock limits the release of radioactive material to the environment during normal unit operation and through a range of transients and accidents up to and including postulated DBAs.

Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a DBA in primary containment. Each of the doors contains a single gasketed seal to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in primary containment internal pressure results in increased sealing force on each door).

The air lock is nominally a right circular cylinder, 12 ft in diameter, with doors at each end that are interlocked to prevent simultaneous opening. During periods when primary containment is not required to be OPERABLE, the air lock interlock mechanism may be disabled, allowing both doors of the air lock to remain open for extended periods when frequent primary containment entry is necessary. Under some conditions, as allowed by the primary containment air lock LCO, the primary containment may be accessed through the air lock, when the interlock mechanism has failed, by manually performing the interlock function.

The primary containment air lock forms part of the primary containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining primary containment leakage rate to within limits in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the safety analysis.

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For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door to be opened at a time. This provision ensures that a gross breach of primary containment does not exist when primary containment is required to be OPERABLE. Closure of a single door in the air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry and exit from primary containment.

The air lock interlock mechanism is designed to prevent simultaneous opening of both doors in the air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum-expected post accident primary containment pressure, closure of either door will support primary containment OPERABILITY. Thus, the interlock feature supports primary containment OPERABILITY while the air lock is being used for personnel transit into and out of the containment.

Maintaining the primary containment air lock OPERABLE requires compliance with the leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. The acceptance criteria were established during initial air lock and primary containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall primary containment leakage rate. The frequency is required by the Primary Containment Leakage Rate Testing Program.

Testing of the air lock requires the installation of a strongback on the inner door to keep it closed during testing, since the air lock is tested by pressurizing the space between the inner and outer doors. Without the strongback, the inner door could be forced open by the pressure against it in the non-accident direction. Opening the air lock door to remove the strongback (or other test equipment), does not require further leak testing, as long as the inner door seal is not disturbed.

The primary containment air lock surveillance requirements have been modified by two notes. One note states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. The other note requires the results of air lock leakage tests be evaluated against the acceptance criteria of the Primary Containment Leakage Rate Testing Program (TS Section 6.12). This ensures that the air lock leakage is properly accounted for in determining the combined Type B and C primary containment leakage.

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3.7.A & 4.7.A REFERENCES

1. "Duane Arnold Energy Center Power Uprate", NEDC-30603-P, May, 1984 and Attachment 1 to letter L. Lucas to R.E. Lessly, "Power Update BOP Study Report," June 18, 1984.

2. ASME Boiler and Pressure Vessel Code, Nuclear Vessels, Section III, maximum allowable internal pressure is 62 psig.

3. Staff Safety Evaluation of DAEC, USAEC, Directorate of Licensing, January 23, 1973.

4. ~~10 CFR Part 50, Appendix J, Reactor Containment Testing Requirements, Federal Register, April 19, 1976.~~ Deleted 2

5. Deleted

6. Deleted

7. General Electric Company, Duane Arnold Energy Center Suppression Pool Temperature Response, NEDC-22082-P, March 1982.

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d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.11.3 UNIQUE REPORTING REQUIREMENTS

Special reports shall be submitted to the Director of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification.

a. Reactor vessel base, weld and heat affected zone metal test specimens (Specification 4.6.A.2).

b. deleted

c. Inservice inspection (Specification 4.6.G.).

d. deleted ~~Reactor Containment Integrated Leakage Rate Test (Specification 4.7.A).~~

e. deleted

f. deleted

g. deleted

h. Radioactive Liquid or Gaseous Effluent - calculated dose exceeding specified limit (ODAM Sections 6.1.3, 6.2.3 and 6.2.4).

i. Off-Gas System inoperable (ODAM Section 6.2.5).

j. Measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level values of ODAM Table 6.3-3 when averaged over any calendar quarter sampling period (ODAM Section 6.3.2.1).

k. Annual dose to a MEMBER OF THE PUBLIC determined to exceed 40 CFR Part 190 dose limit (ODAM Section 6.3.1.1).

l. Radioactive liquid waste released without treatment when activity concentration is equal to or greater than 0.01μci/ml (ODAM Section 6.1.4.1).

m. Explosive Gas Monitoring Instrumentation Inoperable (Specification 3.2.I.1).

n. Liquid Holdup Tank Instrumentation Inoperable (Specification 3.14.B.1).

