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

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 174
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District (the licensee) dated June 14, 1993, as supplemented by letter dated April 12, 1994, 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 license 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 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-46 is hereby amended to read as follows:

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 174 , 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Hall

James R. Hall, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of Issuance: November 28, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 174

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

REMOVE PAGES	INSERT PAGES
ii	ii
iv	iv
5a	5a
185	-
186	-
187, 188, 189, 190, 191, 192	185, 186, 187, 188, 189, 190, 191, 192
216x	216x
216z	216z
216a8	216a8
216a21	216a21
216a22	216a22
216a26	216a26
226	226
229	229
231	231

TABLE OF CONTENTS (cont'd)

TABLE OF CONTENTS (Cont'd.)

W.A Solidification - SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems to a solid which is as uniformally distributed as reasonably achievable with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free-standing).

X. Spiral Reload - Pertains to the spiral reloading of the core with fuel, at least 50% of which has previously accumulated a minimum exposure of 1000 MWD/T.

Y. Surveillance Frequency - Surveillance requirements shall be applicable during the operational conditions associated with individual LCO's unless otherwise stated in an individual Surveillance Requirement.

Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.

Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with operability requirements for an LCO unless otherwise required by the specification.

The Surveillance Frequency establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance schedule and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillance that are performed at each refueling outage and are specified with an 18 month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of this definition is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

Z. Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations and examinations to be performed upon an instrument or component when it is required to be operable. These tests may be waived when the instrument, component or system is not required to be operable, but the instrument, component or system shall be tested prior to being declared operable or as practicable following its return to service.

Z.A Venting - Venting is the controlled process of discharging air or gas from a confinement to establish temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during venting. Vent, used in system names, does not imply a venting process.

Z.B Offsite - Offsite means outside of the exclusion area as defined in 10CFR Part 100.3. The exclusion area boundary around Cooper Nuclear Station is defined in Figure 1.1 and may also be referred to as the Site Boundary.

Z.C Member of the Public - A Member of the Public is a person in a controlled or unrestricted area who does not receive an occupational dose.

"INTENTIONALLY LEFT BLANK"

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.21 (Cont'd)

B. Liquid Effluents

Applicability: At all times.

Specification:

1. Concentration

a. The concentration of radioactive material in water OFFSITE (Figure 1.1) due to radioactive liquid effluent shall not exceed the concentration specified in 10 CFR Part 20.1302 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall not exceed 2×10^{-4} μCi/ml total activity.

b. With the concentration of radioactive material released OFFSITE exceeding the limit, attend to the cause without delay and restore the concentration within the above limit.

c. The provisions of Specification 6.5.2 do not apply.

4.21 (Cont'd)

B. Liquid Effluents

1. Concentration

a. Radioactive liquid wastes shall be sampled and analyzed according to Table 4.21.B.1.

b. The analytical results shall be used with methods in the ODAM to verify that the average concentration beyond the SITE BOUNDARY does not exceed Specification 3.21.B.1.a, when Sr-89, Sr-90 and Fe-55 concentrations are averaged over no more than 3 months and other radionuclide concentrations are averaged over no more than 31 days.

NOTES FOR TABLE 4.21.B.1

(1) The LLD is the smallest concentration of the radioactive material in a sample that will be detected with 95% probability (5% probability of falsely concluding that a blank observation represents a "real" signal).

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as picocurie per unit mass or volume),

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

V is the sample size (in units of mass or volume),

2.22 is the number of transformations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y, and Δt shall be used in the calculation.

(2) For certain radionuclides with low gamma yield or low energies, or for certain radionuclide mixtures, it may not be possible to measure radionuclides in concentrations near the LLD. Under these circumstances, the LLD may be increased inversely proportionally to the magnitude of the gamma yield (i.e., $5 \times 10^{-7}/I$, where I is the photon abundance expressed as a decimal fraction), but in no case shall the LLD, as calculated in this manner for a specific radionuclide, be greater than 10% of the values specified in 10 CFR 20, Appendix B, Table 2, Column 2.

(3) A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

(4) To be representative of the quantities and concentrations of radioactive materials in liquid effluents, daily grab samples shall be collected in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

NOTES FOR TABLE 4.21.C.1

(1) The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability (5% probability of falsely concluding that a blank observation represents a "real" signal).

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E\ .\ V\ .\ 2.22\ .\ Y\ .\ \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as picocurie per unit mass or volume),

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

V is the sample size (in units of mass or volume),

2.22 is the number of transformations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y, and Δt shall be used in the calculation.

