50-331

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 4, 1996

Mr. Lee Liu
Chairman of the Board and
Chief Executive Officer
IES Utilities Inc.
Post Office Box 351
Cedar Rapids, IA 52406

SUBJECT: AMENDMENT NO. 219 TO FACILITY OPERATING LICENSE NO. DPR-49 - DUANE ARNOLD ENERGY CENTER (TAC NO. M94455)

Dear Mr. Liu:

The Commission has issued the enclosed Amendment No. 219 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. This amendment consists of changes to the Technical Specifications (TS) in response to your application dated December 22, 1995 and supplemented on September 20, 1996.

The amendment revises the Duane Arnold Energy Center (DAEC) Technical Specifications (TS) Sections 3.7.A and 4.7.A, "Primary Containment," by deleting information also contained in 10 CFR Part 50, Appendix J, Option A and incorporating references to the Primary Containment Leakage Rate Testing Program. The amendment allows the use of the performance based option of containment leak testing. The amendment adds Operability and Surveillance Requirements (SRs) for the drywell air lock. The amendment relocates certain requirements from the DAEC TS to licensee controlled documents including replacement of the T-ring inflatable seals for the 18 inch purge valves every four years and verification (during Type C testing) that the mechanical modification that limits the maximum angle for the 18 inch purge valve is intact. Minor administrative changes are also made. The amendment is consistent with comparable specifications in the Improved Standard Technical Specifications (ITS), NUREG-1433.

In addition, the staff has executed administrative changes and corrections to the TS Bases, as submitted in letters (2) dated February 13, 1995. Sections that changed or corrected are Section 1.2, Bases; Section 2.2, Bases Reactor Coolant System Integrity; Section 3.7.H/4.7.H, Bases Containment Atmosphere Dilution; and Section 3.7.I/4.7.I, Bases Oxygen Concentration.

A copy of the Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Glenn B. Kelly, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosures: 1. Amendment No. 219 to License No. DPR-49
2. Safety Evaluation

cc w/encls: See next page

A copy of the Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original signed by:

Glenn B. Kelly, Project Manager
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A copy of the Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's next biweekly Federal Register notice.

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Mr. Lee Liu
IES Utilities Inc.

Duane Arnold Energy Center

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

IES UTILITIES INC.

CENTRAL IOWA POWER COOPERATIVE

CORN BELT POWER COOPERATIVE

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 219
License No. DPR-49

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by IES Utilities Inc., et al., dated December 22, 1995, and supplemented September 20, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended to approve the relocation of certain Technical Specification requirements to licensee-controlled documents, as described in Licensee's application dated December 22, 1995, as supplemented on September 20, 1996, and reviewed in the Staff's safety evaluation report dated October 4, 1996. This license is also hereby amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-49 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 219, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of issuance and shall be implemented within 90 days of the date of issuance. Implementation shall include the relocation of Technical Specification requirements to the appropriate licensee-controlled document as identified in the Licensee's application dated December 22, 1995, as supplemented September 20, 1996, and reviewed in the Staff's safety evaluation report dated October 4, 1996.

FOR THE NUCLEAR REGULATORY COMMISSION

Glenn Kelly

Glenn B. Kelly, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: October 4, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 219

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by vertical lines.

Remove	Insert
iii	iii
iv	iv
1.2-4	1.2-4
1.2-5	1.2-5
3.7-1	3.7-1
3.7-2	3.7-2
3.7-3	3.7-3
3.7-4	3.7-4
3.7-4a	3.7-4a
3.7-5	3.7-5
3.7-6	3.7-6
3.7-22	3.7-22
3.7-23	3.7-23
3.7-24	3.7-24
---	3.7-24a
---	3.7-24b
3.7-35	3.7-35
3.7-42	3.7-42
6.11-5	6.11-5
6.11-7	6.11-7
---	6.12-1 (new page)

