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 B. 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)

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.