(2) For certain radionuclides with low gamma yield or low energies, or for certain radionuclide mixtures, it may not be possible to measure radionuclides in concentrations near the LLD. Under these circumstances, the LLD may be increased inversely proportional to the magnitude of the gamma yield (i.e., $1 \times 10^{-4}/I$, where I is the photon abundance expressed as a decimal fraction), but in no case shall the LLD, as calculated in this manner for a specific radionuclide, be greater than 10% of the values specified in 10 CFR 20, Appendix B, Table 2, Column 1.

(3) Analyses shall also be performed following an increase as indicated by the gaseous release monitor of greater than 50% in the steady state release, after factoring out increases due to power changes or other operational occurrences, which could alter the mixture of radionuclides.

3.21 & 4.21 <u>BASES</u>

3.21.A & 4.21.A <u>INSTRUMENTATION</u>

3.21.A.1 & 4.21.A.1 <u>Liquid Effluent Monitoring</u>

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the release of radioactive material in liquid effluents. The OPERABILITY and use of these instruments implements the requirements of 10 CFR Part 50, Appendix A, General Design Criteria 60, 63, and 64. The alarm and/or trip setpoints for these instruments are calculated in the manner described in the ODAM to assure that the alarm and/or trip will occur before the limit specified in 10 CFR Part 20.1302 is exceeded. Control of the normal liquid discharge pathway is assured by station procedures governing locked discharge valves and valve line-up verification.

3.21.A.2 & 4.21.A.2 <u>Gaseous Effluent Monitoring</u>

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The location of this instrumentation is indicated by a Figure in the ODAM, a simplified flow diagram showing gaseous effluent treatment and monitoring equipment. The alarm/trip setpoints for these instruments shall be calculated in accordance with methods in the ODAM, which have been reviewed by NRC, to ensure that the alarm will occur prior to exceeding the limits of 10 CFR Part 20. The process monitoring instrumentation includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the augmented offgas treatment system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

In the event no flow rate measurement device is operable on a gaseous stream, alternative 24-hour estimates are adequate since the system design is constant flow and loss of flow is alarmed in the control room.

3.21.B & 4.21.B <u>LIQUID EFFLUENTS</u>

3.21.B.1 & 4.21.B.1 <u>Concentration</u>

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to unrestricted areas will be less than the concentration levels specified in 10 CFR Part 20.1302. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will not result in exposures within (1) the Section IV.A guides on technical specifications in Appendix I, 10 CFR Part 50, for an individual and (2) the limits of 10 CFR Part 20.1301 and 20.1302(b)(2)(i) to the population. The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

Since SERVICE WATER is not a normal or expected source of significant radioactive release, routine sampling and monitoring for radioactivity is precautionary. An activity concentration of 3×10^{-6} μCi/ml in SERVICE WATER effluent is diluted in the discharge canal to about 1.5% of the 10 CFR 20 Appendix B Table 2 Column 2 concentration with only one circulating water pump operating. During normal Station operation the dilution would be even greater. By monitoring SERVICE WATER effluent continuously for radioactivity and by confirmatory sampling weekly, reasonable assurance that its activity concentration can be kept to a small fraction of the 10 CFR Part 20.106 limit and within the Specification 3.21.B.2.a limit is provided.

By monitoring SERVICE WATER continuously and liquid radwaste continuously during discharge with the monitor set to alarm or trip before the limit specified in 10 CFR 20.1302 is exceeded, reasonable assurance of compliance with Specification 3.21.B.1.2 is provided. Verification that radioactivity in liquid effluent averaged only a small fraction of the concentration limit is provided by calculations demonstrating compliance with Specification 3.21.B.2.a.

3.21.B & 4.21.B LIQUID EFFLUENTS (Cont'd)

3.21.B.1 & 4.21.B.1 Concentration (Cont'd)

Compliance with 10CFR Part 20.1302 implies that the concentration limit represented by 10CFR Part 20, Appendix B, Table 2 will be met within a suitable and reasonable averaging time for assessing compliance. That averaging time is dependent upon the resolving time of the measurements or estimates which are used to evaluate compliance. Assessment of compliance is done by sampling and analysis according to Specification 4.21.B.1.2, by estimating or measuring the maximum release flow and the minimum dilution flow coincident during the period of release represented by the sample, and by computing the concentration as a fraction of the limit in the UNRESTRICTED area periodically on the basis of these data.