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TABLE 6.11-1 (cont)

REPORTING SUMMARY - ROUTINE REPORTS

Requirement	Report	Timing of Submittal
§50.59(b)	Changes, Tests, and Experiments	Within 6 months after each REFUELING OUTAGE.
§70.53	Special Nuclear Material Status	Within 30 days after March 31 and September 30 of each year.
§70.54	Transfer of Special Nuclear Material	Promptly upon transfer
§70.54	Receipt of Special Nuclear Material	Within 10 days after material is received
Appendix G to 10 CFR Part 50	Fracture Toughness	On an individual-case basis at least 3 years prior to the date when the predicted fracture toughness levels will no longer satisfy section V.B. of Appendix G to 10 CFR Part 50.
Appendix H to 10 CFR Part 50	Reactor Vessel Material Surveillance	Completion of tests after each capsule withdrawal.
Appendix I to 10 CFR Part 50	Annual Radioactive Material Release Report	On or before May 1.
Appendix I to 10 CFR Part 50	Annual Radiological Environmental Report	On or before May 1.
~~Appendix J to 10 CFR Part 50~~	~~Reactor Containment Building Integrated Leak Rate Test~~	~~Approximately 3 months following conduct of test.~~

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6.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 43 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 2.0 % of primary containment air weight per day.

Leakage Rate acceptance criteria are:

a. Primary Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first startup following testing in accordance with this program, the leakage rate acceptance criteria are: $\leq 0.60\ L_a$ for the Type B and Type C tests; and, $\leq 0.75\ L_a$ for the Type A tests;

b. The air lock testing acceptance criterion is overall air lock leakage rate $\leq 0.05\ L_a$ when tested at $\geq P_a$.

The 25% extension, per definition # 26 for Surveillance Frequency, does not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

RTS-269
12/95

SAFETY ASSESSMENT

1. INTRODUCTION

By letter dated December 22, 1995, IES Utilities Inc. submitted a request for revision to the Technical Specifications for the Duane Arnold Energy Center (DAEC). The proposed change adopts the guidance provided in NUREG 1433, Improved Standard Technical Specifications (ITS), for the performance of tests in accordance with the Primary Containment Leakage Rate Testing Program.

2. ASSESSMENT

Information already contained in 10 CFR 50, Appendix J was deleted and references to the Primary Containment Leakage Rate Testing Program were added. These are administrative changes to allow the use of performance based containment leakage testing methods. The proposed amendment requires compliance with the regulatory requirements of 10 CFR 50, Appendix J, Option B. Any exemptions to the requirements of 10 CFR 50, Appendix J require prior NRC approval.

The proposed Technical Specification change does not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor do they affect any assumptions or conditions in the accident analysis. No changes in either plant design or operational strategies will be made as a result of this revision. The use of Option B will significantly reduce the frequency of leak testing for highly reliable components provided their performance remains acceptable. This will result in reduced occupational radiological exposure, while at the same time assuring the performance of the containment safety functions as a barrier to the release of radioactivity to the environment. The addition of drywell air lock surveillance requirements provides further assurance that primary containment integrity will be maintained.

The proposed revision does not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor does it affect any assumptions or conditions in any of the accident analysis. The proposed revision does not degrade any existing plant programs, nor modify any functions of safety related systems or accident mitigation functions DAEC has previously been credited with. The proposed changes do not impact initiators of analyzed events. They also do not impact the assumed mitigation of accidents or transient events. These TS changes will not alter assumptions made in the safety analysis and licensing basis.

The proposed changes are consistent with NUREG-1433 which was approved by the NRC Staff and with NRC guidance provided for the implementation of Option B. Therefore, revising the CTS to reflect the NRC accepted level of detail and requirements ensures no reduction in a margin of safety.

Based upon the above assessment, we conclude that this request is acceptable.

ENVIRONMENTAL CONSIDERATION

10 CFR Section 51.22(c)(9) identifies certain licensing and regulatory actions which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; and (3) result in a significant increase in individual or cumulative occupational radiation exposure.

IES Utilities Inc. has reviewed this request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR Section 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows:

Basis

The change meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9) for the following reasons:

1. As demonstrated in Attachment 1 to this letter, the proposed amendment does not involve a significant hazards consideration.

2. The proposed amendment includes changes which delete information already contained in 10 CFR 50, Appendix J and adds references to the Primary Containment Leakage Rate Testing Program. These are administrative changes to allow the use of performance based containment leakage testing methods. The proposed amendment requires compliance with the regulatory requirements of 10 CFR 50, Appendix J, Option B. Any exemptions to the requirements of 10 CFR 50, Appendix J require prior NRC approval. No change in either plant design or operational strategies will be made as a result of this revision.

   Thus, there will be no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

3. The proposed amendment includes changes which delete information already contained in 10 CFR 50, Appendix J and adds references to the Primary Containment Leakage Rate Testing Program. These are administrative changes to allow the use of performance based containment leakage testing methods. The proposed amendment requires compliance with the regulatory requirements of 10 CFR 50, Appendix J, Option B. Any exemptions to the requirements of 10 CFR 50, Appendix J require prior NRC approval. The use of Option B will significantly reduce the frequency of leak testing for highly reliable components provided their performance remains acceptable. This will result in reduced occupational radiological exposure, while at the same time assuring the performance of the containment safety functions as a barrier to the release of radioactivity to the environment. No changes in either plant design or operational strategies will be made as a result of this revision.

   Thus, there will be no significant increase in either individual or cumulative occupational radiation exposure.