	LIMITING CONDITIONS FOR OPERATIONS		SURVEILLANCE REQUIREMENTS	PAGE NO.
3.7	Plant Containment Systems		4.7	3.7-1
	A.	Primary Containment and Primary Containment Air Lock	A	3.7-1
	B.	Primary Containment Power Operated Isolation Valves	B	3.7-7
	C.	Drywell Average Air Temperature	C	3.7-9
	D.	Pressure Suppression Chamber - Reactor Building Vacuum Breakers	D	3.7-10
	E.	Drywell - Pressure Suppression Chamber Vacuum Breakers	E	3.7-11
	F.	Deleted	F	3.7-12
	G.	Suppression Pool Level and Temperature	G	3.7-13
	H.	Containment Atmospheric Dilution	H	3.7-15
	I.	Oxygen Concentration	I	3.7-16
	J.	Secondary Containment	J	3.7-17
	K.	Secondary Containment Automatic Isolation Dampers	K	3.7-18
	L.	Standby Gas Treatment System	L	3.7-19
	M.	Mechanical Vacuum Pump	M	3.7-21
3.8	Auxiliary Electrical Systems		4.8	3.8-1
	A.	AC Power Systems	A	3.8-1
	B.	DC Power Systems	B	3.8-3
	C.	Onsite Power Distribution Systems	C	3.8-5
	D.	Auxiliary Electrical Equipment-CORE ALTERATIONS	D	3.8-5
	E.	Emergency Service Water System	E	3.8-6
3.9	Core Alterations		4.9	3.9-1
	A.	Refueling Interlocks	A	3.9-1
	B.	Core Monitoring	B	3.9-5
	C.	Spent Fuel Pool Water Level	C	3.9-6
	D.	Auxiliary Electrical Equipment-CORE ALTERATIONS	D	3.9-6
3.10	Additional Safety Related Plant Capabilities		4.10	3.10-1
	A.	Main Control Room Ventilation	A	3.10-1
	B.	Remote Shutdown Panels	B	3.10-2a
	C.	Control Building Chillers	C	3.10-2a
3.11	River Level Specification		4.11	3.11-1

		PAGE NO.
5.0	Design Features	5.1-1
5.1	Site	5.1-1
5.2	Reactor	5.2-1
5.3	Reactor Vessel	5.3-1
5.4	Containment	5.4-1
5.5	Spent and New Fuel Storage	5.5-1
5.6	Seismic Design	5.6-1
6.0	Administrative Controls	6.1-1
6.1	Management - Authority and Responsibility	6.1-1
6.2	Organization	6.2-1
6.3	Plant Staff Qualifications	6.3-1
6.4	Retraining and Replacement Training	6.4-1
6.5	Review and Audit	6.5-1
6.6	Reportable Event Action	6.6-1
6.7	Action to be Taken if a Safety Limit is Exceeded	6.7-1
6.8	Plant Operating Procedures	6.8-1
6.9	Radiological Procedures and Programs	6.9-1
6.10	Records Retention	6.10-1
6.11	Reporting Requirements	6.11-1
6.12	Primary Containment Leakage Rate Testing Program	6.12-1
6.13	Deleted	
6.14	Offsite Dose Assessment Manual	6.14-1
6.15	Process Control Program	6.15-1

design pressure (120% x 1150 = 1380 psig; 120% x 1325 = 1590 psig).

The analysis of the worst overpressure transient, a 3 second closure of all main steam isolation valves with a direct valve position scram failure (i.e., scram is assumed to occur on high neutron flux), shows that the peak vessel pressure experienced is much less than the code allowable overpressure limit of 1375 psig (Reference 1). Thus, the pressure safety limit is well above the peak pressure that can result from reasonably expected overpressure transients.

A SAFETY LIMIT is applied to the shutdown cooling suction piping of the Residual Heat Removal System (RHR) when it is operating in the shutdown cooling mode. While in shutdown cooling, the RHR system forms part of the reactor coolant system.

1.2 References

1. Supplemental Reload Licensing Submittal for Duane Arnold Atomic Energy Center, Unit 1.*

* Refer to analyses for the current operating cycle.

2.2 BASES

Reactor Coolant System Integrity

The discussion in section 3.6.D and 4.6.D Bases is applicable for discussion of pressure relief.

The design pressure of the RHR shutdown cooling suction piping is 175 psig. ANSI B31.1.0 permits pressure transients up to 15% over design pressure (1.15 x 175 = 201.25 psig) for durations of less than 10% of any 24 hour operating period or up to 20% over design pressure (120 x 175 = 210) if the event occurs less than 1% of any 24 hour operating period.