3.21.B.2 & 4.21.B.2 Liquid Dose

Specifications 3.21.B.2, 3.21.C.2 and 3.21.C.3 implement the requirements of 10 CFR Part 50.36a and of 10 CFR Part 50, Appendix I, Section IV. These specifications state LIMITING CONDITIONS FOR OPERATION (LCO) to keep levels of radioactive materials in LWR effluents as low as is reasonably achievable. Compliance with these specifications will also keep average releases of radioactive material in effluents at small percentages of the limits specified in 10 CFR Part 20.1301. Surveillance Requirements provide for the measurement of releases and calculation of doses to verify compliance with the Specifications. Action statements in these Specifications implement the requirements of 10 CFR Part 50.36(c)(2) and 10 CFR Part 50, Appendix I, Section IV.A in the event an LCO is not met. Annual dose limitations stated in Specifications 3.21.B.2, 3.21.C.2, and 3.21.C.3 are not strict limits as used elsewhere in the Technical Specifications (are not an immediate safety concern) but do obligate NPPD to take the applicable reporting action required in Specifications 3.21.B.2.b, 3.21.C.2.b, or 3.21.C.3.b.

10 CFR Part 50 contains two distinctly separate statements of requirements pertaining to effluents from nuclear power reactors. The first concerns a description of equipment to maintain control over radioactive materials in effluents, determination of design objectives, and means to be employed to keep radioactivity in effluents ALARA. This requirement is stated in Part 50, Section 34a and Appendix I, Section II. Appendix I, Section III stipulates that conformance with the guidance on design objectives be demonstrated by calculations (since demonstration is expected to be prospective). The other is a requirement for developing LIMITING CONDITIONS FOR OPERATION in technical specifications. It is stated in 10 CFR Part 50, Section 36a and Appendix I, Section IV. Both the intent of the Commission and the requirement are clearly stated in the Opinion of the Commission;[1] relevant paragraphs from that document follow:

> Section 50.36a(b) of 10 CFR Part 50 provides that licensees shall be guided by certain considerations in establishing and implementing operating procedures specified in technical specifications which take into account the need for operating flexibility and at the same time ensure that the licensee will exert his best efforts to keep levels of radioactive materials in effluents as low as practicable. The Appendix I that we adopt provides more specific guidance to licensees in this respect.

3.21 & 4.21 BASES (Cont'd)

3.21.C & 4.21.C GASEOUS EFFLUENTS

3.21.C.1 & 4.21.C.1 Concentration

Specification 3.21.C.1.a is included to assure that a measure of control is provided over the concentration of radionuclides in air leaving the exclusion area. Radioactive noble gases are monitored by instruments that provide a measure of release rate and cause automatic alarm when the noble gas concentration OFFSITE is expected to exceed the dose rate specified in 3.21.C.1.a. With prompt action to reduce the radioactive noble gas concentration in effluent following alarm initiation, it can be maintained at a small fraction of the annual limit. The specified release rate limits restrict the corresponding gamma and beta dose rates above background to an individual at or beyond the exclusion area boundary to ≤ 500 mrem/year to the total body or to ≤ 3000 mrem/year to the skin.

Radioiodines and radionuclides in particulate form are sampled with integrating samplers that permit assessment of the average release rate during each sample collection period. By complying with Specifications 3.21.C.2 and 3.21.C.3 the average OFFSITE concentration will be maintained at a small fraction of the 10 CFR Part 20.1302(b)(2)(i) concentration limit.

3.21.C.2 & 4.21.C.2 Noble Gases

Assessments of dose required by Specifications 4.21.C.2 and 4.21.C.3 to verify compliance with Appendix I, Section IV is based on measured radioactivity in gaseous effluent and on calculational methods stated in the ODAM. Pathways of exposure and location of individuals are selected such that the dose to a nearby resident is unlikely to be underestimated. Dose assessment methodology described in the ODAM for gaseous effluent will be consistent with the methodology in Regulatory Guides 1.109 and 1.111. Cumulative and projected assessments of dose made during a quarter are based on historical average, or reference (the same period of record used in the design objective Appendix I evaluation) atmospheric conditions. Assessments made for the annual radiological environmental report will be based on quarterly and annual averages of atmospheric conditions during the period of release.

The bases for Specifications 3.21.C.2 and 4.21.C.2 are also discussed in the bases for Specifications 3.21.B.2 and 4.21.B.2.

3.21.C.3 & 4.21.C.3 Iodine and Particulates

The bases for Specifications 3.21.C.3 and 4.21.C.3 are discussed in the bases for Specifications 3.21.B.2 and 4.21.B.2.