Maintaining reactor vessel dome pressure at or below 135 psig when operating a Residual Heat Removal pump in shutdown cooling mode ensures that the pressure inside the shutdown cooling suction piping is within the SAFETY LIMIT.

LIMITING CONDITIONS FOR OPERATION

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

1. Primary Containment

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.

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.

SURVEILLANCE REQUIREMENTS

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

1. Primary Containment

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

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.

c. 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).

LIMITING CONDITIONS FOR OPERATION

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.

SURVEILLANCE REQUIREMENTS

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 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 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 design pressure of the drywell and suppression chamber is 56 psig, (Reference 2). The primary containment is designed with a maximum allowable leakage rate (L_a) of 2.0% by weight of the containment air per 24 hours at the calculated maximum peak containment pressure (P_a) of 43 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.

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

*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).

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.

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. The 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.

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 operability of the whole system annually. The H_2 and O_2 analyzers are provided redundantly. There are two H_2 and O_2 analyzers. By permitting continued reactor operation at rated power with one of the two analyzers of a given type (H_2 or O_2) inoperable, redundancy of the analyzing capability will be maintained while not imposing an unnecessary interruption in plant operation.

Due to the nitrogen addition, the pressure in the containment after a LOCA could possibly increase with time. Under the worst expected conditions the containment pressure will reach 30 psig in approximately 70 days. If and when that pressure is reached, venting from the containment shall be manually initiated. The venting path will be through the Standby Gas Treatment System in order to minimize the offsite dose.

Following a LOCA, periodic operation of the drywell and torus sprays may be used to assist the natural convection and diffusion mixing of hydrogen and oxygen.

3.7.I and 4.7.I BASES

Oxygen Concentration

Safety Guide No. 7 assumptions for metal-water reactions result in hydrogen concentrations in excess of Safety Guide No. 7 flammability limit. By keeping oxygen concentrations less than 4%, Safety Guide No. 7 requirements are satisfied. The Containment Atmosphere Dilution System further assures that a combustible hydrogen/oxygen atmosphere will not be created in a post-LOCA condition.

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 Uprate 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. Deleted

5. Deleted

6. Deleted

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

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

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 0.01 µci/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).

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.

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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 219 TO FACILITY OPERATING LICENSE NO. DPR-49

IES UTILITIES INC.

CENTRAL IOWA POWER COOPERATIVE

CORN BELT POWER COOPERATIVE

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

On September 12, 1995, the U.S. Nuclear Regulatory Commission (NRC) approved issuance of a revision to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The NRC added Option B, "Performance-Based Requirements," to allow licensees to voluntarily replace the prescriptive testing requirements of 10 CFR Part 50, Appendix J, with testing requirements based on both overall performance and the performance of individual components.

By letters dated December 22, 1995, and September 20, 1996, IES Utilities Inc. (the licensee) requested changes to the Technical Specifications (TS) for the Duane Arnold Energy Center. The proposed changes would permit implementation of 10 CFR Part 50, Appendix J, Option B. The licensee has established a "Primary Containment Leakage Rate Testing Program" and proposed adding this program to the TS. The program references Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, which specifies a method acceptable to the NRC for complying with Option B.

2.0 BACKGROUND

Compliance with 10 CFR Part 50, Appendix J, provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate specified in the TS and Bases. The allowable leakage rate is determined so that the leakage rate assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety that impose a significant regulatory burden. Part 50, Appendix J, "Primary Containment Leakage Testing for Water-Cooled Power Reactors," was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program".

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B, "Performance-Based Requirements," to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, was developed as a method acceptable to the NRC staff for implementing Option B. This regulatory guide states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that the Regulatory Guide or other implementation document used by a licensee to develop a performance-based leakage testing program must be included, by general reference, in the plant TS. The licensee has referenced Regulatory Guide 1.163, dated September 1995, in the proposed DAEC TS.

Regulatory Guide 1.163, dated September 1995, specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TS to implement Option B. After some discussion, the staff and NEI agreed on final TS which were transmitted to NEI in a letter dated November 2, 1995. These TS are to serve as a model for licensees to develop plant specific TS in preparing amendment requests to implement Option B.