6.3 (Cont'd)

A. High Radiation Area

In lieu of the "control device" or "alarm signal" required by paragraph 20.1601 of 10 CFR Part 20, each high radiation area in which the deep dose equivalent in excess of 100 mrem but less than 1000 mrem in one hour* shall be barricaded (barricade will impede physical movement across the entrance or access to the high radiation area; i.e., doors, yellow and magenta rope, turnstile) and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Special Work Permit (SWP)**. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

1. A monitoring device which continuously indicates the radiation dose rate in the area.

2. A monitoring device which continuously integrates the radiation dose in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rates in the area have been established and personnel have been made knowledgeable of them.

3. A radiation protection qualified individual (i.e., qualified in radiation protection procedures), with a dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic dose rate monitoring at the frequency specified by Health Physics supervision.

In addition to the requirements of the above specification, areas accessible to personnel with dose rates such that a major portion of the body could receive in 1 hour a deep dose equivalent in excess of 1000 mrem* shall be provided with locked doors to prevent unauthorized entry. Doors shall remain locked except during periods of access by personnel under an approved SWP which shall specify the dose rates in the immediate work area. For individual high radiation areas accessible to personnel that are located within large areas, such as the containment, or areas where no enclosure exists for purposes of locking and no enclosure can be reasonably constructed around the individual areas, then that area shall be barricaded and conspicuously posted. Area radiation monitors that have been set to alarm if radiation levels increase, provide both a visual and an audible signal to alert personnel in the area of the increase. These monitors may be used to meet specification 6.3.A.1 provided that the dose rates and alarms have been established by radiation protection personnel. Stay times or continuous surveillance, direct or remote (such as use of closed circuit TV cameras), may be made by personnel qualified in radiation protection procedures to provide additional positive exposure control over the activities within the area.

---

*Measurement made at 12 inches from source of radiation.

**Radiation protection personnel or personnel escorted by radiation protection personnel shall be exempt from the SWP issuance requirement during the performance of their assigned duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

6.4.2.G (cont'd)

usage evaluation per the ASME Boiler and Pressure Vessel Code Section III was performed[1] for the conditions defined in the design specification. The locations to be monitored shall be:

a. The feedwater nozzles

b. The shell at or near the waterline

c. The flange studs

2. Monitoring, Recording, Evaluating, and Reporting

a. Operational transients that occur during plant operations will, at least annually, be reviewed and compared to the transient conditions defined in the component stress report for the locations listed in 1 above, and used as a basis for the existing fatigue analysis.

b. The number of transients which are comparable to or more severe than the transients evaluated in the stress report Code fatigue usage calculations will be recorded in an operating log book. For those transients which are more severe, available data, such as the metal and fluid temperatures, pressures, flow rates, and other conditions will be recorded in the log book.

c. The number of transient events that exceed the design specification quantity and the number of transient events with a severity greater than that included in the existing Code fatigue usage calculations shall be added. When this sum exceeds the predicated number of design condition events by twenty-five[2], a fatigue usage evaluation of such events will be performed for the affected portion of the RCPB.

H. Records of current individual plant staff members showing qualifications and the completion of training.

I. Records for Environmental Qualification which are covered under the provisions of Specification 6.3.

J. Records of the service lives of all hydraulic and mechanical snubbers noted in 3.6.H.1, including the date at which the service life commences and associated installation and maintenance records.

---

1 See paragraph N-415.2, ASME Section III, 1965 Edition.

2 The Code rules permit exclusion of twenty-five (25) stress cycles from secondary stress and fatigue usage evaluation. (See paragraphs N-412(t)(3) and N-417.10(f) of the Summer 1968 Addenda to ASME Section III, 1968 Edition.)

6.5.1.C (Cont'd)

1. A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions[1], e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

2. A summary description of facility changes, tests or experiments in accordance with the requirements of 10CFR50.59(b). This report may be submitted annually or along with the Updated Safety Analysis Report (UFSAR) updates as required by 10CFR50.71(e).

3. Documentation of all challenges to relief valves or safety valves.

D. Monthly Operating Report

Routine reports of operating statistics, shutdown experience, and a narrative summary of operating experience relating to safe operation of the facility, shall be submitted on a monthly basis to the individual designated in the current revision of Reg. Guide 10.1 no later than the tenth of each month following the calendar month covered by the report.

E. Annual Radiological Environmental Report

1. Routine radiological environmental reports covering the surveillance activities related to the Station operation during the previous calendar year shall be submitted to the NRC before May 1 of each year.

2. The Annual Radiological Environmental Report shall include the following:

   a. A summary of doses to a MEMBER OF THE PUBLIC OFFSITE due to Cooper Nuclear Station aqueous and airborne radioactive effluents, calculated in accordance with methods compatible with the ODAM.

   b. A summary of the results of the land use census required in Specification 4.21.F.2.

[1] This tabulation supplements the requirements of §20.2206 of 10CFR Part 20.