In order for a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 EVALUATION

The licensee's December 22, 1995, and September 20, 1996, letters to the NRC propose to establish a "Primary Containment Leakage Rate Testing Program" and propose to add this program to the TS. The program references Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, which specifies methods acceptable to the NRC for complying with Option B. This requires a change to existing TS 3.7.A, 4.7.A, 6.11.3, Table 6.11-1, and the addition of the "Primary Containment Leakage Rate Testing Program" to Section 6.12. Corresponding bases were also modified. Option B permits a licensee to choose Type A; or Type B and C; or Type A, B and C; testing to be done on a performance basis. The licensee has elected to perform Type A, B, and C testing on a performance basis.

The TS changes proposed by the licensee are in compliance with the requirements of Option B and consistent with the guidance of Regulatory Guide 1.163, dated September 1995. Further, despite the different format of the licensee's current TS, all of the important elements of the model TS guidance provided in the NRC letter to NEI dated November 2, 1995, are included in the proposed TS. The licensee has proposed a change that either deviates from those in the model TS. It is discussed below.

Air Lock Door Seal Leak Rate Testing (Deviation)

The proposed TS 6.12, "Primary Containment Leakage Rate Testing Program," deviates from the model TS in that the air lock leakage rate testing acceptance criteria do not include a criterion for the separate testing of a door (door seals). This is because the air lock doors at DAEC are equipped with a single seal each, rather than dual seals, so the doors are not separately testable. The only way to test is to test the entire air lock; therefore, the proposed TS only includes an acceptance criterion for testing in that manner. The staff finds this acceptable.

3.1 Additional TS Changes

The licensee proposed several additional changes that are separate from the Appendix J, Option B TS modification request. Current TS 4.7.A.1.e., "Seal Replacement and Mechanical Limiter," requires the T-ring inflatable seals on seven containment purge isolation valves to be replaced at 4-year intervals; it also requires periodic verification that the mechanical modification which limits the maximum opening angle for the same seven valves is intact. The proposal would delete this TS.

3.1.1 Seal Replacement

The licensee's original submittal, dated December 22, 1995, requested deletion of the set interval for purge valve seal replacement. By letter dated September 20, 1996, the licensee provided additional details. The requirement was added to the TS in 1984 in response to generic staff concerns regarding purging and venting operations; also added was a requirement to leak test the valves every 3 months. The staff was concerned that potential rapid degradation of the resilient seals on containment purge/vent valves, due to wear, aging, and environmental exposure, could lead to undetected gross failure of the valves' leak-tight integrity. In addition to either ceasing purging/venting during plant operation or limiting it to a strict minimum, the staff requested more frequent testing (compared to the Appendix J interval [refueling]) and/or seal replacement on a set interval, but not necessarily both. The licensee conservatively chose to do both.

The licensee's testing experience since 1984 has been very good, with few problems, and none in the last 7 years. Further, licensee engineering evaluations have found a longer replacement interval (7.5 years) to be appropriate.

Seal replacement at a set interval is essentially an alternative to increased leak test frequency. The DAEC TS will retain the 3-month testing interval and the seal replacement requirements will be retained in the UFSAR and plant procedures. In 10 CFR 50.59(a)(1), it specifically states that changes made by a licensee to the UFSAR and plant procedures are subject to the requirements of 10 CFR 50.59.

Based on the above discussion, the staff finds that quarterly purge valve leak testing is sufficient for timely detection of seal degradation, and that the deletion from the TS of the set interval for seal replacement is therefore acceptable.

3.1.2 Mechanical Limiter

Periodic surveillance of the mechanical modification that limits the opening angle of the purge valves is only necessary if the limiter is not permanently installed, consistent with the Improved Standard TS. However, the subject purge valves are permanently blocked to restrict opening to 30 degrees. Therefore, the staff finds acceptable the deletion of this requirement from the TS.

3.2 Administrative Changes

This amendment also includes several minor administrative changes in the TS Table of Contents which the staff has reviewed and finds acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Iowa State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (61 FR 3499). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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